

DEVELOPMENT OF NUPAC 140B 100 TON RAIL/BARGE CASK

PRELIMINARY DESIGN REPORT VOLUME I

APRIL 1990

Prepared for
U.S. Department of Energy
Idaho Operations Office
Under Contract DE-AC07-88ID12700

DISCLAIMER

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

Nuclear Packaging
1010 South 336th Street
Federal Way, WA 98003

MASTER

db
DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

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

DISCLAIMER

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

DISCLAIMER

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

NO FORGIVENESS
12/18 2012

Vol. I
SECTION A
TABLE OF CONTENTS

1.0 GENERAL INFORMATION

1.1 Introduction

1.2 Package Description

1.2.1 Packaging

1.2.2 Operation and Transport

1.2.3 Contents of Packaging

1.3 Appendix

1.3.1 References

1.3.2 Packaging General Arrangement Drawings

1.3.3 Unique Features

1.3.4 Human Factors/Operability

1.3.5 Technical Certification Issues Requiring NRC Resolution

1.3.6 Technical Issues Requiring Experimental Verification

2.0 STRUCTURAL EVALUATION

2.1 Structural Design

2.1.1 Discussion

2.1.2 Design Criteria

2.2 Weights and Centers of Gravity

2.3 Mechanical Properties of Materials

2.4 General Standards for All Packages

2.4.1 Minimum Package Size

2.4.2 Tamper-Indicating Feature

2.4.3 Positive Closure

2.4.4 Chemical and Galvanic Reactions

2.5 Lifting and Tie down Standards for All Packages

2.5.1 Lifting Devices

2.5.2 Tiedown Devices

2.6 Normal Conditions of Transport

- 2.6.1 Heat
- 2.6.2 Cold
- 2.6.3 Reduced External Pressure
- 2.6.4 Increased External Pressure
- 2.6.5 Vibration
- 2.6.6 Water Spray
- 2.6.7 Free Drop
- 2.6.8 Payload Baskets

2.7 Hypothetical Accident Conditions

- 2.7.1 Free Drop
- 2.7.2 Puncture
- 2.7.3 Thermal
- 2.7.4 Immersion - Fissile Material
- 2.7.5 Immersion - All Packages
- 2.7.6 Summary of Damage
- 2.7.7 Payload Baskets

2.8 Special Form

2.9 Fuel Rods

2.10 Appendix

- 2.10.1 Nuclear Packaging Proprietary Drop Programs
- 2.10.2 Drop Program Evaluations
- 2.10.3 Buckling Evaluation
- 2.10.4 Finite Element Analysis
- 2.10.5 Closure Bolt Stress Evaluations

2.11 References

3.0 THERMAL EVALUATION

3.1 Discussion

3.2 Summary of Thermal Properties of Materials

3.3 Technical Specifications of Components

3.4 Thermal Evaluation for Normal Conditions of Transport

- 3.4.1 Thermal Model
- 3.4.2 Maximum Temperatures
- 3.4.3 Minimum Temperatures
- 3.4.4 Maximum Internal Pressures

3.4.5 Maximum Thermal Stresses

3.4.6 Evaluation of Package Performance for Normal
Conditions of Transport

3.5 Hypothetical Accident Thermal Evaluation

3.5.1 Thermal Models

3.5.2 Package Conditions and Environment

3.5.3 Package Temperatures

3.5.4 Maximum Internal Pressures

3.5.5 Maximum Thermal Stresses

3.5.6 Evaluation of Package Performance for
Hypothetical Accident Thermal Conditions

3.6 Appendix

3.6.1 Contract Requirements Regarding Thermal Design Aspects of
NuPac 140-B Cask and Fuel Package per Contract Statement of
Work, Reference (1)

3.6.2 Axisymmetric Cask Thermal Model Input, Element and Nodal
Definitions, and The Normal Condition Nodal Temperatures

3.6.3 Thirty Degree 2-D Model of Cask Wall Input

3.6.4 Quarter 2-D Model of PWR Basket Input

3.6.5 Quarter 2-D Model of BWR Basket Input

3.6.6 Full 2-D Model of Center PWR Fuel Assembly Input

3.6.7 Accident Condition Equilibrium Prior To 30 minute, 1475°F
Hypothetical Thermal Test Condition. Nodal Temperatures From
Axisymmetric Cask Thermal Model.

3.6.8 Comparisons of NuPac 140-B Cask Analytical Thermal Model
Results With The Westinghouse MC-10 Storage Cask Test Data
Given In Reference (13)

3.6.9 ANSYS-PC/THERMAL Element Descriptions Utilized for NuPac
140-B Cask Thermal Models

3.7 References

4.0 CONTAINMENT

4.1 Containment Boundary

- 4.1.1 Containment Vessel
- 4.1.2 Containment Penetrations
- 4.1.3 Seals and Welds
- 4.1.4 Closure

4.2 Requirements for Normal Conditions of Transport

- 4.2.1 Containment of Radioactive Material
- 4.2.2 Pressurization of Containment Vessel
- 4.2.3 Containment Criterion

4.3 Containment Requirements for the Hypothetical Accident Conditions

- 4.3.1 Fission Gas Products
- 4.3.2 Containment of Radioactive Material
- 4.3.3 Containment Criterion

4.4 Special Requirements

4.5 Appendix

- 4.5.1 References

5.0 SHIELDING EVALUATION

5.1 Discussion and Results

5.2 Source Specification

- 5.2.1 Gamma Source
- 5.2.2 Neutron Source

5.3 Model Specification

5.3.1 Description of Radial and Axial Shielding
Configuration

5.3.2 Shield Regional Densities

5.4 Shielding Evaluation

5.5 Appendix

5.5.1 References

5.5.2 ANISN Input Files

5.5.3 ANISN Dose Rates

5.5.4 QAD-CGGP Input Files

6.0 CRITICALITY EVALUATION

6.1 Discussions and Results

6.2 Package Fuel Loading

6.3 Model Specification

6.3.1 Description of the Calculation Model

6.3.2 Package Regional Densities

6.4 Criticality Calculations

6.4.1 Computational Method

6.4.2 Fuel Loading or Other Contents Loading
Optimization

6.4.3 Criticality Results

6.5 Critical Benchmark Experiments

6.6 Appendix

6.6.1 References

6.6.2 Input Decks for Calculations for the NuPac
140-B Cask with BWR Fuel

7.0 OPERATIONAL PROCEDURES

7.1 Introduction

7.2 Cask Handling Procedure at a Reactor

7.2.1 Prerequisites

7.2.2 Cask and Rail Car Receipt

7.2.3 Prepare the Cask for Removal from the Rail Car

7.2.4 Placing the Cask on the Decon Pad

7.2.5 Preparing the Cask for Loading on the Decon Pad

7.2.6 Loading the Cask

7.2.7 Preparing the Cask for Shipment on the Decon Pad

7.2.8 Place the Cask on the Transportation Vehicle

7.2.9 Preparing the Cask for Shipment on the Rail Car

7.2.10 Shipping the Cask

7.3 Procedure for Unloading at the Repository/MRS Using a Hot Cell

7.3.1 Cask Receiving

7.3.2 Cask Preparation on Rail Car

7.3.3 Cask Preparation at the Work Station

7.3.4 Cask Unloading in Hot Cell

7.3.5 Cask Preparation at Work Station

7.3.6 Load Casks on Rail Car

7.3.7 Preparation of Casks for Shipment

7.4 References

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Acceptance Tests

8.1.1 Visual Inspection

8.1.2 Structural and Pressure Tests

8.1.3 Leakage Tests

8.1.4 Polyurethane Foam Test

8.1.5 Lead Inspection and Installation

8.1.6 Thermal Acceptance Tests

8.1.7 Criticality Control Materials Acceptance Tests

8.2 Maintenance Program

- 8.2.1 Structural and Pressure Tests
- 8.2.2 Leak Tests
- 8.2.3 Subsystems Maintenance
- 8.2.4 Valves, Rupture Discs, and Gaskets on the
Containment Vessel
- 8.2.5 Shielding
- 8.2.6 Thermal
- 8.2.7 Criticality Control Materials

8.3 Appendix

- 8.3.1 References

9.0 QUALITY ASSURANCE

9.1 Introduction

9.2 Description of the PNSI, 10 CFR 71, Subpart H, Quality Program

- 9.2.1 Organization
- 9.2.2 Quality Assurance Program
- 9.2.3 Design Control
- 9.2.4 Procurement Document Control
- 9.2.5 Instruction, Procedures, and Drawings
- 9.2.6 Document Control
- 9.2.7 Control of Purchased Materials, Parts and
Components
- 9.2.8 Identification and Control of Materials, Parts,
and Components
- 9.2.9 Control of Special Processes
- 9.2.10 Inspection
- 9.2.11 Test Control
- 9.2.12 Control of Measuring and Testing Equipment
- 9.2.13 Handling, Storage, and Shipping
- 9.2.14 Inspection, Test and Operating Status
- 9.2.15 Non-Conforming Material, Parts, or Components
- 9.2.16 Corrective Action

9.2.17 Quality Assurance Records

9.2.18 Audits

9.3 References

Vol. II
SECTION B

B.1.0 140-B CASK ANCILLARY EQUIPMENT

B.1.1 140-B Cask Lifting Devices

- B.1.1.1 System Requirements
- B.1.1.2 Component Description
- B.1.1.3 Operational Description
- B.1.1.4 140-B Cask Lifting Fixture Structural Analysis

B.1.2 140-B Cask Tiedown Systems

- B.1.2.1 General Description
- B.1.2.2 Operational Details
- B.1.2.3 Analytical Results
- B.1.2.4 Future Recommended Studies
- B.1.2.5 Structural Verification

B.1.3 140-B Cask Uprighting System

- B.1.3.1 General Description
- B.1.3.2 Component Description
- B.1.3.3 Operational Description

B.1.4 140-B Cask Sunshield/Personnel Barrier

- B.1.4.1 System Description
- B.1.4.2 Operational Description

B.1.5 140-B Cask Railcar

- B.1.5.1 General Description
- B.1.5.2 Operational Description

B.1.6 Other 140-B Auxiliary Equipment

- B.1.6.1 Cask Seal Surface Protector Device
- B.1.6.2 Cask Leak Detecting Equipment
- B.1.6.3 Cask Vacuum Drying Equipment
- B.1.6.4 Draining and Inert Cask Equipment
- B.1.6.5 Cask Lid Bolt Torque Wrench System
- B.1.6.6 Other Special Cask Handling Equipment

B.1.7 Appendix - Structural Verification Calculations for the
140-B Cask Critical Lift Fixture

B.1.8 Appendix - Structural Verification Calculations for the
140-B Cask Transport Tiedown System

B.2.0 IMPACT LIMITER POLYURETHANE FOAM HIGH TEMPERATURE PERFORMANCE TEST RESULTS

B.3.0 O-RING SEAL TEST PROGRAM REPORT

SECTION C

C.1.0 SPENT FUEL RAIL SHIPMENT TURNAROUND TIME

C.1.1 General

C.1.1.1 Turnaround Times

C.1.1.2 Bases for Turnaround Time Estimates

C.1.2 Turnaround Time Estimate at a BWR Reactor with Redundant Crane

C.1.2.1 Cask Receiving

C.1.2.2 Cask Preparation on Rail Car

C.1.2.3 Cask Preparation on Decon Pad

C.1.2.4 Load Cask

C.1.2.5 Cask Preparation on Decon Pad

C.1.2.6 Load Casks on Rail Car

C.1.2.7 Preparation of Cask for Shipment

C.1.3 Turnaround Time Estimate at a PWR Reactor with a Single Yoke

C.1.3.1 Cask Receiving

C.1.3.2 Cask Preparation on Rail Car

C.1.3.3 Cask Preparation on Decon Pad

C.1.3.4 Load Cask

C.1.3.5 Cask Preparation on Decon Pad

C.1.3.6 Load Casks on Rail Car

C.1.3.7 Preparation of Cask for Shipment

C.1.4 Turnaround Time Estimate Using a Hot Cell

C.1.4.1 Cask Receiving

C.1.4.2 Cask Preparation on Rail Car

C.1.4.3 Cask Preparation at Work Station

C.1.4.4 Cask Unloading in Hot Cell

- C.1.4.5 Cask Preparation at Work Station
- C.1.4.6 Load Casks on Rail Car
- C.1.4.7 Preparation of Cask for Shipment

C.2.0 SAFETY COMPLIANCE

C.2.1 Introduction

C.2.2 10 CFR 71

- C.2.2.1 General Standards for all Packaging
- C.2.2.2 Structural Standards for Large Quantity Shipping
- C.2.2.3 Criticality Standards for Fissile Material Packages
- C.2.2.4 Evaluation of a Single Package
- C.2.2.5 Standards for Normal Conditions of Transport for a Single Package
- C.2.2.6 Standards for Hypothetical Accident Conditions for a Single Package

C.2.3 49 CFR 173

C.2.3.1 General Packaging Requirements

C.2.4 Basic Components (Safety Related)

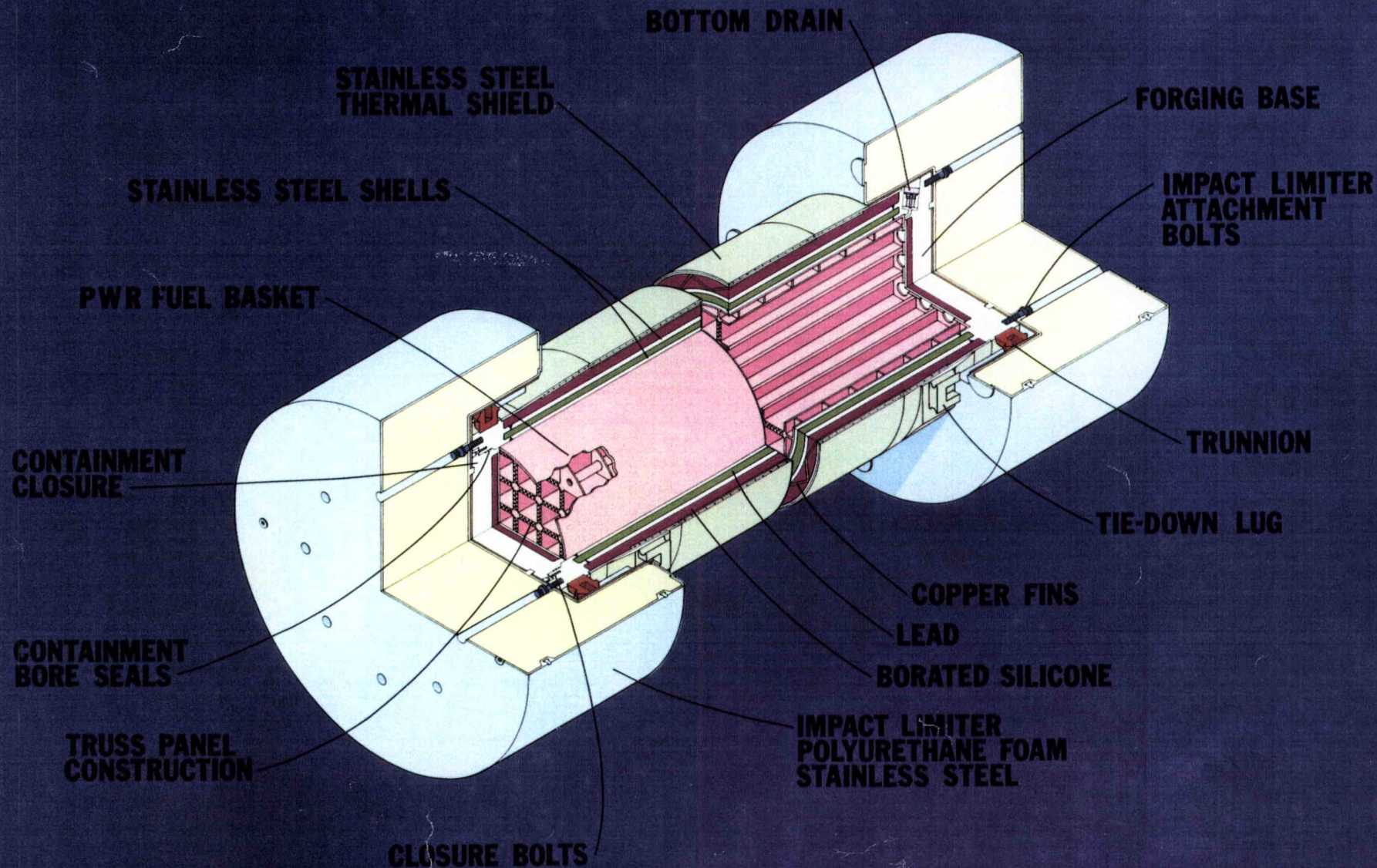
C.3.0 TRADE OFF STUDIES

C.4.0 ESTIMATION OF FABRICATION COSTS

C.5.0 ESTIMATION OF FABRICATION SCHEDULE

C.6.0 RESPONSE TO PRELIMINARY DESIGN REVIEW COMMENTS

NUPAC 140-B RAIL/BARGE SHIPPING CASK



PRELIMINARY DESIGN

1.0 GENERAL INFORMATION

This section of the NuPac 140-B 100 Ton Rail/Barge Shipping Cask Preliminary Design Report (PDR) presents a general introduction to, and description of, the NuPac 140-B Cask and its fuel payload. Terminology used throughout the report is presented in Table 1.0-1. To better visualize the major components of the NuPac 140-B Cask, renderings of the cask are included in Appendix 1.3.2.

1.1 Introduction

The NuPac 140-B Cask, Model: NuPac 140-B, is being designed by Nuclear Packaging, Inc., to meet or exceed all NRC and Department of Transportation regulations governing the shipment of radioactive material. Specifically the Cask is being developed as a safe means of transporting spent light-water-reactor (LWR) fuels from existing and proposed reactor facilities to a repository and/or a monitored retrievable storage (MRS) facility. The primary transportation mode is by railroad, although the shipping package is designed to be transported by barge and by truck shipment on a special overweight basis for short distances. This feature allows the servicing of reactor sites and other facilities which lack direct railroad access. The cask design has been optimized for weight and maximum safety during the loading, transport, and unloading operations. The NuPac 140-B Cask is comprised of an outer cask shell which provides a primary containment boundary and acts as an environmental barrier. Polyurethane foam filled energy absorbers (impact limiters) are attached to each end of the outer cask to limit the consequences of damage which may occur during normal operation and hypothetical accident events. The cask has a design service life of 25 years.

The NuPac 140-B Cask is a rail/barge cask designed to transport up to 21 PWR or 52 BWR fuel assemblies per shipment. The NuPac 140-B Cask provides a single level of testable containment for the fuel during both normal and hypothetical accident conditions, per the requirements of 10 CFR 71.63(b) (1), Reference 1.3.1.1.

Authorization shall be sought from the USNRC to allow shipment of the NuPac 140-B Cask and its spent fuel payload as a Type B(M), Fissile Class III, package per the definitions delineated in 10 CFR 71.4, Reference 1.3.1.1.

TABLE 1.0-1
TERMINOLOGY AND NOTATION

Package:

The Packaging with its radioactive contents, or Payload, as presented for transportation (10 CFR 71.4), Reference 1.3.1.1. Within this PDR, the Package is denoted as the NuPac 140-B Cask and Fuel Package.

Packaging:

The assembly of components necessary to ensure compliance with packaging requirements (10 CFR 71.4), Reference 1.3.1.1. Within this PDR, the Packaging is denoted as the NuPac 140-B Cask.

NuPac 140-B Cask:

Cask, Basket, energy absorbing impact limiters, and internal fuel spacers.

Payload: PWR configuration:

- o Maximum number of assemblies - Twenty-one (21).
- o Maximum initial enrichment with 0 MWD/MTU burnup - 3.2 w/o U-235; with 18,000 MWD/MTU burnup - 4.5 w/o U-235.
- o Maximum burnup - 35,000 MWD/MTU.
- o Minimum decay time - 10 years.

TABLE 1.0-1 (CONTINUED)
TERMINOLOGY AND NOTATION

Payload: BWR configuration:

- o Maximum number of assemblies - Fifty-two (52).
- o Maximum initial enrichment with 0 MWD/MTU burnup - 4.5 w/o U-235.
- o Maximum burnup - 30,000 MWD/MTU.
- o Minimum decay time - 10 years.

Spacers:

Stainless steel spacers located in the fuel basket to account for differing lengths of the various fuel assemblies.

Cask:

The assembly (comprised of a lid and body) which provides the primary level of containment for the Payload. The Cask surrounds the Payload and provides appropriate gamma and neutron shielding capabilities.

Impact Limiters:

Polyurethane foam filled energy absorbers which cover each end of the Cask during transport.

1.2 Package Description

The NuPac 140-B Cask and Fuel Package is being designed in accordance with the criteria of Federal Regulations 10 CFR 71 and 49 CFR 172. Drawings of the NuPac 140-B Cask are contained in Appendix 1.3.2.

1.2.1 Packaging

1.2.1.1 Cask

The cask is a lead and borated silicone shielded vessel. The cask provides the primary containment boundary for the payload and also acts as an environmental barrier. A separate inner container is not required for this package since the payload is reactor fuel elements (reference 10 CFR 71.63(b)(1)), Reference 1.3.1.1. During shipment the cask is protected at each end by energy absorbing impact limiters which consist of stainless steel skins filled with medium density, closed-cell polyurethane foam. The impact limiters also provide thermal insulation which protects seal areas during the hypothetical accident thermal event. The payload thermal energy is dissipated to the environment through a passive heat transfer system. Copper fins embedded in the neutron shield provide the thermal conduction path through the relatively low thermally conductive neutron shield material. The overall arrangement of the cask and design details of the baskets are presented in Section 1.4. The cask, baskets and impact limiters are more fully described in the following sections.

The cask is fabricated primarily of ASTM A240 Type XM-19 austenitic stainless steel. Non-stainless members include the cast lead shielding, elastomeric O-rings, borated silicone neutron shield, copper heat transfer fins, and ASTM A320, Grade L43, carbon steel closure bolts. Thirty-two (32) closure bolts secure the cask closure lid to the cask body. The body of the cask is constructed of concentric stainless steel shells which sandwich the cast lead gamma shielding. The outer portion of the cask body is covered with the neutron shield and copper fins. A thin stainless steel shell covers the neutron shield and prevents communication of the neutron shield area with the environment.

The containment boundary consists of the 8.50 inch thick stainless steel cask bottom, 57.00 inch inside diameter stainless steel inner shell, the upper (or lid end) forging, and the 8.50 inch thick cask closure lid complete with two O-ring seals and the cask closure bolts. The cask cavity is 57.00 inches in diameter by 180.5 inches long.

The 1.5 inch thick, 66.25 inch inside diameter stainless steel outer shell and the 3.5 inch thickness of lead shielding which fills the annular space between the inner and outer shells, are non-containment. An eighth inch (0.125) thick ASTM A240 Type XM-19 stainless steel thermal shield surrounds the cask neutron shield. This shield is spaced out from the neutron shield by a 10 gauge (0.134 inch diameter) wire wrap on a 3.25 inch pitch spacing.

The cask is 203.5 inches in length and has a maximum diameter of 81.8 inches. The weight of the cask is approximately 132,000 pounds; 50,000 pounds of which is the 3.5 inch thick cast lead shielding.

The cask has 10 significant protrusions; namely, 6 stainless steel lifting trunnions and 4 tiedown lugs.

1.2.1.2 Baskets

There are two removable baskets for the NuPac 140-B Cask. Either PWR or BWR fuel bundles can be accommodated through the use of the removable baskets and spacers.

1.2.1.2.1 PWR Basket

The 21 cell PWR basket is made up of light weight jointed truss core structural panels fabricated primarily from ASTM A240 Type XM-19 stainless steel. The only non-stainless material is the neutron absorbing material and the copper which serves as a heat transfer medium (see Section 6.3.1.1). The panels are fabricated such that a flux trap is built into each wall of the fuel cell. Water is contained in this area of the basket during wet fuel loading/unloading and serves, along with the neutron absorbing material, to ensure that the K_{eff} of the system remains less than 0.95 including all calculational uncertainties at a 95% confidence level. The water is drained

from the cask cavity, including the flux trap, prior to shipment. The copper, used for thermal conductivity, is not allowed to contact the fuel pool water.

1.2.1.2.2 BWR Basket

The 52 cell BWR basket is fabricated using panels which have dovetail joints along the entire length of the panel. The panels are joined at the edges by use of a tie bar. The panels are made with ASTM A240 Type XM-19 stainless steel, a neutron absorbing material (which provides criticality control during wet fuel loading/unloading) and copper which serves as a heat transfer medium (see Section 6.3.1.2). Unlike the PWR basket, a flux trap between the fuel assembly cells is not required. The water is drained from the cask cavity and thus the fuel cell prior to shipment. As in the PWR basket, the copper is not allowed to contact the fuel pool water.

1.2.1.3 Impact Limiters

The impact limiter skins is fabricated from ASTM A240, Type XM-19 austenitic stainless steel. Non-stainless steel members include the closed-cell polyurethane foam, the copper heat transfer liner, and the ASTM A320, Grade L43, impact limiter attachment bolts. The polyurethane foam has a nominal density of 15 pounds per cubic foot. Eight (8) bolts secure each impact limiter to the outer cask body.

The outside diameter of the impact limiter is 128 inches and the length is 70 inches. The maximum weight of each impact limiter is 11,200 pounds.

1.2.1.4 Receptacles, Valves, Testing, and Sampling Ports

The cask lid has a seal test port and a combination vent/sampling port. The seal test port accesses the volume between the O-ring seals on the lid. After assembly, containment O-ring integrity is verified prior to shipping the loaded package. The vent port is opened at the loading end of shipment for ease of lid installation and can be used as a sampling port at the receiving end of shipment to determine contamination levels.

1.2.1.5 Heat Dissipation

The payload thermal energy is dissipated from the NuPac 140-B Cask by passive heat transfer means. There are no special devices utilized for the transfer or dissipation of heat. The package maximum design capacity is 11,000 thermal watts. A more detailed discussion of the package thermal characteristics is provided in Section 3.0.

1.2.1.6 Coolants

There are no coolants utilized within the package other than the normal transportation atmosphere of inert gasses.

1.2.1.7 Protrusions

There are no outer or inner protrusions on the package other than the six (6) external lifting trunnions and the four (4) tiedown lugs. Refer to the General Arrangement Drawings in Appendix 1.3.2 for more detail.

1.2.1.8 Lifting and Tiedown Devices

Of the six trunnions located on the exterior of the package, the four located on the upper, or closure end, are for lifting the cask and the two located at the bottom are used to support the cask during the uprighting operation. Attachment of the cask to the rail car is by the tiedown lugs only. A more detailed discussion of the package lifting and tiedown features is provided in Section B.1.

1.2.1.9 Operating Pressure

The maximum package normal operating internal pressure is 284 psig.

1.2.1.10 Pressure Relief System

A rupture disc is provided in the cask lid to preclude catastrophic damage to the package should inadvertent over pressurization (in excess of Regulatory requirements) occur. Since the rupture disc requires a containment penetration, leak test interfacing is provided to verify containment integrity.

Plastic pipe plugs are included on the impact limiter skins to preclude excessive pressurization of the impact limiters in a hypothetical accident thermal event.

1.2.1.11 Shielding

A minimum of 3.5 inches of lead, 2.75 inches of steel, 5.88 inches of borated silicone, and 0.6 inches of copper is maintained between the fuel and exterior surface of the package for the attenuation of radiation. Further detail is provided in Section 5.0.

Radiation levels shall not exceed the requirements delineated in 10 CFR 71.47 or 49 CFR 173.441.

1.2.1.12 Gross Weight

The maximum NuPac 140-B Fuel Spent Shipping Cask weight when fully loaded is approximately 206,600 pounds (BWR fuel). A summary of overall component weights, detailed in Table 2.2-1, is listed below:

PWR Configuration:

Component	Weight (lbs)
Cask	132,087
Basket	14,225
Fuel	31,815
<u>Impact Limiter</u>	<u>22,376</u>
Total	<u>200,503</u>

BWR Configuration:

Component	Weight (lbs)
Cask	132,087
Basket	16,508
Fuel & Channels	35,568
<u>Impact Limiters</u>	<u>22,376</u>
Total	<u>206,539</u>

1.2.1.13 Criticality

In both the PWR and the BWR cases, the use of criticality control material is necessary to maintain subcriticality during wet loading, unloading or accident conditions. For the PWR configuration, a water flux trap is included in the basket design. In addition, a neutron absorbing material is built into the walls of the fuel storage cells. For the BWR configuration the use of a flux trap is not required, however neutron absorbing material is also included in the fuel cell walls. See Section 6.0 for a detailed description of the criticality analysis.

1.2.1.14 Quality Assurance

Quality Assurance at Nuclear Packaging is an integral part of work performed from design to manufacturing and use. During the course of the preliminary design effort on the 140-B Cask the USNRC approved QA program was invoked to

assure that all items affecting quality were controlled in accordance with the approved program requirements.

Quality Instructions (QI's) were issued which provide detailed instructions as to how the QA program was to be implemented.

Formal training for personnel performing quality related functions was instituted to inform and familiarize personnel with the QA requirements. This quality training program is an on-going effort which has been expanded to include quality requirements which are specific to the 140-B cask.

NuPac utilizes a Project Planning Form to define responsibilities and illustrate such specifics as interface controls, design inputs, design verification and detailed quality requirements.

Computer code verification has been performed in accordance with NuPac approved QIs, as will the various computer codes and backup data which will be utilized in preparation of the final design report.

The preliminary design was verified through a review by engineering personnel who are competent in the applicable field of design and were not responsible for preparation of the original data. Management assessments were performed during the preliminary design phase to assure that the various disciplines involved operated in accordance with contract and procedural requirements.

A graded approach to quality has been established at NuPac through documented procedures which correspond with the requirements of the Quality Level classification List prepared for this project. This approach also represents NuPac's procurement policy. Quality requirements are included with procurement documents via a Quality Engineering review and approval. The Approved Suppliers List is also based on the graded quality system and is continually reviewed and revised as appropriate.

Control of measuring and test equipment, qualification of special processes/operators and engineering testing are controlled in accordance with the NuPac QA Program. Testing for scoping purposes was performed by the test engineer and verified by QA.

Quality documents (plans, instructions, procedures, drawings, etc.) are processed by Document Control in accordance with approved procedures.

The quality program will continue to be enforced during all phases of the project.

1.2.2 Operation and Transport

The NuPac 140-B Cask is not considered to be operationally complex. All operational features are reasonably apparent from an inspection of the General Arrangement Drawings provided in Appendix 1.3.2. A complete operating manual will be written and provided to each cask user. In addition, Nuclear Packaging will provide instruction on cask handling prior to use. The cask is mounted horizontally on a cradle which is attached to a specially designed railcar during normal transport. The skid is designed to allow transport of the cask by barge or heavy haul special permit trailer. This multi-mode configuration permits the use of the NuPac 140-B Cask at those reactor sites which have no direct rail access. See Section B.1 for a complete description of the railcar system.

1.2.3 Contents of Packaging (Payload)

The baskets have been designed to maintain the geometric configuration of the PWR and BWR fuels during transportation in the NuPac 140-B Cask.

1.2.3.1 Fuel Assembly Descriptions

The fuel assemblies which can be shipped in the NuPac 140-B Cask are as follows:

PWR Spent Fuels

<u>Assembly Manufacturer</u>	<u>Size</u>	<u>Array Version</u>
Westinghouse	14 x 14	Std/ZCA
Westinghouse	14 x 14	OFA
Westinghouse	14 x 14	Std/ZCB
Westinghouse	14 x 14	Std/SC
Westinghouse	14 x 14	Model C
Westinghouse	15 x 15	Std/ZC
Westinghouse	15 x 15	OFA
Westinghouse	15 x 15	Std/SC

Westinghouse	17 x 17	Std
Westinghouse	17 x 17	OFA
Westinghouse	17 x 17	Vant 5
Babcock and Wilcox	15 x 15	Mark B
Babcock and Wilcox	15 x 15	St.Stl.
Babcock and Wilcox	15 x 15	Mark BZ
Babcock and Wilcox	17 x 17	Mark C
Combustion Engineering	14 x 14	Std
Combustion Engineering	14 x 14	Ft. Cal.
Combustion Engineering	15 x 15	Palis.
Combustion Engineering	16 x 16	Onofre
Combustion Engineering	16 x 16	Lucie 2
Combustion Engineering	16 x 16	ANO2
Combustion Engineering	16 x 16	SYS80
Combustion Engineering	15 x 16	Yankee
Exxon / ANF	14 x 14	WE
Exxon / ANF	14 x 14	CE
Exxon / ANF	14 x 14	TOP ROD
Exxon / ANF	15 x 15	WE
Exxon / ANF	15 x 15	CE
Exxon / ANF	15 x 16	WE
Exxon / ANF	17 x 17	WE

BWR Spent fuels

<u>Assembly Manufacturer</u>	<u>Size</u>	<u>Array Version</u>
General Electric	6 x 6	DRES-1
General Electric	6 x 6	HUM. BAY
General Electric	7 x 7	/2,3;V1
General Electric	7 x 7	/2,3;V2
General Electric	7 x 7	/4,5
General Electric	7 x 7	HUM. BAY
General Electric	8 x 8	/2.3
General Electric	8 x 8	/4,5;V1
General Electric	8 x 8	/4,5;V2
Westinghouse	8 x 8	QUAD+
Exxon / ANF	6 x 6	GE
Exxon / ANF	6 x 6	HUM. BAY
Exxon / ANF	7 x 7	GE
Exxon / ANF	8 x 8	JP-3
Exxon / ANF	8 x 8	JP-4,5
Exxon / ANF	9 x 9	JP-3
Exxon / ANF	9 x 9	JP-4,5
Exxon / ANF	10 x 10	AC

The following fuel assemblies can be shipped in the NuPac 140-B Cask with a redesigned basket. (Cell size must be increased to approx. 7.0 inches).

Exxon / ANF	(BWR)	9 x 9	BRP
Exxon / ANF	(BWR)	11 x 11	GE
General Electric	(BWR)	9 x 9	BRP
General Electric	(BWR)	11 x 11	BRP

No determination has been made as to the suitability of the NuPac 140-B Cask to ship the following fuels.

Babcock & Wilcox	(PWR)	14 x 14
Westinghouse	(PWR)	15 x 16
Westinghouse	(PWR)	13 x 13
Gulf United	(PWR)	17 x 17
Allis Chalmers	(BWR)	10 x 10
Nuclear Fuel	(BWR)	9 x 9
United Nuclear	(BWR)	6 x 6

The following fuels cannot be shipped in the NuPac 140-B Cask due to the increased length of the fuel bundle.

Combustion Engineering	(PWR)	14 x 14 XL
Westinghouse	(PWR)	17 x 17 XLR

1.2.3.2 Maximum Contents Weight

The maximum number of PWR fuel assemblies that may be shipped in a NuPac 140-B Cask at any one time on a single rail car is limited to twenty-one (21), for a maximum package fuel payload weight of 31,815 pounds.

The maximum number of BWR fuel assemblies that may be shipped in a NuPac 140-B Cask at any one time on a single rail car is limited to fifty-two (52) for a maximum package fuel payload weight of 35,568 pounds, including the flow channels.

1.2.3.3 Maximum Decay Heat

The maximum decay heat with either 21 PWR, or 52 BWR, fuel assemblies shall be 11,000 thermal watts.

Appendix 1.3

- 1.3.1 References
- 1.3.2 Drawings
- 1.3.3 Unique Features
- 1.3.4 Human Factors/Operability
- 1.3.5 Technical Certification Issues Requiring NRC Resolution
- 1.3.6 Technical Issues Requiring Experimental Verification

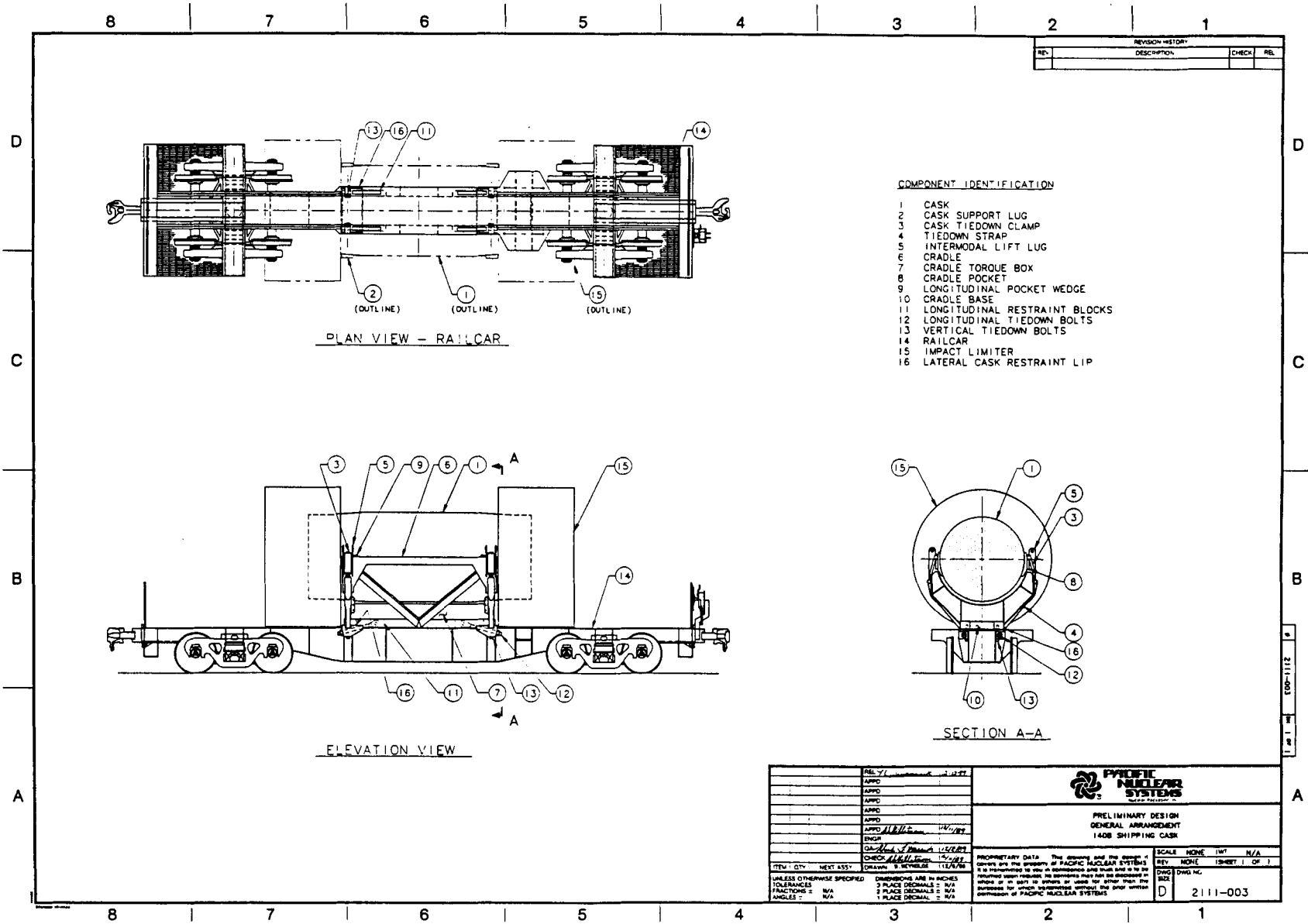
Appendix 1.3.1
References

1.3.1 References

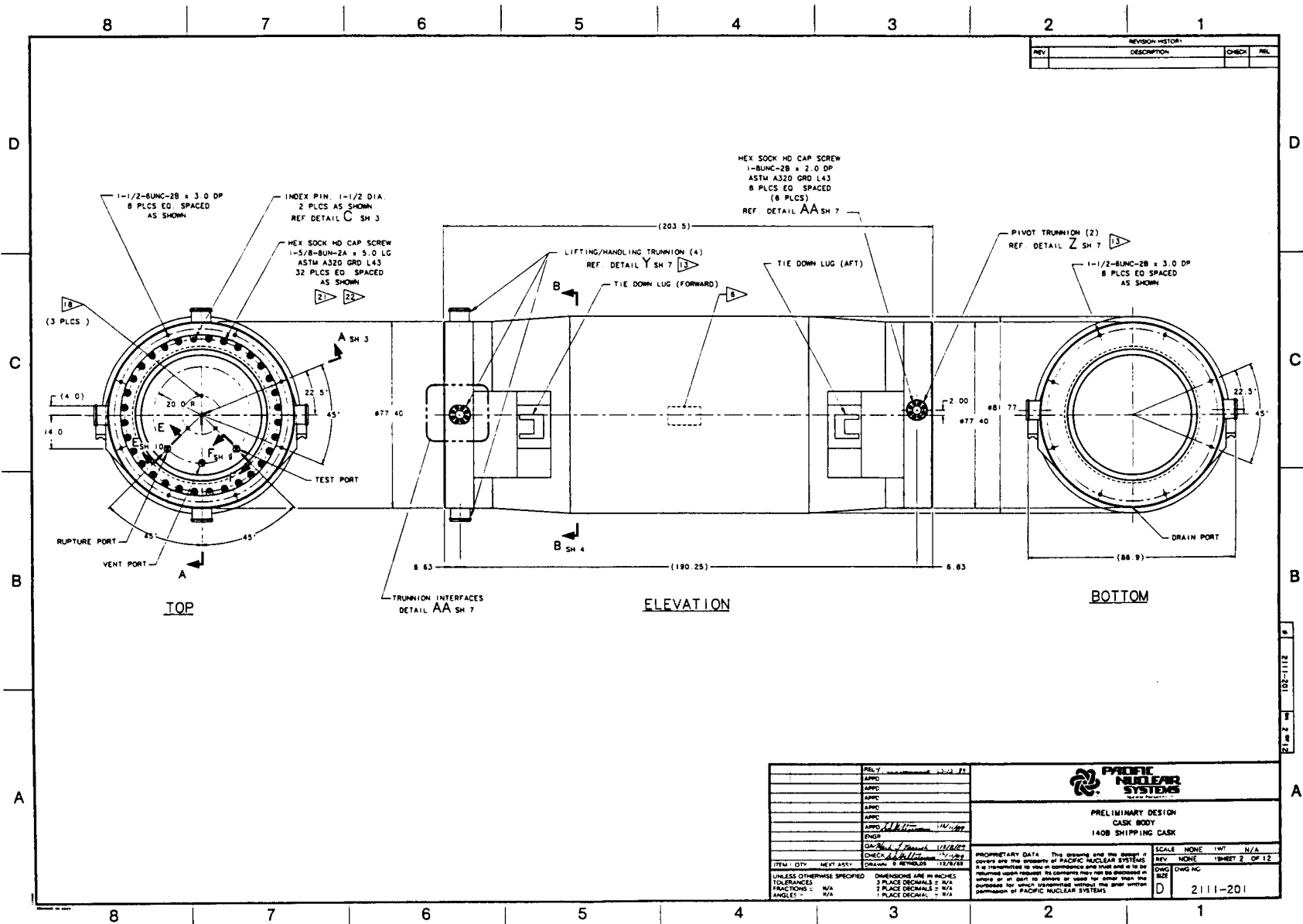
- 1.3.1.1 Title 10, Code of Federal Regulations, Part 71, (10 CFR 71),
Packaging and Transportation of Radioactive Materials, August 24,
1983.

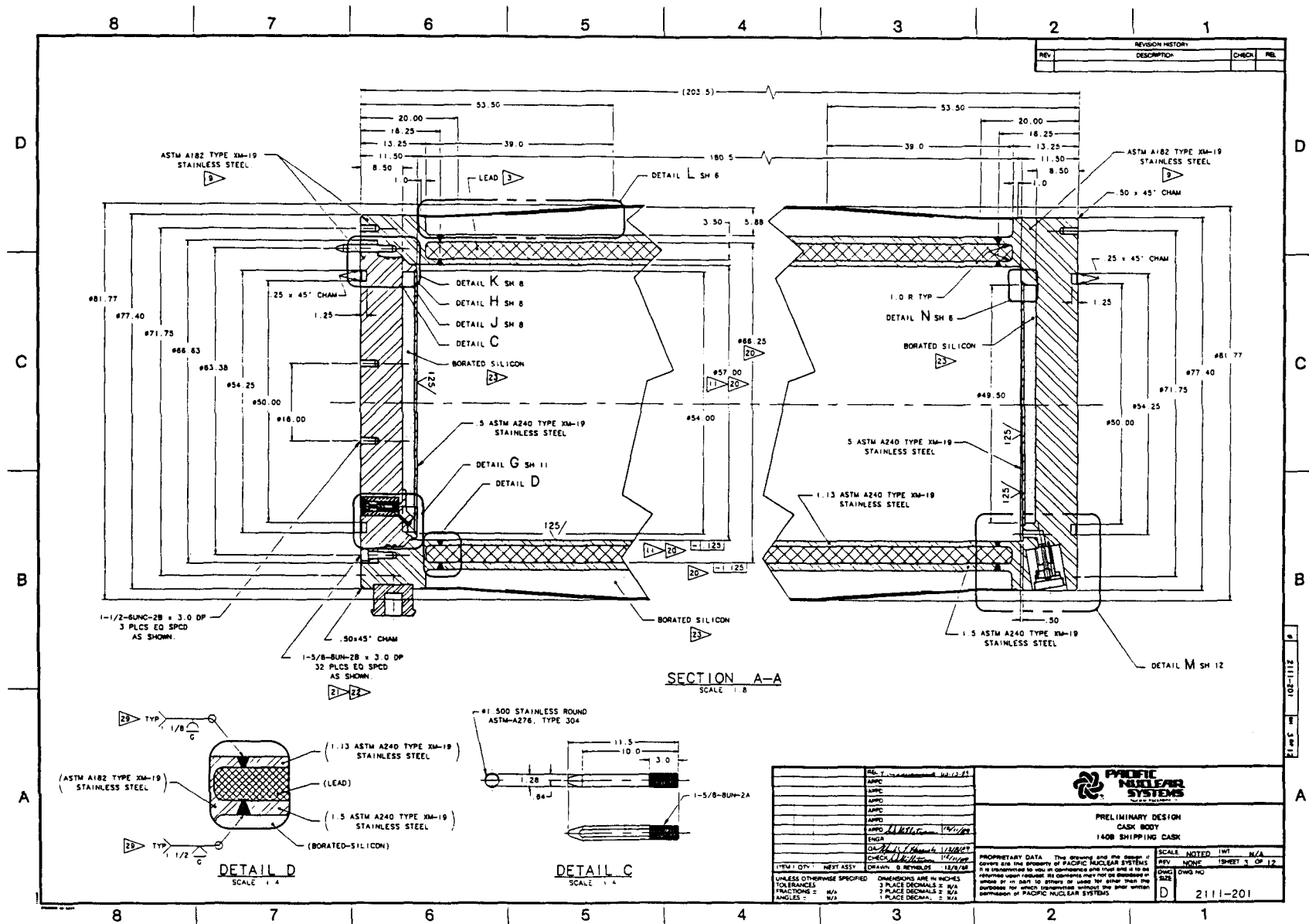
Appendix 1.3.2

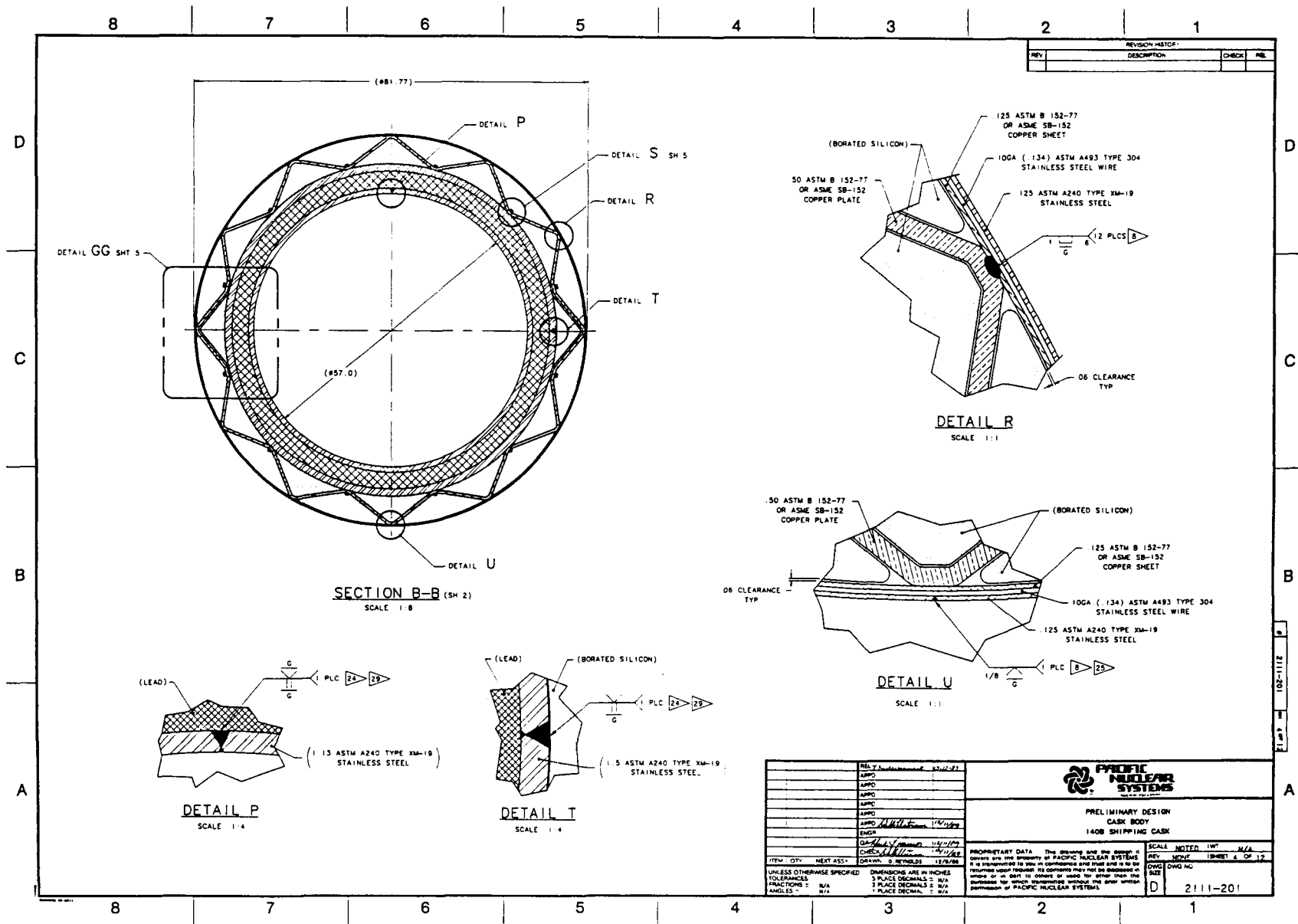
Packaging General Arrangement Drawings



		8	7	6	5	4	3	2	1																
D	1	INTERPRET DRAWING PER ANSI Y-14.5																							
	2	ALL WELDING PROCEDURES AND PERSONNEL SHALL BE QUALIFIED IN ACCORDANCE WITH AWS D1.1 OR ASME CODE, SECTION IX. (ALL STAINLESS STEEL MATERIAL WELDING SHALL BE PERFORMED PER ASME CODE, SECTION IX.) WELD PROCEDURES AND WELDER QUALIFICATIONS SHALL BE AVAILABLE FOR AUDIT OR REVIEW																							
	3	LEAD PER FEDERAL SPECIFICATION QQ-171a, GRADE A OR C.																							
	4	LEAD SHALL BE CAST IN PLACE																							
	5	THREADS PER ANSI B1.1 1980 EDITION																							
	6	IDENTIFICATION CASK SHALL BE IDENTIFIED IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 71.85 (c)																							
	7	ALL WELDS SHALL BE VISUALLY EXAMINED IN ACCORDANCE WITH AWS D1.1, SECTION 8.15.1.																							
	8	WELDS SHALL BE LIQUID PENETRANT INSPECTED ON ROOT AND FINAL PASS IN ACCORDANCE WITH ASME CODE, SECTION III, DIVISION 1, SUBSECTION NB, ARTICLE NB-5000 AND SECTION V, ARTICLE 6.																							
	9	ULTRASONIC INSPECTION SHALL BE PERFORMED ON WELDS AND ALL FORGINGS IN ACCORDANCE WITH ASME CODE, SECTION III, DIVISION 1, SUBSECTION NB, ARTICLE NB-5000 AND SECTION V, ARTICLE 5.																							
	C	10	RADIOGRAPHIC INSPECTION SHALL BE PERFORMED IN ACCORDANCE WITH ASME CODE SECTION III, DIVISION 1, SUBSECTION NB, ARTICLE NB-5000 AND SECTION V, ARTICLE 2. RADIOGRAPHIC INSPECTION SHALL BE PERFORMED ON THE FOLLOWING: (a) ALL CASTINGS (b) ALL LONGITUDINAL SEAM WELDS OF CASK INNER AND OUTER SHELLS (c) ALL CIRCUMFERENTIAL SEAM WELDS OF CASK INNER AND OUTER SHELLS																						
11		DIMENSIONAL INSPECTION OF THE CASK INNER SHELL FOR STRAIGHTNESS AND ROUNDNESS SHALL BE REQUIRED BEFORE AND AFTER LEAD POUR																							
12		MATERIAL ASTM-A182, TYPE 304-18 STAINLESS STEEL, FORGINGS ASTM-A240, TYPE 304-18 STAINLESS STEEL PLATE, SHEET AND STRIP. ASTM-A479, TYPE 304-18 STAINLESS STEEL BARS AND SHAPES. ASTM-A240, TYPE 304 STAINLESS STEEL PLATE, SHEET AND STRIP. ASTM-A276, TYPE 304 STAINLESS STEEL BARS AND SHAPES. ASTM-A479, TYPE 304 STAINLESS STEEL BARS AND SHAPES. ASTM-B132-77 OR ASME SB-152, COPPER SHEET AND PLATE.																							
13		CASK TRANSPORTS SHALL BE LOAD TESTED TO 150% OF THE MAXIMUM WORKING LOAD.																							
14		DESIGN OPERATING PRESSURE IS 400 PSIG. CASK CONTAINMENT BOUNDARIES SHALL BE SUBJECTED TO AN INTERNAL TEST PRESSURE EQUAL TO A MINIMUM OF 150% OF THE MAXIMUM NORMAL OPERATING PRESSURE PER 10CFR 71.85(b).																							
15		CONTAINMENT BOUNDARIES SHALL BE LEAK TESTED TO DEMONSTRATE A LEAK RATE NOT TO EXCEED 1×10^{-7} ATMOSPHERE CUBIC CENTIMETERS PER SECOND																							
16		OPTIONAL: HARD CHROME OR NICKEL PLATE SURFACE TO A THICKNESS OF .0005 MINIMUM / .001 MAXIMUM HARD CHROME SHALL BE IN ACCORDANCE WITH QQ-C-320, CLASS 2C NICKEL PLATING SHALL BE IN ACCORDANCE WITH MIL-C-28074A.																							
17		AREAS INDICATED MAY BE REFABRICATED AND REPLACED IN THE EVENT OF DAMAGE.																							
18		INDICATED HOLES ARE PROVIDED FOR THE INSTALLATION OF THREADED DEVICES WHICH ESTABLISH THE INTERFACE FOR LIFTING AND HANDLING THE OUTER CASK LID. EACH LOCATION SHALL BE STENCILED OR OTHERWISE LABELED WITH THE FOLLOWING WARNING: "LID LIFT ONLY".																							
B		19	LEAD SHIELDING INTEGRITY SHALL BE DETERMINED BY GAMMA SCAN.																						
	20	CASK, INNER AND OUTER SHELLS SHALL COMPLY WITH THE TOLERANCE REQUIREMENTS OF ASME CODE, SECTION III, DIVISION 1, SUBSECTION NE, ARTICLE NE-4220.																							
	21	FASTENER SHALL BE CADMIUM PLATED IN ACCORDANCE WITH MIL-C-8837, TYPE II, CLASS I OR FED SPEC QQ-P-4182, TYPE II, CLASS I.																							
	22	TORQUE CASK LID BOLTS (1-5/8" - 8UN) TO 1,200 - 1,450 FT-LBS (LUBRICATED)																							
	23	NEUTRON SHIELDING MATERIAL BORATED SILICONE																							
	A	24	LONGITUDINAL WELD JOINTS ON INNER AND OUTER SHELLS SHALL BE ORIENTED SUCH THAT THEY ARE A MINIMUM OF 22 1/2° APART IN RELATIVE LOCATION																						
		25	MANUFACTURER SHALL DEFINE THE NUMBER AND POSITION OF INDICATED WELDS																						
		26	MATERIAL FLUORELASTOMER DUREMETER 70																						
		27	INDICATED SURFACES SHALL BE SANDBLASTED TO WHITE METAL IN ACCORDANCE WITH SSPC-SP-5. SURFACES SHALL THEN BE CLEANED AND COATED WITH COLOR BLACK																						
		28	ALL DIMENSIONS ARE NOMINAL. TOLERANCES DO NOT APPLY AT PRELIMINARY DESIGN STAGE																						
29		ALL CONTAINMENT SHELL WELDS SHALL BE FULL PENETRATION WELDS AND BE INSPECTED BY RADIOGRAPH																							
<table border="1" style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th colspan="2">REVISION HISTORY</th> <th>CHECK</th> <th>REL.</th> </tr> <tr> <th>REV.</th> <th>DESCRIPTION</th> <th></th> <th></th> </tr> </thead> <tbody> <tr> <td> </td> <td> </td> <td> </td> <td> </td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> </tr> </tbody> </table>									REVISION HISTORY		CHECK	REL.	REV.	DESCRIPTION											
REVISION HISTORY		CHECK	REL.																						
REV.		DESCRIPTION																							
<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td colspan="2" style="text-align: center;"> PACIFIC NUCLEAR SYSTEMS INCORPORATED </td> <td colspan="2" style="text-align: center;"> PRELIMINARY DESIGN CASK BODY 140B SHIPPING CASK </td> </tr> <tr> <td colspan="2"> PROPRIETARY DATA: This drawing and the design it covers are the property of PACIFIC NUCLEAR SYSTEMS. It is transmitted to you in confidence and shall not be reproduced or used in any way for other than the purposes for which it was prepared without the written permission of PACIFIC NUCLEAR SYSTEMS. </td> <td colspan="2"> <table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>SCALE: NONE</td> <td>UNIT: N/A</td> </tr> <tr> <td>REV: NONE</td> <td>SHEET 1 OF 12</td> </tr> <tr> <td>DWG NO: 2111-201</td> <td></td> </tr> </table> </td> </tr> </table>									 PACIFIC NUCLEAR SYSTEMS INCORPORATED		PRELIMINARY DESIGN CASK BODY 140B SHIPPING CASK		PROPRIETARY DATA: This drawing and the design it covers are the property of PACIFIC NUCLEAR SYSTEMS. It is transmitted to you in confidence and shall not be reproduced or used in any way for other than the purposes for which it was prepared without the written permission of PACIFIC NUCLEAR SYSTEMS.		<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>SCALE: NONE</td> <td>UNIT: N/A</td> </tr> <tr> <td>REV: NONE</td> <td>SHEET 1 OF 12</td> </tr> <tr> <td>DWG NO: 2111-201</td> <td></td> </tr> </table>		SCALE: NONE	UNIT: N/A	REV: NONE	SHEET 1 OF 12	DWG NO: 2111-201				
 PACIFIC NUCLEAR SYSTEMS INCORPORATED		PRELIMINARY DESIGN CASK BODY 140B SHIPPING CASK																							
PROPRIETARY DATA: This drawing and the design it covers are the property of PACIFIC NUCLEAR SYSTEMS. It is transmitted to you in confidence and shall not be reproduced or used in any way for other than the purposes for which it was prepared without the written permission of PACIFIC NUCLEAR SYSTEMS.		<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>SCALE: NONE</td> <td>UNIT: N/A</td> </tr> <tr> <td>REV: NONE</td> <td>SHEET 1 OF 12</td> </tr> <tr> <td>DWG NO: 2111-201</td> <td></td> </tr> </table>		SCALE: NONE	UNIT: N/A	REV: NONE	SHEET 1 OF 12	DWG NO: 2111-201																	
SCALE: NONE	UNIT: N/A																								
REV: NONE	SHEET 1 OF 12																								
DWG NO: 2111-201																									
<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>ITEM</td> <td>QTY</td> <td>NEXT ASSY</td> <td>CHECKED BY</td> <td>DATE</td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> </table>									ITEM	QTY	NEXT ASSY	CHECKED BY	DATE												
ITEM	QTY	NEXT ASSY	CHECKED BY	DATE																					
<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>UNLESS OTHERWISE SPECIFIED</td> <td>DIMENSIONS ARE IN INCHES</td> </tr> <tr> <td>TOLERANCES</td> <td>3 PLACE DECIMALS ± N/A</td> </tr> <tr> <td>FRACTIONS ±</td> <td>2 PLACE DECIMALS ± N/A</td> </tr> <tr> <td>ANGLES ±</td> <td>1 PLACE DECIMAL ± N/A</td> </tr> </table>									UNLESS OTHERWISE SPECIFIED	DIMENSIONS ARE IN INCHES	TOLERANCES	3 PLACE DECIMALS ± N/A	FRACTIONS ±	2 PLACE DECIMALS ± N/A	ANGLES ±	1 PLACE DECIMAL ± N/A									
UNLESS OTHERWISE SPECIFIED	DIMENSIONS ARE IN INCHES																								
TOLERANCES	3 PLACE DECIMALS ± N/A																								
FRACTIONS ±	2 PLACE DECIMALS ± N/A																								
ANGLES ±	1 PLACE DECIMAL ± N/A																								

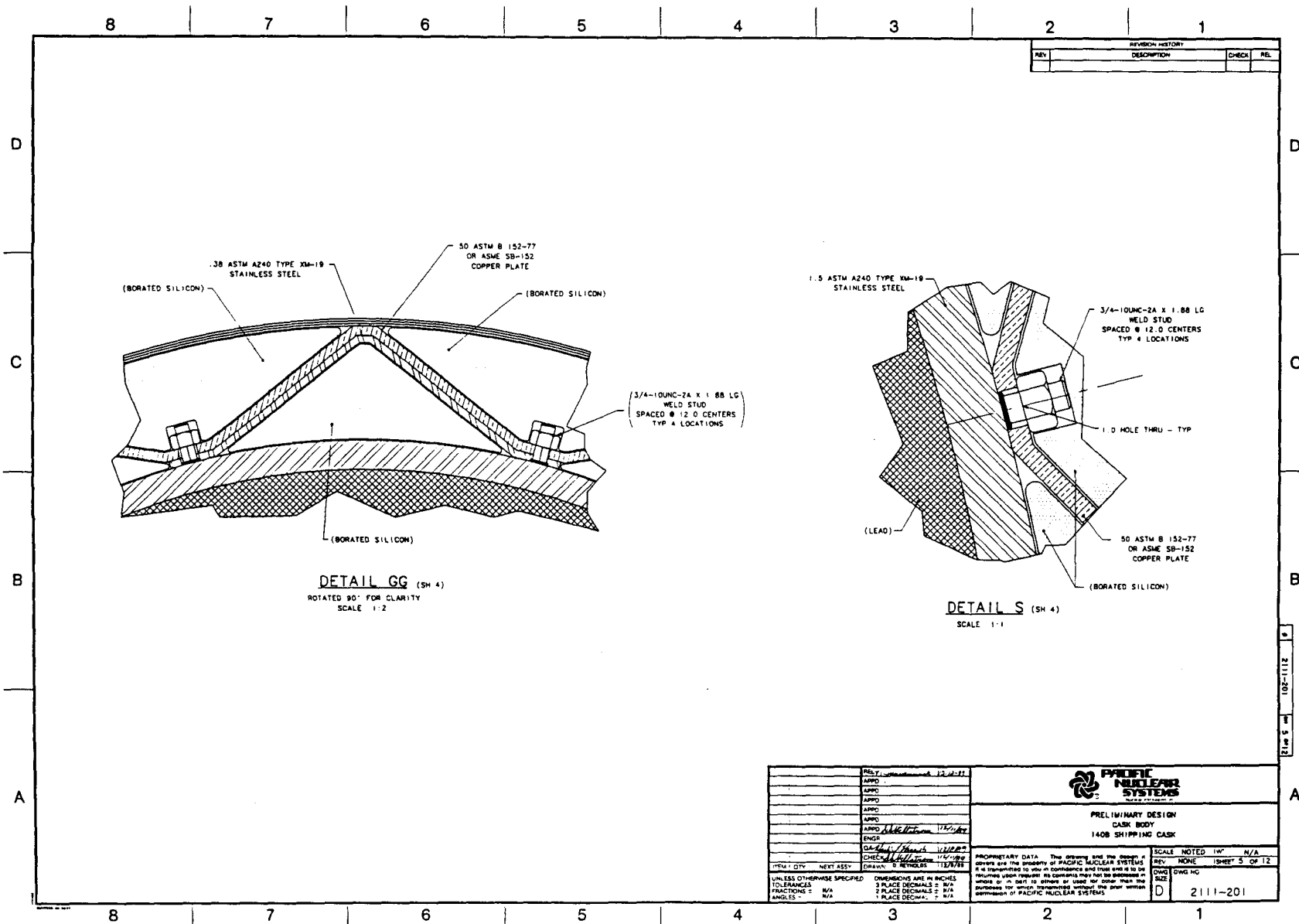






REVISION HISTORY			
REV	DESCRIPTION	CHECK	DATE

ONLY <i>[Signature]</i> 12/1/79 APPRO <i>[Signature]</i> 12/1/79 APPRO <i>[Signature]</i> 12/1/79 APPRO <i>[Signature]</i> 12/1/79 APPRO <i>[Signature]</i> 12/1/79 ENGR <i>[Signature]</i> 12/1/79	PACIFIC NUCLEAR SYSTEMS PRELIMINARY DESIGN CASK BODY 1408 SHIPPING CASK	
ITEM QTY NEXT ASS CHECKED <i>[Signature]</i> 12/1/79 DRAWN <i>[Signature]</i> 12/1/79 UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES 3 PLACE DECIMALS = 3/16" 2 PLACE DECIMALS = 1/8" 1 PLACE DECIMAL = 1/4" ANGLES = 30°	PROPRIETARY DATA: The drawings and the design covers are the property of PACIFIC NUCLEAR SYSTEMS. It is intended for use in connection with the design and construction of the cask body and shall not be reproduced or used for any other purpose without the prior written permission of PACIFIC NUCLEAR SYSTEMS.	SCALE NOTED 1/4"=1'-0" REV NONE 15 SHEET 4 OF 12 DWG CHG NG SIZE D 2111-201

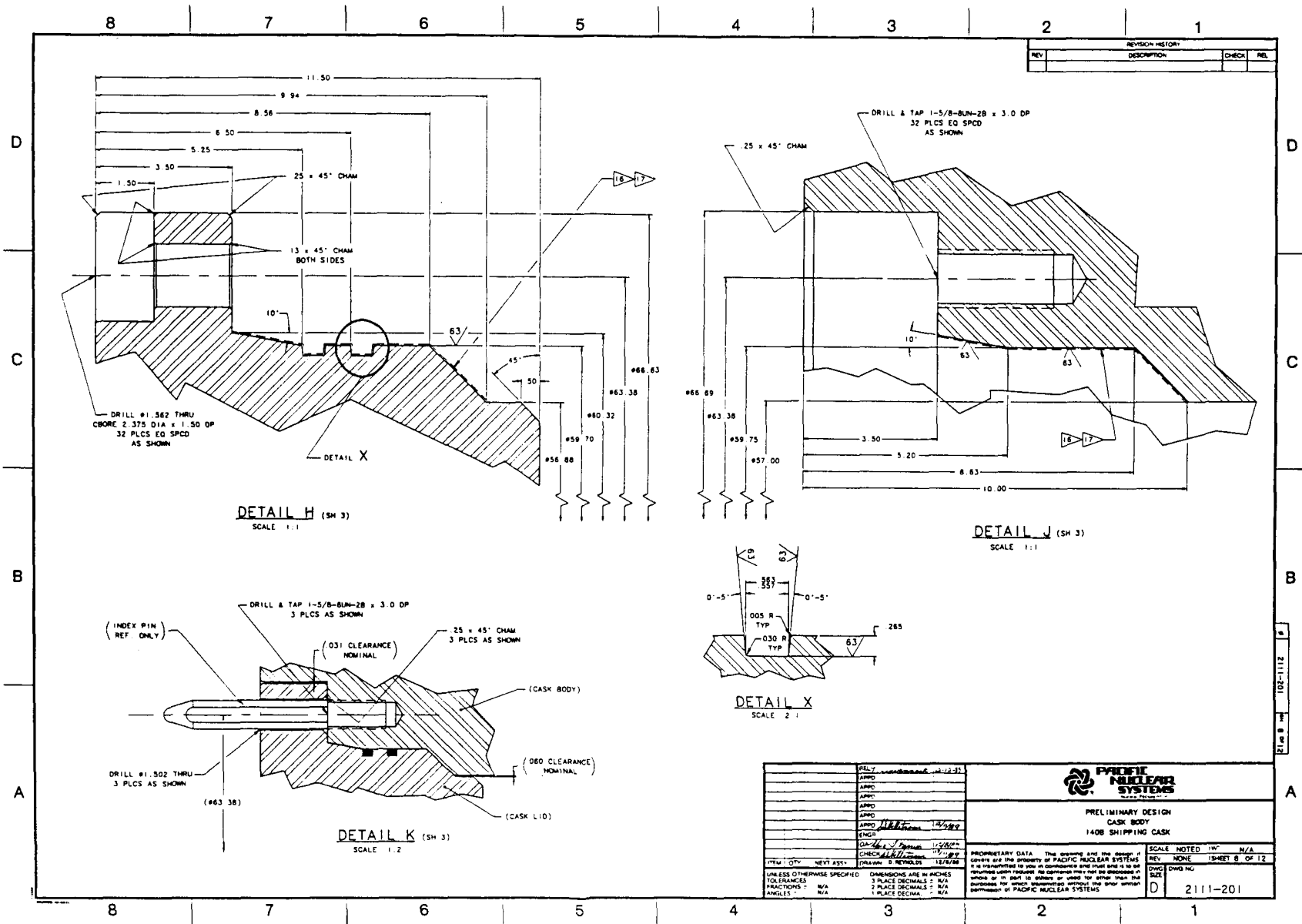


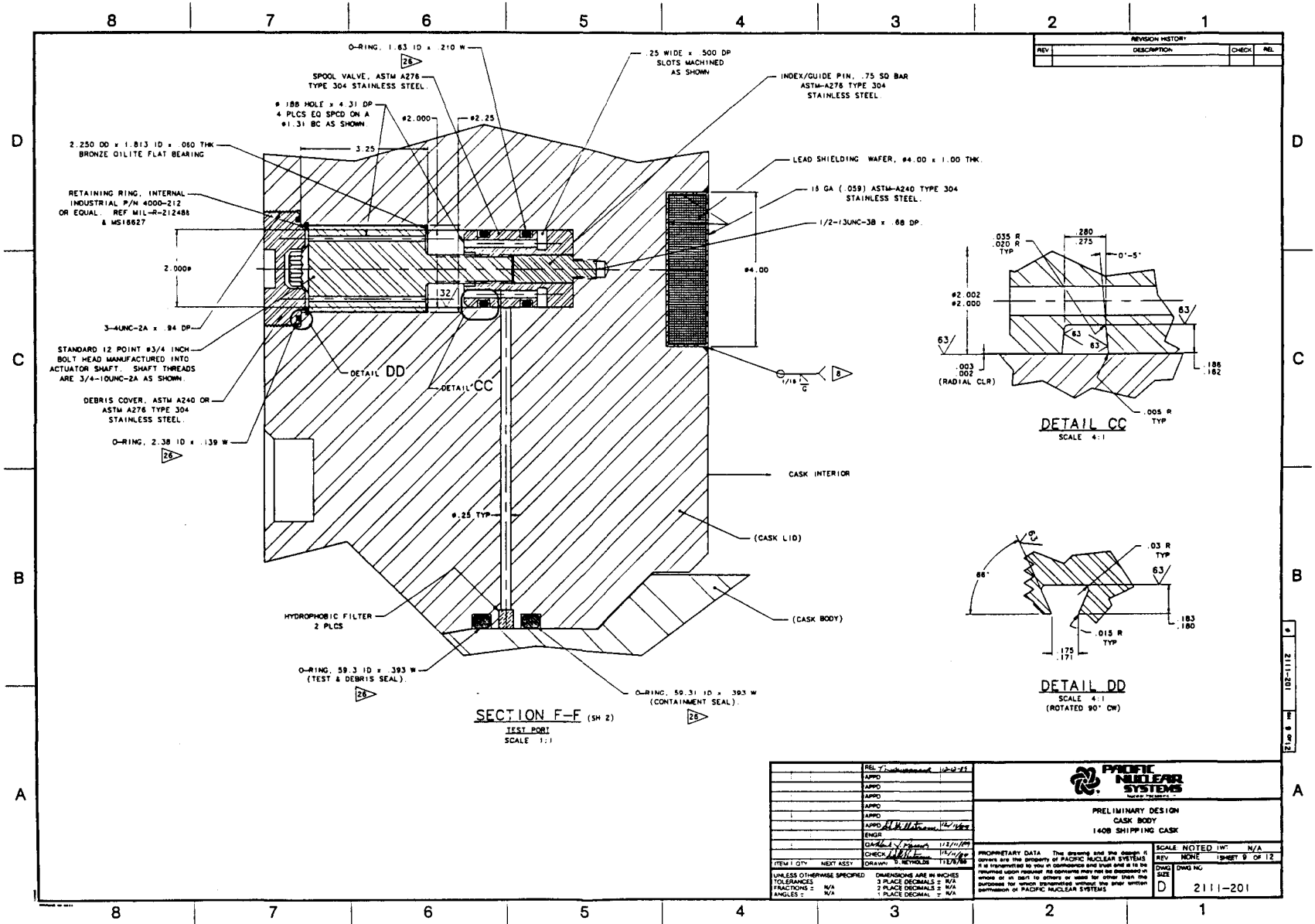
REVISION HISTORY		
REV	DESCRIPTION	REL

DESIGNED: <i>[Signature]</i> DRAWN: <i>[Signature]</i> CHECKED: <i>[Signature]</i> APP'D: <i>[Signature]</i> APP'D: <i>[Signature]</i> APP'D: <i>[Signature]</i> APP'D: <i>[Signature]</i> APP'D: <i>[Signature]</i> APP'D: <i>[Signature]</i> ENGR: <i>[Signature]</i> QA: <i>[Signature]</i> CHECKED: <i>[Signature]</i> DATE: 12/1/81	PACIFIC NUCLEAR SYSTEMS PRELIMINARY DESIGN CASK BODY 140B SHIPPING CASK SCALE NOTED 1/4" N/A REV. NONE DWG NO. 2111-201 SHEET 5 OF 12

UNLESS OTHERWISE SPECIFIED
 DIMENSIONS ARE IN INCHES
 TOLERANCES
 FRACTIONS: N/A
 ANGLES: N/A
 DECIMALS: N/A
 DECIMALS: N/A
 DECIMALS: N/A

PROPRIETARY DATA: The drawing and the design it covers are the property of PACIFIC NUCLEAR SYSTEMS. It is loaned to you in confidence and must not be reproduced, stored in a retrieval system, or used for any other purpose without the written permission of PACIFIC NUCLEAR SYSTEMS.



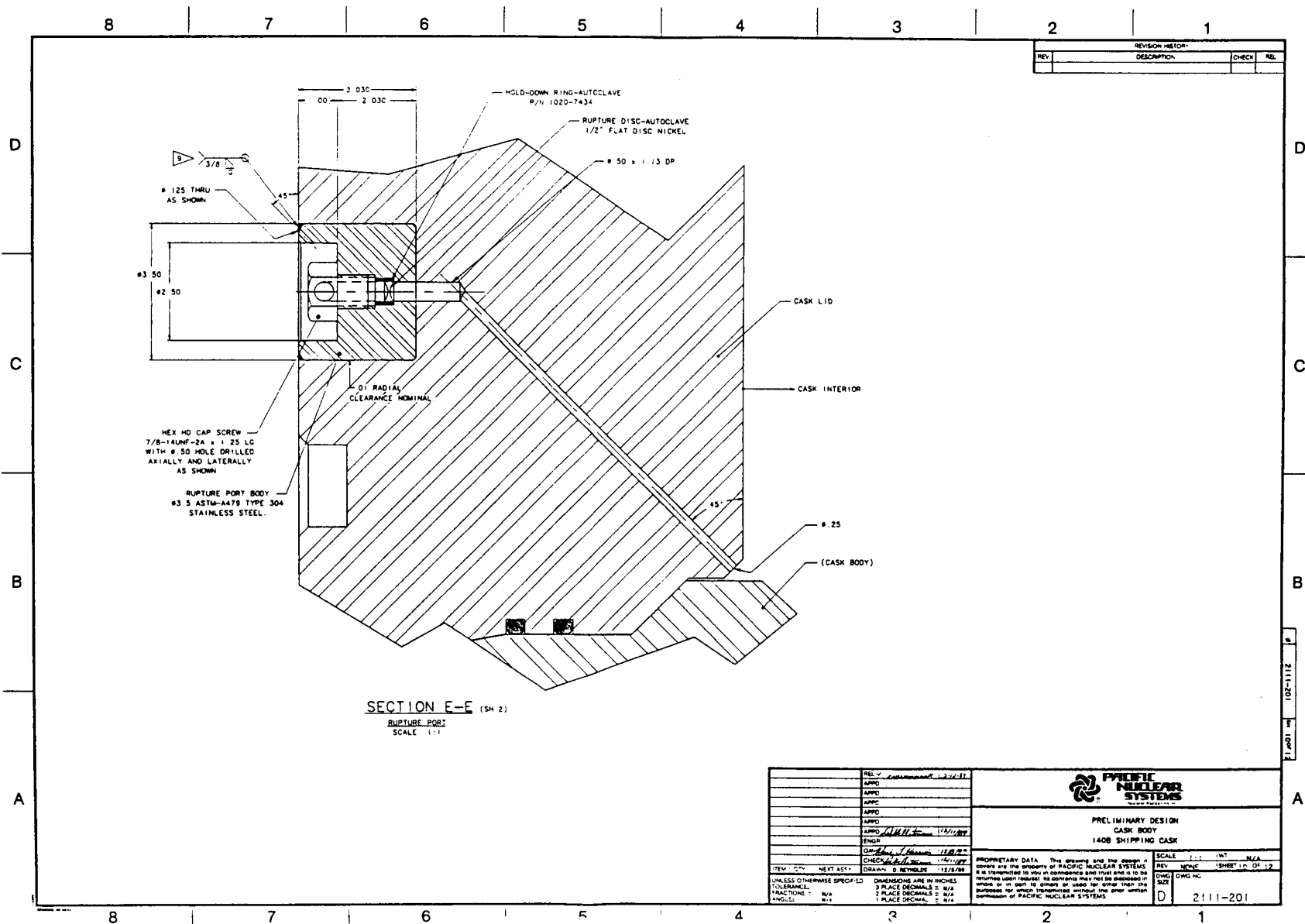


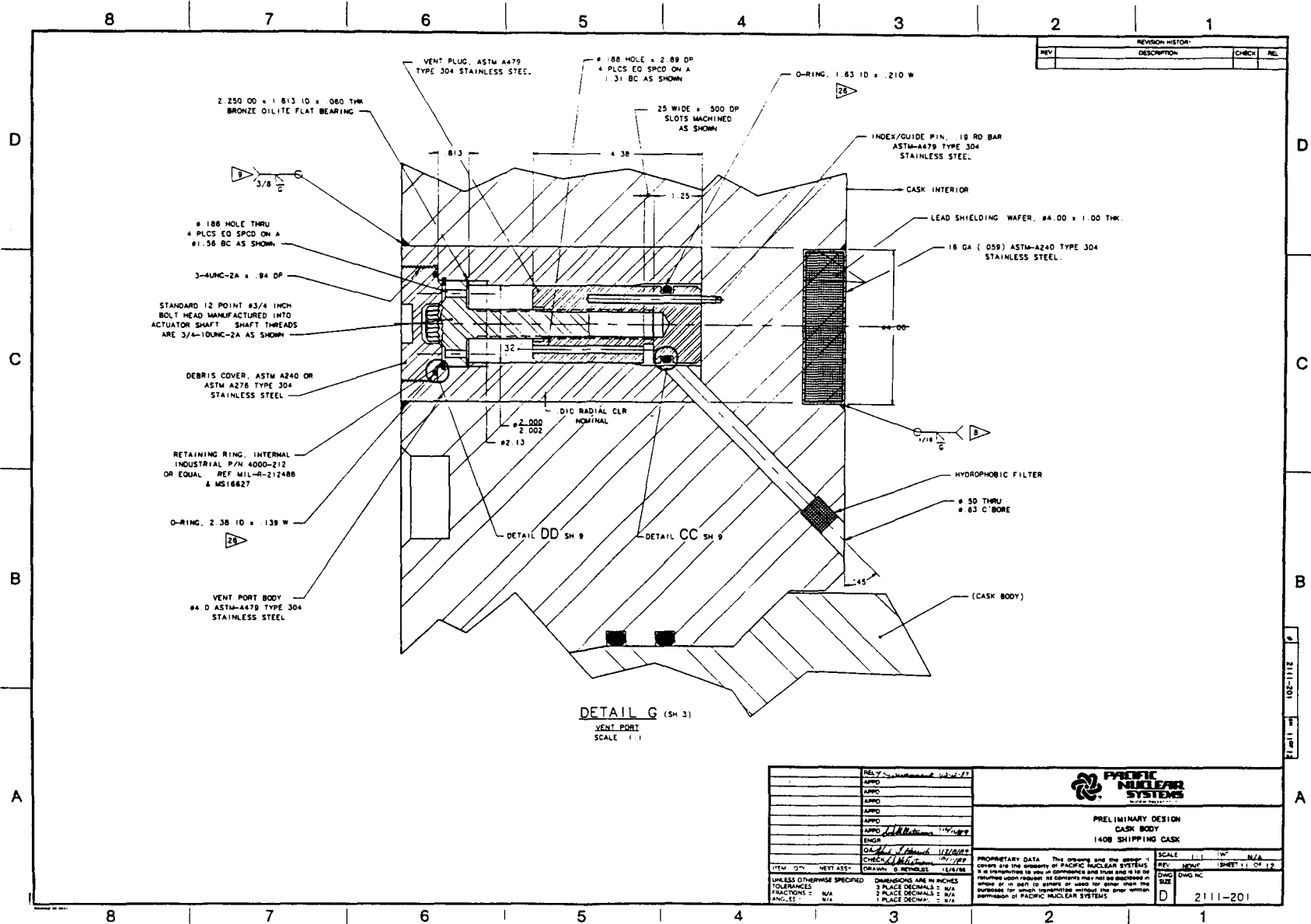
REVISION HISTORY REV. DESCRIPTION CHECK REL.	
1	
2	
3	
4	
5	
6	
7	
8	
9	
10	
11	
12	
13	
14	
15	
16	
17	
18	
19	
20	
21	
22	
23	
24	
25	
26	
27	
28	
29	
30	
31	
32	
33	
34	
35	
36	
37	
38	
39	
40	
41	
42	
43	
44	
45	
46	
47	
48	
49	
50	
51	
52	
53	
54	
55	
56	
57	
58	
59	
60	
61	
62	
63	
64	
65	
66	
67	
68	
69	
70	
71	
72	
73	
74	
75	
76	
77	
78	
79	
80	
81	
82	
83	
84	
85	
86	
87	
88	
89	
90	
91	
92	
93	
94	
95	
96	
97	
98	
99	
100	

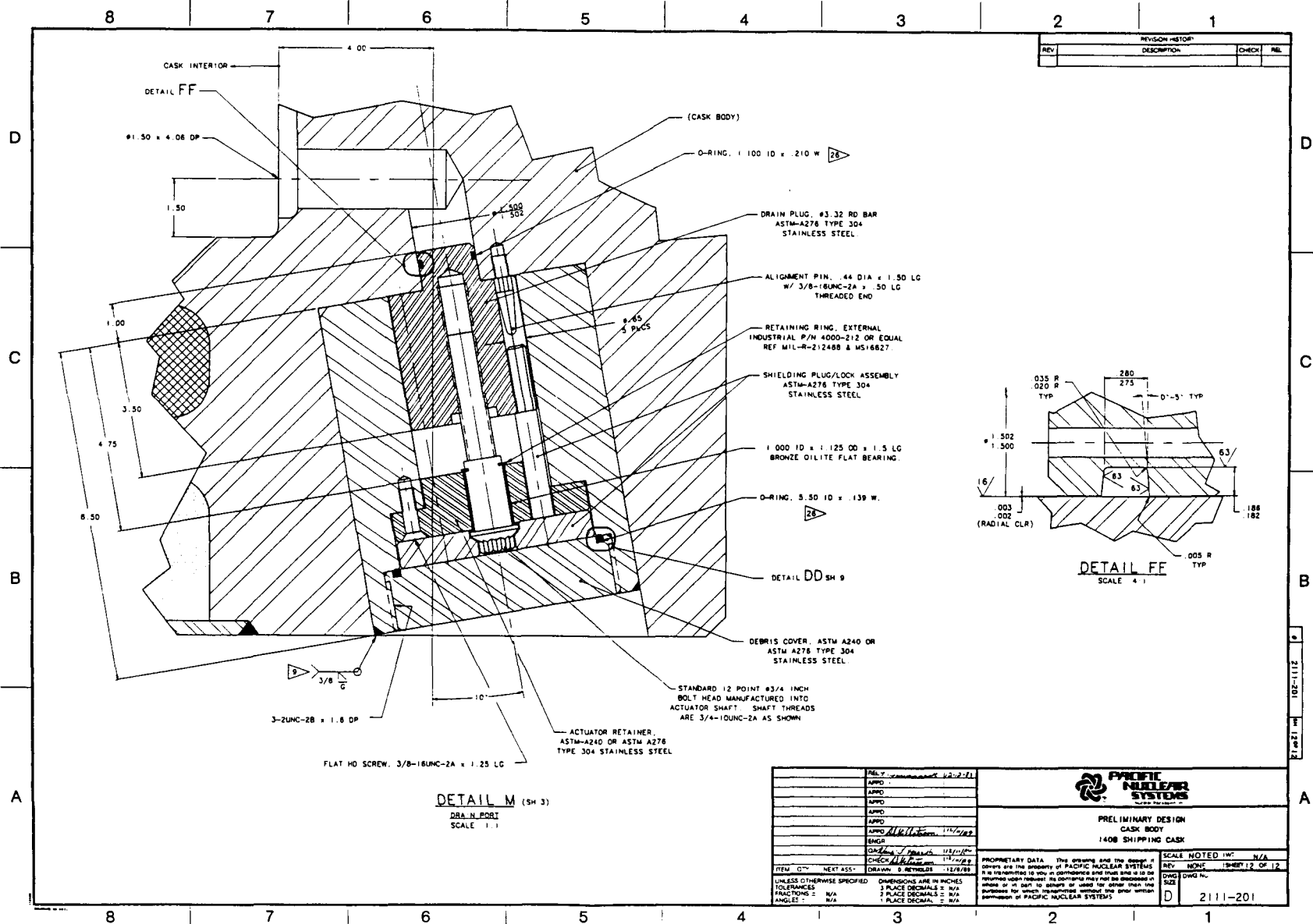


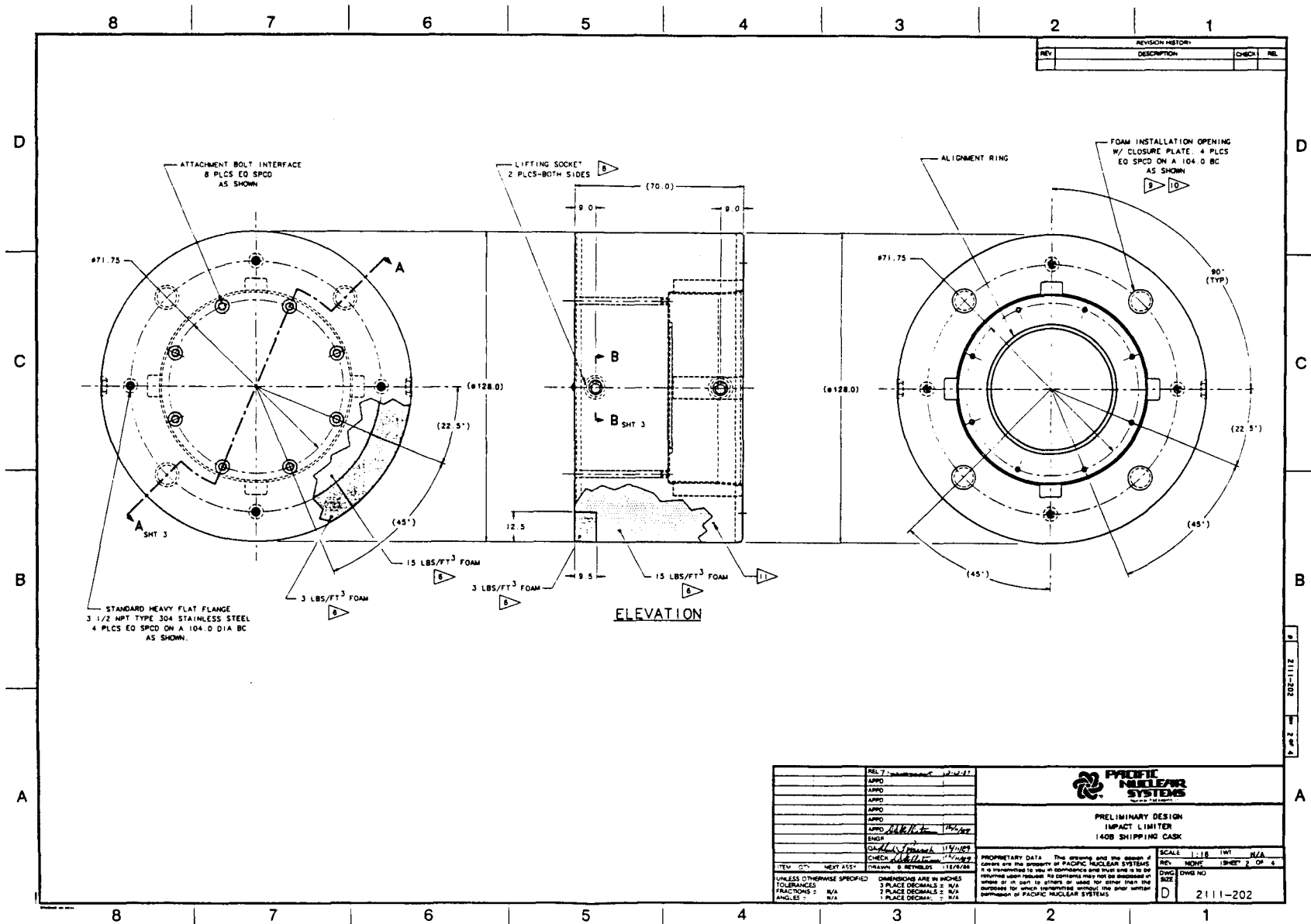
PRELIMINARY DESIGN
CASK BODY
1400 SHIPPING CASK

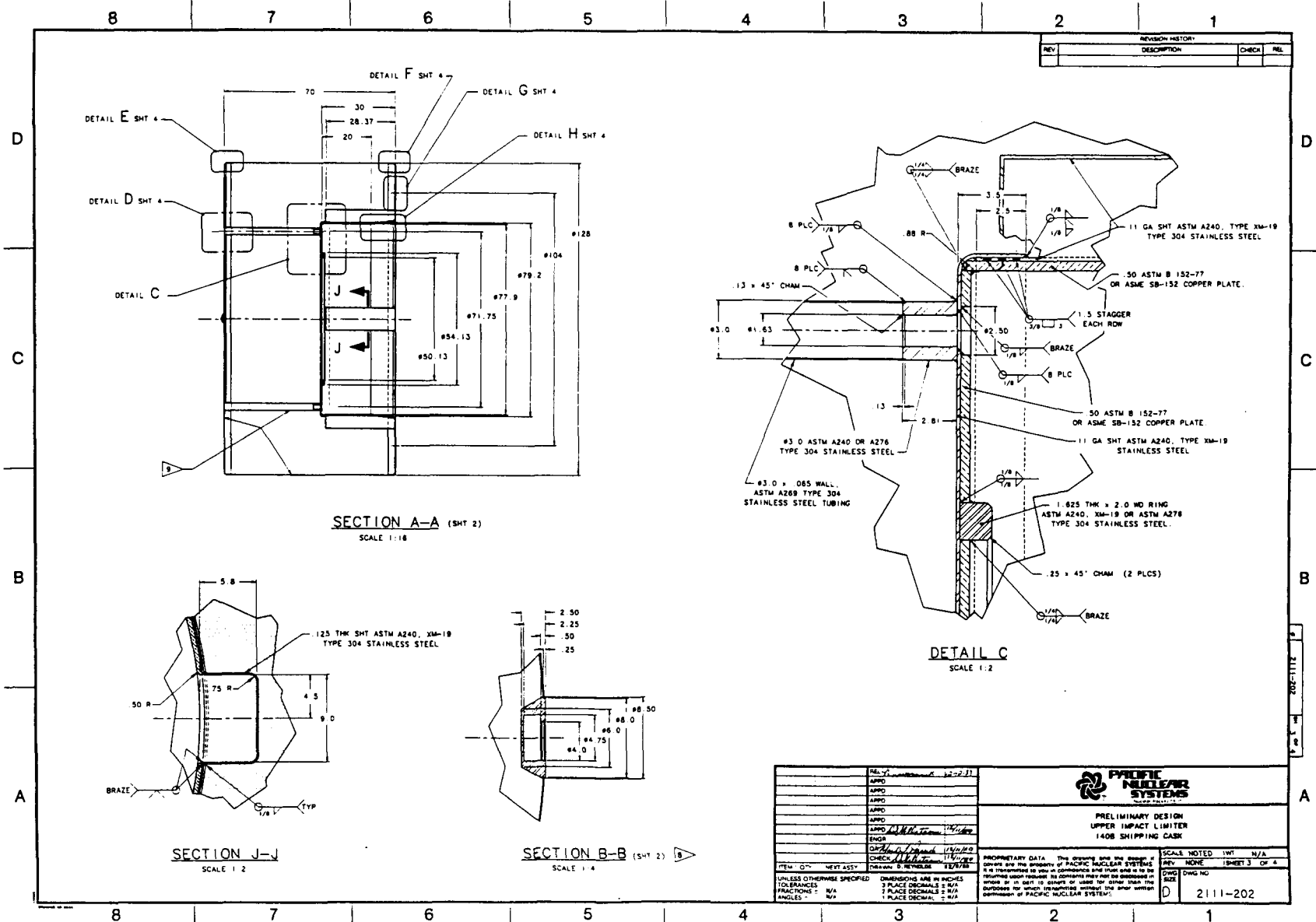
PROPRIETARY DATA: The drawings and the design of the cask are the property of PACIFIC NUCLEAR SYSTEMS. It is represented to you in confidence and that it is to be returned upon request. No contents may not be disclosed in whole or in part to others or used for other than the purposes for which they were created. See letter written to you by PACIFIC NUCLEAR SYSTEMS.	SCALE NOTED: 1/4" = 1" N/A REV. MORE: 12/1/78 Dwg No: 2111-201

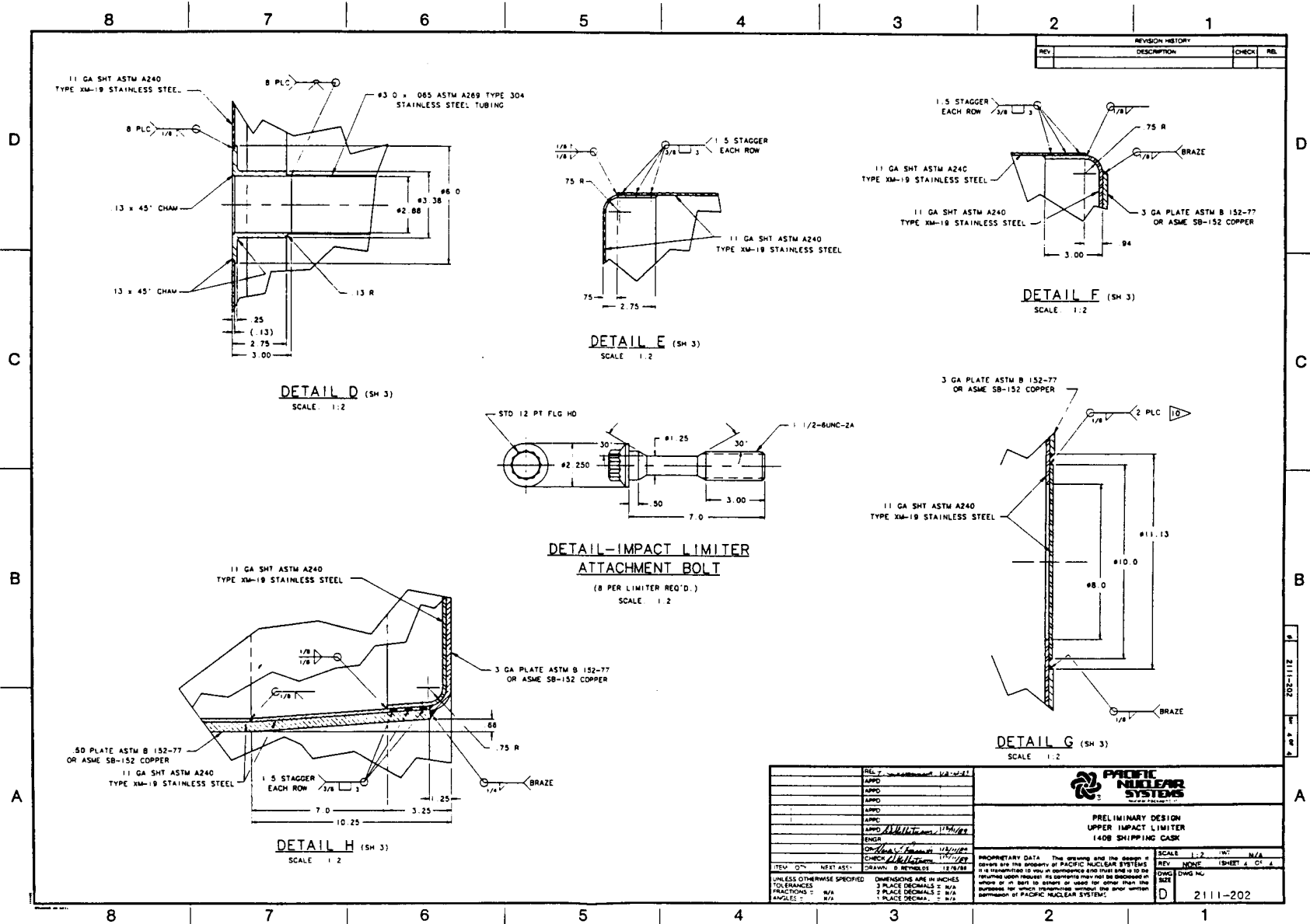








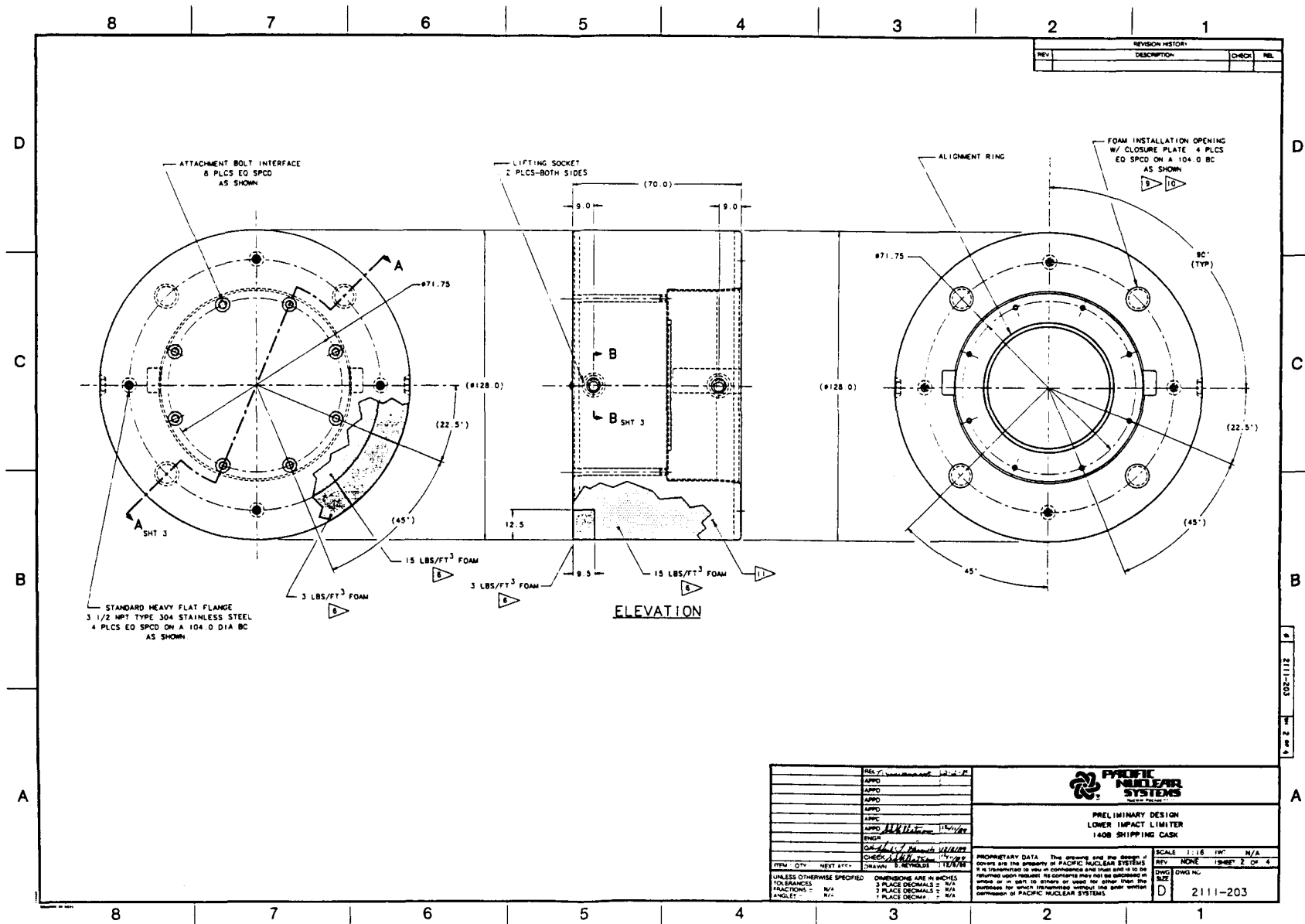


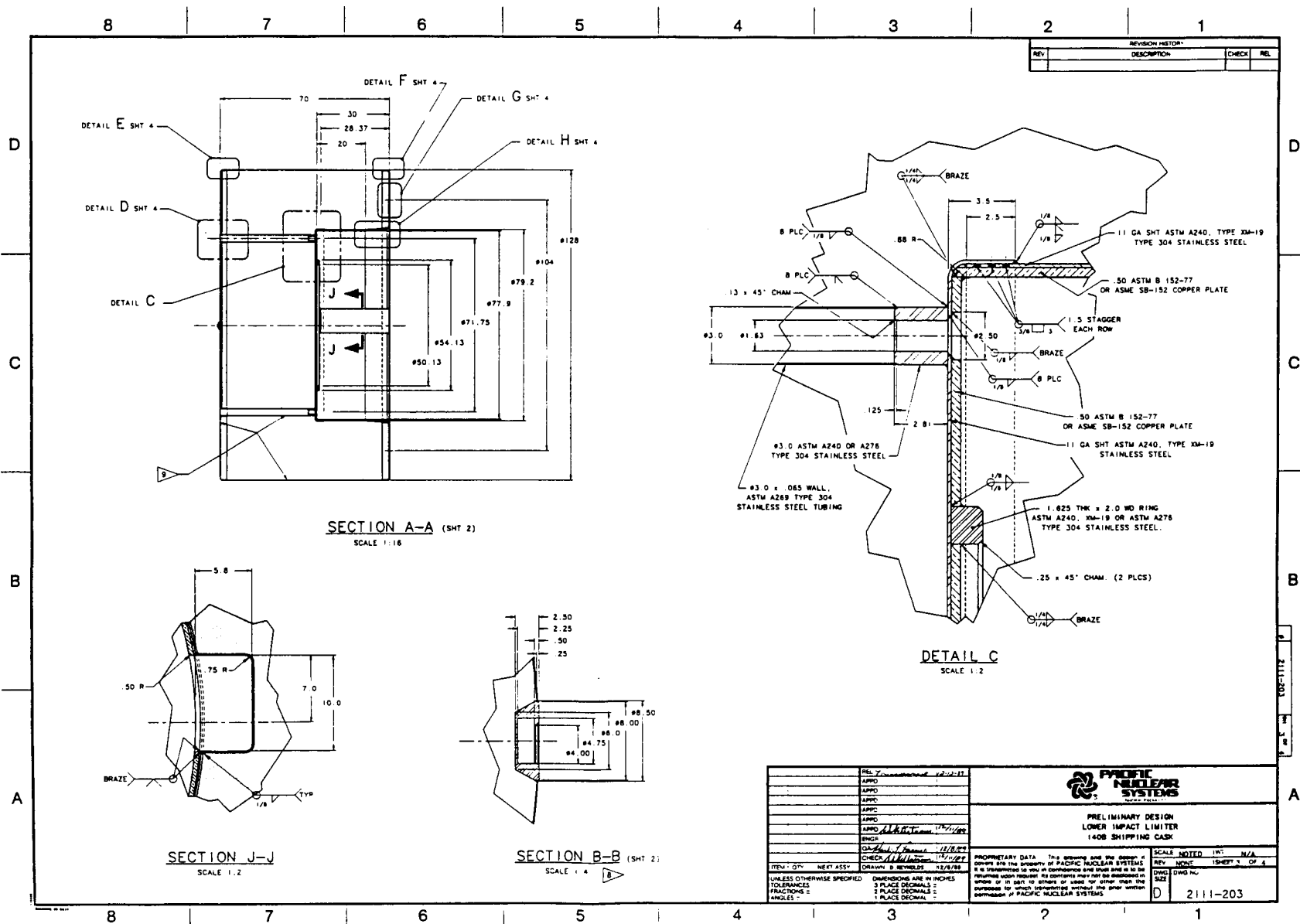


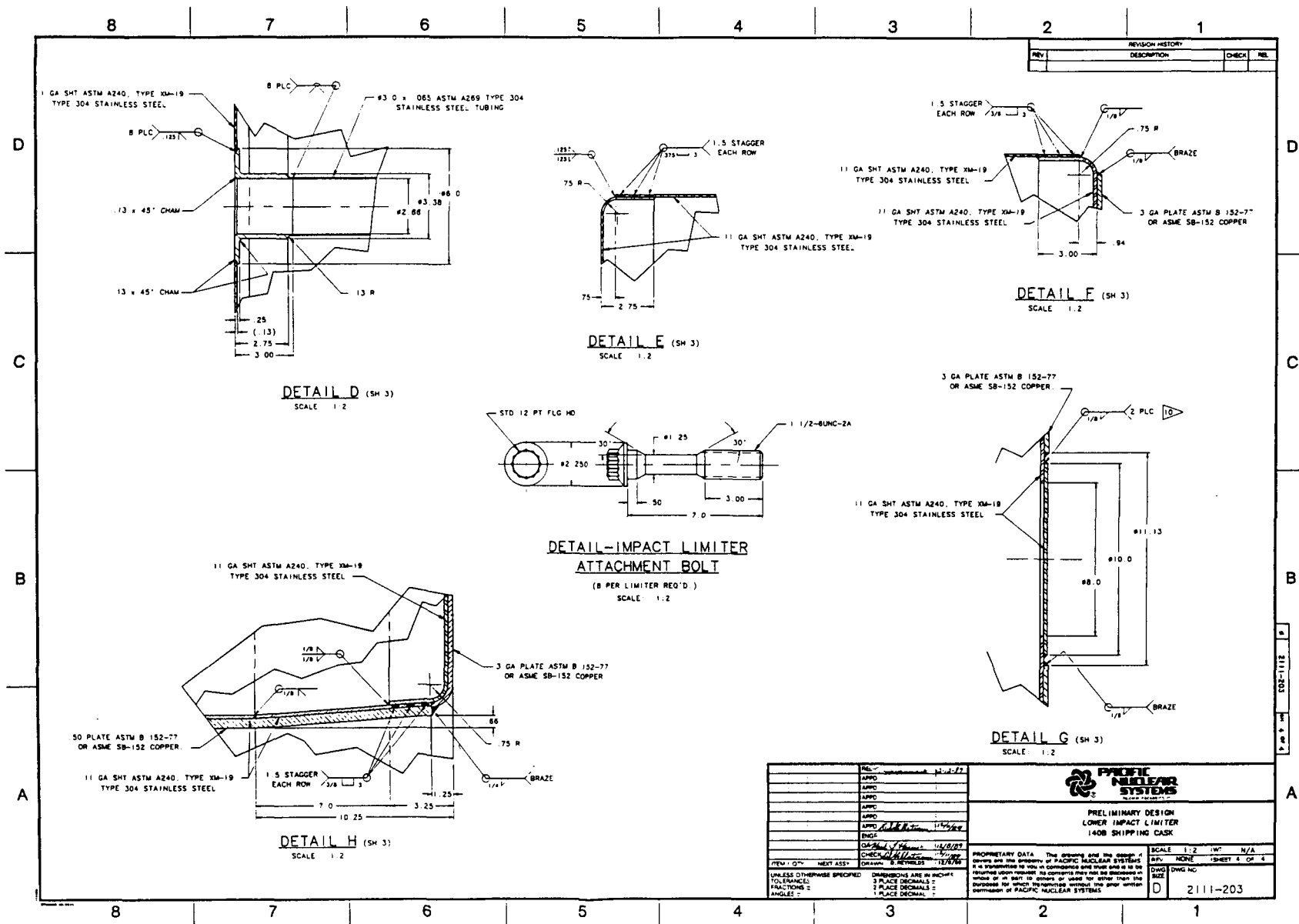
		8	7	6	5	4	3	2	1																																																																									
D	C	NOTES: UNLESS OTHERWISE SPECIFIED 1. INTERPRET DRAWING PER ANSI Y-14.5 2. ALL WELDING PROCEDURES AND PERSONNEL SHALL BE QUALIFIED IN ACCORDANCE WITH AWS D1.1 OR ASME CODE, SECTION IX. (ALL STAINLESS STEEL MATERIAL WELDING SHALL BE PERFORMED PER ASME CODE, SECTION IX.) WELD PROCEDURES AND WELDER QUALIFICATIONS SHALL BE AVAILABLE FOR AUDIT OR REVIEW 3. THREADS PER ANSI B1.1 1982 EDITION 4. ALL WELDS SHALL BE VISUALLY EXAMINED IN ACCORDANCE WITH AWS D1.1, SECTION 8.15.1. 5. WELDS SHALL BE LIQUID PENETRANT INSPECTED ON ROOT AND FINAL PASS IN ACCORDANCE WITH ASME CODE, SECTION III, DIVISION 1, SUBSECTION NB, ARTICLE NB-5000 AND SECTION V, ARTICLE 6 6. POLYURETHANE FOAMS WITH NOMINAL DENSITY'S OF 3 LBS/FT³ AND 15 LBS/FT³ ACCEPTABLE DEVIATION IN FOAM PROPERTIES ARE CONTROLLED BY THE FOAM STRESS/STRAIN REQUIREMENTS. 7. MATERIAL: ASTM-A182, TYPE 304-18 STAINLESS STEEL SHEET & PLATE ASTM-A280, TYPE 304 STAINLESS STEEL TUBING (SEAMLESS) ASTM-B152-77 OR ASME SB-152, COPPER SHEET & PLATE 8. IMPACT LIMITER LIFTING INTERFACES SHALL BE LOAD TESTED TO 150% OF MAXIMUM WORKING LOAD 9. PRIOR TO INSTALLATION OF POLYURETHANE FOAM, ALL INTERIOR SURFACES OF IMPACT LIMITER SHALL BE COATED WITH A SUBSTANCE WHICH WILL PREVENT FOAM BONDING 10. FOAM INSTALLATION OPENINGS SHALL BE COVERED AND SEAL WELDED AS SHOWN AFTER INSTALLATION OF POLYURETHANE FOAM. WELD PASS TEMPERATURE SHALL BE MAINTAINED AT 350°F OR LESS 11. FINISH: PAINT ALL EXPOSED SURFACES WITH ONE (1) PRIMER COAT (3-5 MILS DFT) _____ AND ONE (1) FINISH COAT (3-5 MILS DFT) _____ COLOR: WHITE 12. ALL DIMENSIONS ARE NOMINAL. TOLERANCES DO NOT APPLY AT PRELIMINARY DESIGN STAGE.							<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <th colspan="2">REVISION HISTORY</th> </tr> <tr> <th>REV.</th> <th>DESCRIPTION</th> </tr> <tr> <td> </td> <td> </td> </tr> <tr> <td> </td> <td> </td> </tr> </table>		REVISION HISTORY		REV.	DESCRIPTION																																																																				
									REVISION HISTORY																																																																									
									REV.	DESCRIPTION																																																																								
									B	A	<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td colspan="2" rowspan="5"> <table border="1" style="width: 100%; border-collapse: collapse;"> <tr><td>APPROVED</td></tr> <tr><td>APPROVED</td></tr> <tr><td>APPROVED</td></tr> <tr><td>APPROVED</td></tr> <tr><td>END</td></tr> </table> </td> <td colspan="2" rowspan="5" style="text-align: center;"> PACIFIC NUCLEAR SYSTEMS MAKING THE IMPOSSIBLE POSSIBLE </td> </tr> <tr> <td colspan="2" style="text-align: center;"> PRELIMINARY DESIGN LOWER IMPACT LIMITER 140B SHIPPING CASE </td> </tr> <tr> <td colspan="2"> PROPRIETARY DATA: The drawing and the design it contains are the property of PACIFIC NUCLEAR SYSTEMS. It is represented to you in confidence and it is to be returned upon request. Its contents may not be disclosed in whole or in part to others or used for other than the purposes for which it was prepared without the prior written permission of PACIFIC NUCLEAR SYSTEMS. </td> </tr> <tr> <td colspan="2"> <table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>SCALE: NONE</td> <td>UNIT: N/A</td> </tr> <tr> <td>REV: NONE</td> <td>SHEET: 1 OF 4</td> </tr> <tr> <td>DWG NO:</td> <td> </td> </tr> <tr> <td>DATE:</td> <td> </td> </tr> </table> </td> </tr> <tr> <td colspan="2"> <table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>ITEM</td> <td>QTY</td> <td>NEXT ASSY</td> <td>DRAWN</td> <td>BY</td> <td>REVISOR</td> <td>DATE</td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> </table> </td> </tr> <tr> <td colspan="2"> <table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>UNLESS OTHERWISE SPECIFIED:</td> <td>DIMENSIONS ARE IN INCHES</td> </tr> <tr> <td>TOLERANCES:</td> <td>3 PLACE DECIMALS ± N/A</td> </tr> <tr> <td>FRACTIONS ±</td> <td>2 PLACE DECIMALS ± N/A</td> </tr> <tr> <td>ANGLES ±</td> <td>1 PLACE DECIMAL ± N/A</td> </tr> </table> </td> </tr> </table>							<table border="1" style="width: 100%; border-collapse: collapse;"> <tr><td>APPROVED</td></tr> <tr><td>APPROVED</td></tr> <tr><td>APPROVED</td></tr> <tr><td>APPROVED</td></tr> <tr><td>END</td></tr> </table>		APPROVED	APPROVED	APPROVED	APPROVED	END	 PACIFIC NUCLEAR SYSTEMS MAKING THE IMPOSSIBLE POSSIBLE		PRELIMINARY DESIGN LOWER IMPACT LIMITER 140B SHIPPING CASE		PROPRIETARY DATA: The drawing and the design it contains are the property of PACIFIC NUCLEAR SYSTEMS. It is represented to you in confidence and it is to be returned upon request. Its contents may not be disclosed in whole or in part to others or used for other than the purposes for which it was prepared without the prior written permission of PACIFIC NUCLEAR SYSTEMS.		<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>SCALE: NONE</td> <td>UNIT: N/A</td> </tr> <tr> <td>REV: NONE</td> <td>SHEET: 1 OF 4</td> </tr> <tr> <td>DWG NO:</td> <td> </td> </tr> <tr> <td>DATE:</td> <td> </td> </tr> </table>		SCALE: NONE	UNIT: N/A	REV: NONE	SHEET: 1 OF 4	DWG NO:		DATE:		<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>ITEM</td> <td>QTY</td> <td>NEXT ASSY</td> <td>DRAWN</td> <td>BY</td> <td>REVISOR</td> <td>DATE</td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> </table>		ITEM	QTY	NEXT ASSY	DRAWN	BY	REVISOR	DATE								<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>UNLESS OTHERWISE SPECIFIED:</td> <td>DIMENSIONS ARE IN INCHES</td> </tr> <tr> <td>TOLERANCES:</td> <td>3 PLACE DECIMALS ± N/A</td> </tr> <tr> <td>FRACTIONS ±</td> <td>2 PLACE DECIMALS ± N/A</td> </tr> <tr> <td>ANGLES ±</td> <td>1 PLACE DECIMAL ± N/A</td> </tr> </table>		UNLESS OTHERWISE SPECIFIED:	DIMENSIONS ARE IN INCHES	TOLERANCES:	3 PLACE DECIMALS ± N/A	FRACTIONS ±	2 PLACE DECIMALS ± N/A	ANGLES ±	1 PLACE DECIMAL ± N/A	<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>ITEM</td> <td>QTY</td> <td>NEXT ASSY</td> <td>DRAWN</td> <td>BY</td> <td>REVISOR</td> <td>DATE</td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> </table>		ITEM	QTY	NEXT ASSY	DRAWN	BY	REVISOR	DATE							
																				<table border="1" style="width: 100%; border-collapse: collapse;"> <tr><td>APPROVED</td></tr> <tr><td>APPROVED</td></tr> <tr><td>APPROVED</td></tr> <tr><td>APPROVED</td></tr> <tr><td>END</td></tr> </table>		APPROVED	APPROVED	APPROVED			APPROVED	END	 PACIFIC NUCLEAR SYSTEMS MAKING THE IMPOSSIBLE POSSIBLE																																																					
																						APPROVED																																																												
																						APPROVED																																																												
																						APPROVED																																																												
APPROVED																																																																																		
END																																																																																		
PRELIMINARY DESIGN LOWER IMPACT LIMITER 140B SHIPPING CASE																																																																																		
PROPRIETARY DATA: The drawing and the design it contains are the property of PACIFIC NUCLEAR SYSTEMS. It is represented to you in confidence and it is to be returned upon request. Its contents may not be disclosed in whole or in part to others or used for other than the purposes for which it was prepared without the prior written permission of PACIFIC NUCLEAR SYSTEMS.																																																																																		
<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>SCALE: NONE</td> <td>UNIT: N/A</td> </tr> <tr> <td>REV: NONE</td> <td>SHEET: 1 OF 4</td> </tr> <tr> <td>DWG NO:</td> <td> </td> </tr> <tr> <td>DATE:</td> <td> </td> </tr> </table>		SCALE: NONE	UNIT: N/A	REV: NONE	SHEET: 1 OF 4	DWG NO:		DATE:																																																																										
SCALE: NONE	UNIT: N/A																																																																																	
REV: NONE	SHEET: 1 OF 4																																																																																	
DWG NO:																																																																																		
DATE:																																																																																		
<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>ITEM</td> <td>QTY</td> <td>NEXT ASSY</td> <td>DRAWN</td> <td>BY</td> <td>REVISOR</td> <td>DATE</td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> </table>		ITEM	QTY	NEXT ASSY	DRAWN	BY	REVISOR	DATE																																																																										
ITEM	QTY	NEXT ASSY	DRAWN	BY	REVISOR	DATE																																																																												
<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td>UNLESS OTHERWISE SPECIFIED:</td> <td>DIMENSIONS ARE IN INCHES</td> </tr> <tr> <td>TOLERANCES:</td> <td>3 PLACE DECIMALS ± N/A</td> </tr> <tr> <td>FRACTIONS ±</td> <td>2 PLACE DECIMALS ± N/A</td> </tr> <tr> <td>ANGLES ±</td> <td>1 PLACE DECIMAL ± N/A</td> </tr> </table>		UNLESS OTHERWISE SPECIFIED:	DIMENSIONS ARE IN INCHES	TOLERANCES:	3 PLACE DECIMALS ± N/A	FRACTIONS ±	2 PLACE DECIMALS ± N/A	ANGLES ±	1 PLACE DECIMAL ± N/A																																																																									
UNLESS OTHERWISE SPECIFIED:	DIMENSIONS ARE IN INCHES																																																																																	
TOLERANCES:	3 PLACE DECIMALS ± N/A																																																																																	
FRACTIONS ±	2 PLACE DECIMALS ± N/A																																																																																	
ANGLES ±	1 PLACE DECIMAL ± N/A																																																																																	
ITEM	QTY	NEXT ASSY	DRAWN	BY	REVISOR	DATE																																																																												

602-1112

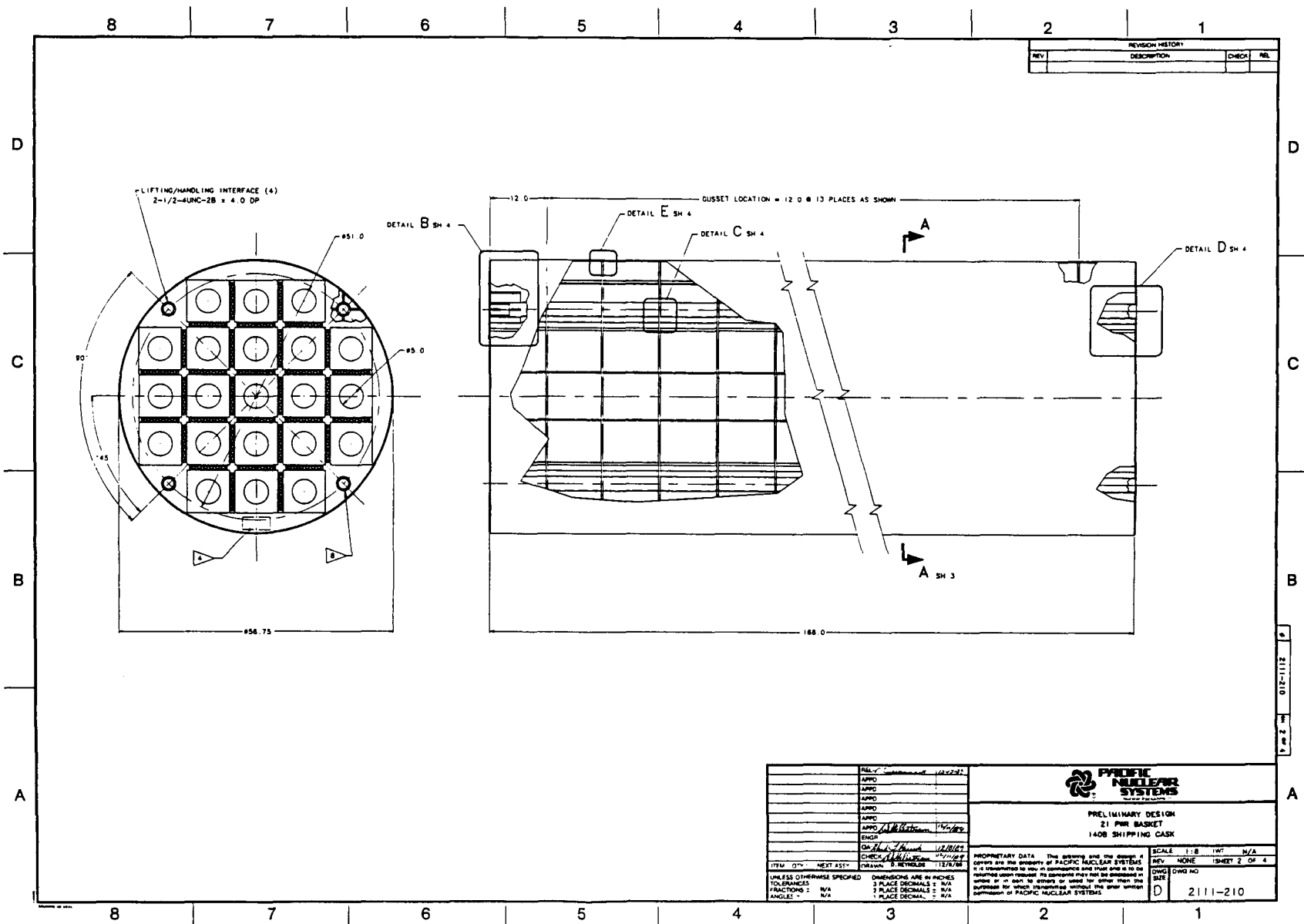
A

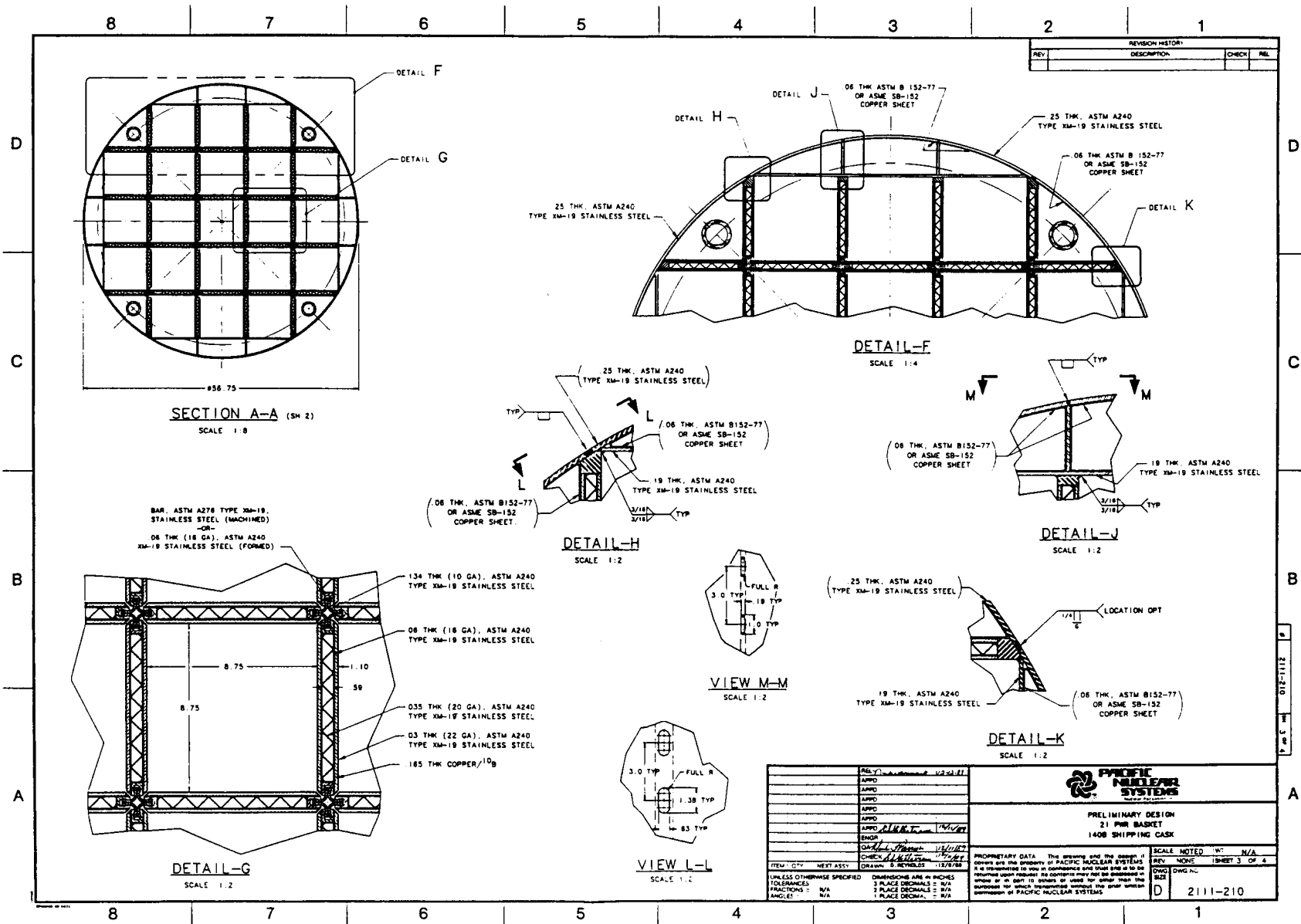


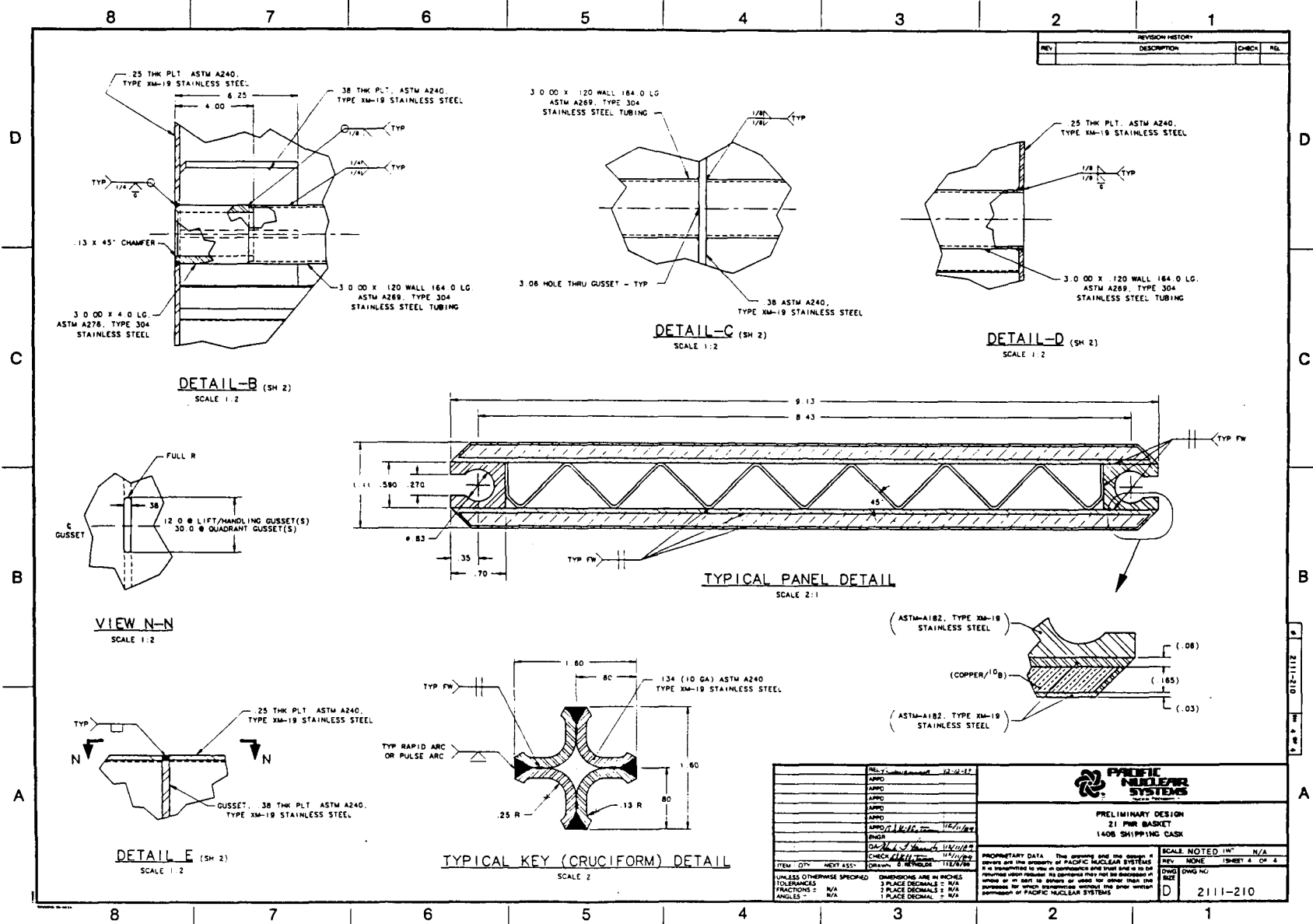




8		7		6		5		4		3		2		1																																															
NOTES, UNLESS OTHERWISE SPECIFIED 1. INTERPRET DRAWING PER ANSI Y-14.5 2. ALL WELDING PROCEDURES AND PERSONNEL SHALL BE QUALIFIED IN ACCORDANCE WITH AWS D1.1 OR ASME CODE, SECTION IX. WELD PROCEDURES AND WELDER'S QUALIFICATIONS SHALL BE AVAILABLE FOR AUDIT OR REVIEW. 3. THREADS PER ANSI B1.1 1982 EDITION 4. IDENTIFICATION BASKET SHALL BE IDENTIFIED IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 71.85 (c) 5. ALL WELDS SHALL BE VISUALLY EXAMINED IN ACCORDANCE WITH AWS D1.1, SECTION 8.15.1. 6. ALL WELDS SHALL BE VISUALLY AND NON-DESTRUCTIVE EXAMINED IN ACCORDANCE WITH THE REQUIREMENTS OF ASME SECTION III, DIVISION 1. EXAMINATION REQUIREMENTS WILL BE ESTABLISHED DURING FINAL DESIGN. 7. MATERIAL ASTM-A240, TYPE 304, STAINLESS STEEL PLATE, SHEET AND STRIP. ASTM-A479, TYPE 304, STAINLESS STEEL BARS AND SHAPES ASTM-A240, TYPE 304 STAINLESS STEEL PLATE, SHEET AND STRIP ASTM-A276, TYPE 304 STAINLESS STEEL BARS AND SHAPES 8. BASKET LIFTING INTERFACES SHALL BE LOAD TESTED TO 150% OF THE MAXIMUM WORKING LOAD. 9. ALL DIMENSIONS ARE NOMINAL. TOLERANCES DO NOT APPLY AT PRELIMINARY DESIGN STAGE.														<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <th colspan="2" style="text-align: center;">REVISION HISTORY</th> <th style="text-align: center;">CHECK</th> <th style="text-align: center;">REL</th> </tr> <tr> <th style="text-align: center;">REV</th> <th style="text-align: center;">DESCRIPTION</th> <th></th> <th></th> </tr> <tr> <td style="text-align: center;">0</td> <td>INITIAL RELEASE</td> <td></td> <td></td> </tr> </table>		REVISION HISTORY		CHECK	REL	REV	DESCRIPTION			0	INITIAL RELEASE																																				
REVISION HISTORY		CHECK	REL																																																										
REV	DESCRIPTION																																																												
0	INITIAL RELEASE																																																												
<table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td colspan="10" style="text-align: center;"> </td> <td colspan="6" style="text-align: center;"> PACIFIC NUCLEAR SYSTEMS INCORPORATED </td> </tr> <tr> <td colspan="10" style="text-align: center;"> PRELIMINARY DESIGN 21 PWR BASKET 1408 SHIPPING CASK </td> <td colspan="6" style="text-align: center;"> SCALE: NONE 1W N/A REV: NONE 1SHEET 1 OF 4 DATE: 01/11/82 SIZE: D 2111-210 </td> </tr> <tr> <td colspan="10" style="text-align: center;"> UNLESS OTHERWISE SPECIFIED: DIMENSIONS ARE IN INCHES TOLERANCES: 3 PLACE DECIMALS = .003 FRACTIONS = 1/16 ANGLES = 1 PLACE DECIMALS </td> <td colspan="6" style="text-align: center;"> PROPRIETARY DATA: This drawing and the design it covers are the property of PACIFIC NUCLEAR SYSTEMS. It is loaned to you in confidence and must not be reproduced or used in any way without the express written permission of PACIFIC NUCLEAR SYSTEMS. </td> </tr> </table>																								PACIFIC NUCLEAR SYSTEMS INCORPORATED						PRELIMINARY DESIGN 21 PWR BASKET 1408 SHIPPING CASK										SCALE: NONE 1W N/A REV: NONE 1SHEET 1 OF 4 DATE: 01/11/82 SIZE: D 2111-210						UNLESS OTHERWISE SPECIFIED: DIMENSIONS ARE IN INCHES TOLERANCES: 3 PLACE DECIMALS = .003 FRACTIONS = 1/16 ANGLES = 1 PLACE DECIMALS										PROPRIETARY DATA: This drawing and the design it covers are the property of PACIFIC NUCLEAR SYSTEMS. It is loaned to you in confidence and must not be reproduced or used in any way without the express written permission of PACIFIC NUCLEAR SYSTEMS.					
										PACIFIC NUCLEAR SYSTEMS INCORPORATED																																																			
PRELIMINARY DESIGN 21 PWR BASKET 1408 SHIPPING CASK										SCALE: NONE 1W N/A REV: NONE 1SHEET 1 OF 4 DATE: 01/11/82 SIZE: D 2111-210																																																			
UNLESS OTHERWISE SPECIFIED: DIMENSIONS ARE IN INCHES TOLERANCES: 3 PLACE DECIMALS = .003 FRACTIONS = 1/16 ANGLES = 1 PLACE DECIMALS										PROPRIETARY DATA: This drawing and the design it covers are the property of PACIFIC NUCLEAR SYSTEMS. It is loaned to you in confidence and must not be reproduced or used in any way without the express written permission of PACIFIC NUCLEAR SYSTEMS.																																																			







8

7

6

5

4

3

2

1

NOTES, UNLESS OTHERWISE SPECIFIED:

1. INTERPRET DRAWING PER ANSI Y-14.5.
2. ALL WELDING PROCEDURES AND PERSONNEL SHALL BE QUALIFIED IN ACCORDANCE WITH AWS D1.1 OR ASME CODE, SECTION IX. WELD PROCEDURES AND WELDER'S QUALIFICATIONS SHALL BE AVAILABLE FOR AUDIT OR REVIEW.
3. THREADS PER ANSI B1.1 1982 EDITION.
4. IDENTIFICATION: BASKET SHALL BE IDENTIFIED IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 71.85 (c).
5. ALL WELDS SHALL BE VISUALLY EXAMINED IN ACCORDANCE WITH AWS D1.1, SECTION 8.15.1.
6. ALL WELDS SHALL BE VISUALLY AND NON-DESTRUCTIVE EXAMINED IN ACCORDANCE WITH THE REQUIREMENTS OF ASME SECTION III, DIVISION 1. EXAMINATION REQUIREMENTS WILL BE ESTABLISHED DURING FINAL DESIGN.
7. MATERIAL: ASTM-A240, TYPE 304, STAINLESS STEEL PLATE, SHEET AND STRIP; ASTM-A479, TYPE 304, STAINLESS STEEL BARS AND SHAPES; ASTM-A276, TYPE 304 STAINLESS STEEL BARS AND SHAPES.
8. BASKET LIFTING INTERFACES SHALL BE LOAD TESTED TO 150% OF THE MAXIMUM WORKING LOAD.
9. ALL DIMENSIONS ARE NOMINAL. TOLERANCES DO NOT APPLY AT PRELIMINARY DESIGN STAGE.

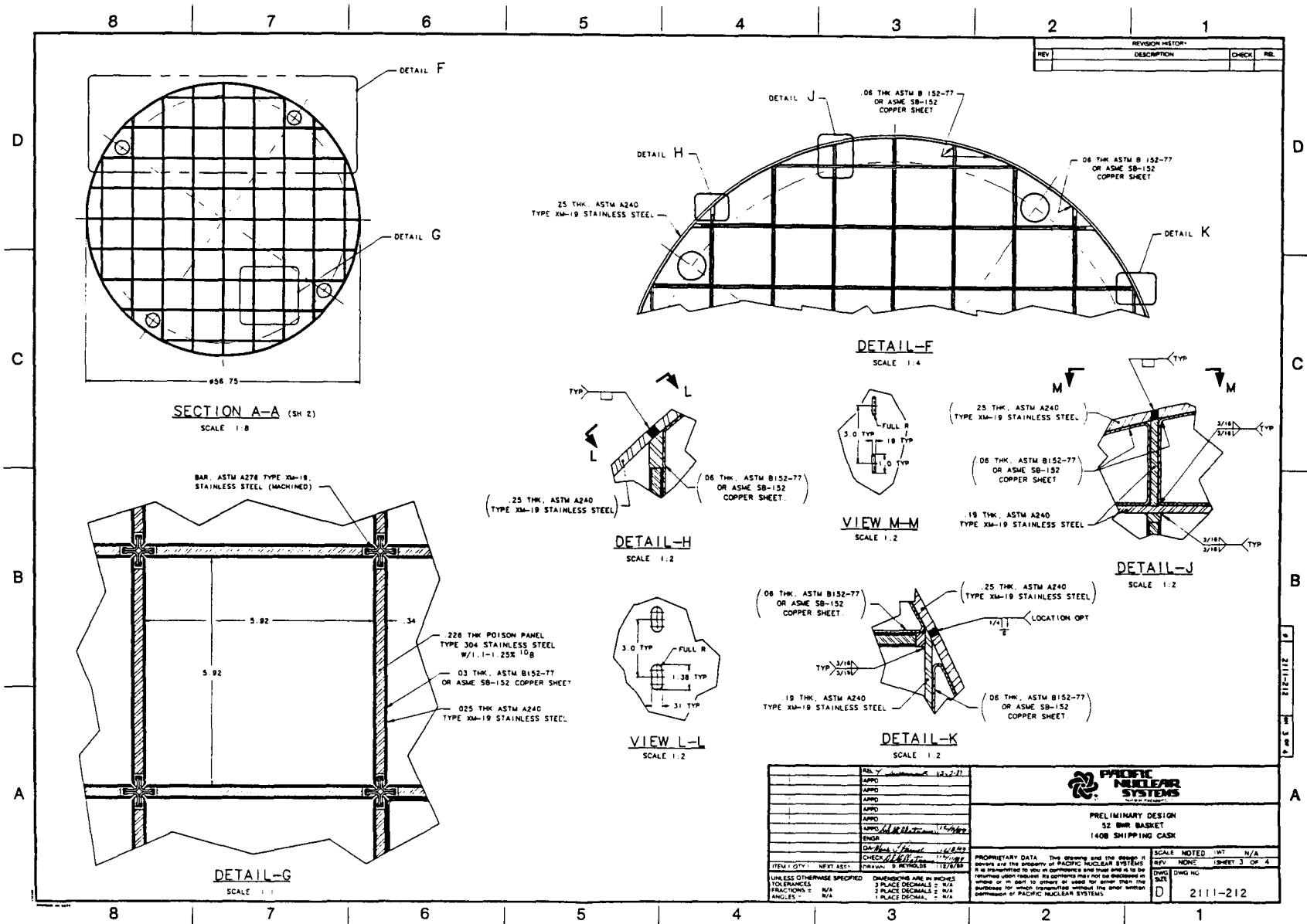
REVISION HISTORY			
REV	DESCRIPTION	CHECK	REL
1	INITIAL RELEASE		

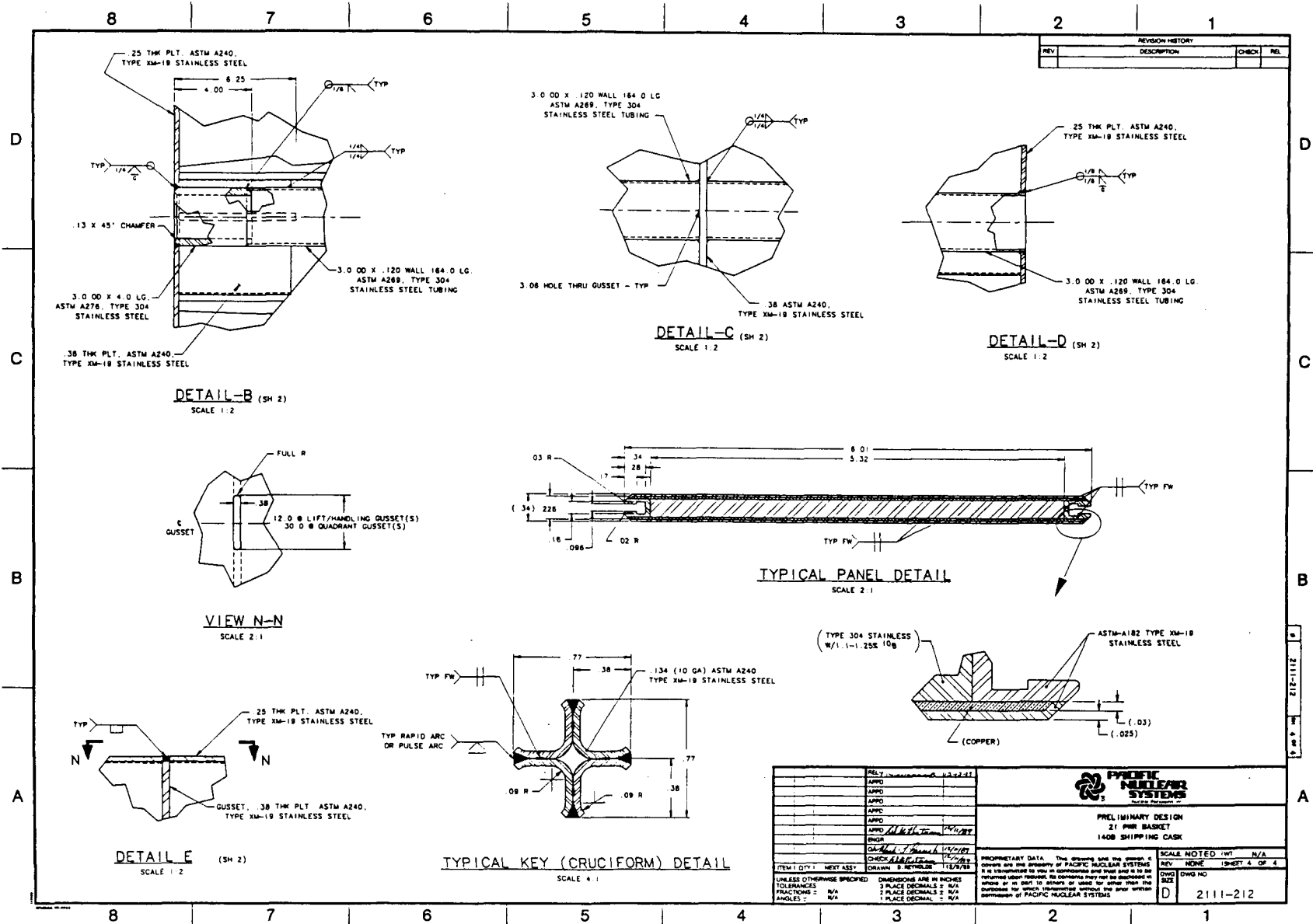
REL	1/1/80	1/1/80
APPD		
APPD		
APPD		
APPD		
APPD		
APPD		
ENDS		
CHKD	1/1/80	1/1/80
CHKD	1/1/80	1/1/80
ITEM: 07	NEXT ASSY:	
UNLESS OTHERWISE SPECIFIED:		
DIMENSIONS ARE IN INCHES		
TOLERANCES:		
FRACTIONS: 1/8		
ANGLES: 1/2		
3 PLACE DECIMALS: .010		
2 PLACE DECIMALS: .03		
1 PLACE DECIMAL:		

PRELIMINARY DESIGN
52 BWR BASKET
1408 SHIPPING CASK

PROPRIETARY DATA: This drawing and the design it covers are the property of PACIFIC NUCLEAR SYSTEMS. It is loaned to you in confidence and shall not be reproduced, stored in a retrieval system, or transmitted in any form or by any means, electronic, mechanical, photocopying, recording, or by any information storage and retrieval system, without the prior written permission of PACIFIC NUCLEAR SYSTEMS.			
SCALE	NONE	1/4"	N/A
REV	1/1/80	SHEET	1 OF 1
DWG NO	2111-212	SIZE	D

[illegible]





Appendix 1.3.3

Unique Features

1.3.3 Unique Features

1.3.3.1 PWR Basket

The PWR basket has been designed using light weight jointed truss core panels for the walls of the fuel assembly cell. The truss panel is constructed by welding a corrugated sheet of ASTM A240 Type XM-19 stainless steel between two sheets of ASTM A240 Type XM-19 stainless steel clad with copper and a neutron absorbing material. The panels are open at each end to allow water to enter the area occupied by the corrugated stainless steel. The copper serves to enhance the heat transfer capabilities of the basket and the neutron absorbing material serves as a capture device for thermalized neutrons. The copper is isolated from the fuel pool water. The panels are approximately 168 inches long and are fabricated as separate components. Each edge of the inner basket panels have a dovetail joint running the full length of the panel. The panels are assembled into a basket by interlocking the dovetail joints with tie bars. Welded truss core panels are located towards the outer radius of the basket. They are identical to the jointed panels except that along one side they have a reinforced fitting which is welded to the gussets and the outer shell of the basket. These welded joints provide added stability to the basket, especially for out of cask handling. The inner jointed connections allow flexibility for the basket to partially conform to the shape of the cask cavity during normal and accident conditions without exceeding yield in the basket. Lateral gusset plates are welded to the outer shell and lie in planes perpendicular to the longitudinal axis of the basket. Longitudinal gussets run parallel to the baskets longitudinal axis and are welded to the outer shell and the lateral gussets. These gussets provide stiffness to the basket and attachment points for the outer truss core panels. The outer shell of the basket is approximately 180 inches long and serves to prevent longitudinal movement of the fuel basket within the cask cavity.

Studies to determine the rate at which water will drain from and fill the flux trap will be performed during final design of the basket assembly.

Methods to decontaminate the cask internal components must be considered and the flux trap shown in the preliminary PWR basket design offers some challenges which must be addressed. Experience gained by NuPac personnel during the past 15 years in operating and maintaining rail mounted spent fuel shipping casks

will be utilized to provide practical solutions to decontamination of the fuel basket.

1.3.3.2 BWR Basket

The BWR basket is also constructed of panels which have dovetail joints along the entire length of the panel. Unlike the PWR basket panels, the BWR sections do not have a water flux trap. The cell walls are made of ASTM A240 Type XM-19 stainless steel, copper and neutron absorbing material. The copper is isolated in the wall of the fuel cell and is not allowed to contact the fuel pool water. The panels are connected and formed into a fuel basket assembly by interlocking the dovetail joints with a tie-bar. The flexibility of the basket joints allows the basket to partially conform to the shape of the cask cavity during normal and accident conditions.

1.3.3.3 Rail Car

The preliminary design has been developed for a dedicated rail car to transport the NuPac 140-B Cask. The maximum gross weight of the entire cask system on the rails can not exceed 263,000 pounds. The maximum allowable weight for the rail car was set at 40,000 pounds and the preliminary design, supported by design analysis, meets the weight budget. The design will allow for qualification of the railcar under Association of American Railroads (AAR) acceptance programs. During cask installation/removal at a reactor site, equipment will be mounted on the railcar which will allow removal of the impact limiters from the cask body by sliding them away from the cask toward the end of the railcar deck. The impact limiters remain on the railcar during cask loading/unloading operations thus eliminating the need for hoists or cranes.

The design service life of the railcar is 1,000,000 carriage miles.

1.3.3.4 Sunshield/Personnel Barrier

The Sunshield/Personnel barrier is designed to provide flexibility of function in addition to reducing solar heat input to the cask body. The screen also serves to ensure low personnel exposure by preventing close contact with the cask body during normal transport. The barriers enclose the cask surfaces

which are exposed between the impact limiters. The screen is attached to the cask support cradle and will provide protection when the cask is on the railcar, on a barge or heavy haul trailer or during off-transporter storage. Extension and retraction of the barrier can be performed by one person at ground level utilizing a powered drive mechanism thus eliminating the need for a facility crane or hoist. The screen can also be operated manually. The barrier will be locked in place during transport and will be equipped with tamper indicating devices.

1.3.3.5 Cask Lid Security System

Features to secure the lid to the cask body after all attachment bolts have been removed are included in the lift fixture design. This system enables transport of the cask in a vertical position within the facility such that the lid will remain in position in the event of a tipping accident. This eliminates the need for operating personnel to remove any lid closure bolts while the cask is suspended over the fuel storage pool or while in a hot cell. The locking wedges and blocks are operated by pneumatic cylinders and gravity allowing operation from a remote location.

Appendix 1.3.4
Human Factors/Operability

1.3.4 Human Factors/Operability

Human factors engineering relative to the operation and maintenance of the NuPac 140-B Cask will be of significant importance during the final design activity. Nuclear Packaging personnel have extensive experience in the design, operation and maintenance of large rail mounted spent fuel shipping systems. This experience has been gained over the past 15 years and includes approximately 216 cask shipments each containing 18 BWR fuel elements and 67 cask shipments each containing 7 PWR fuel elements. The total amount of spent fuel transported during this time exceeds 3600 assemblies, or approximately 815 metric tonnes. In addition, Nuclear Packaging personnel have experience in spent fuel shipments utilizing truck mounted casks. Nuclear Packaging rail mounted fuel shipping casks have also been utilized to transport solid radioactive waste to licensed burial sites. No other cask contractor has such an experience base which can be applied during the final design phase. It is of vital importance that the cask be designed such that it can be easily handled at a reactor site, at the repository or MRS or at the cask maintenance facility. The NuPac 140-B Cask design will undergo extensive reviews by those personnel which have experience as described above. Several items which will be of particular interest include the following:

- o Closure seals - The closure seal areas must be easily accessed by operating personnel utilizing equipment normally available at a cask receiving area. It must be possible to visually inspect the seal surface after loading fuel into the cask but prior to installation of the cask lid. Debris which will be introduced into the cask along with the fuel must be monitored and removed.
- o Cask penetrations - All cask penetrations must be of a design that will allow easy access and field maintenance. The ports should be of such a design that facility lines cannot be inadvertently connected to the wrong location. Nuclear Packaging will investigate the possibility of having quick disconnect couplings at each connect point which will allow external valving at each port. This will simplify maintenance.

- o Impact limiters - The impact limiters have been designed such that they are attached to and removed from the cask body from the end position. This reduces the personnel exposure which will be experienced during cask handling. All areas where operating personnel must work will be analyzed to reduce the time required to perform an operation and/or reduce the exposure risk.
- o Cask drain - The cask drain port must be designed such that it can be easily accessed to remove radioactive debris which may be introduced into the cask during fuel loading operations. Such debris can plug the drain and prevent proper draining and cause hot spots which could increase personnel exposure.
- o Basket decontamination - The fuel baskets must be designed such that decontamination can be easily accomplished during the annual inspections. The flux trap which is part of the PWR basket design must be reviewed to ensure crud does not accumulate during normal operation of the cask.
- o Cask surfaces - The various surfaces of the cask which may become contaminated must be of such a design that they can be easily cleaned at the point of use. The surface must be smooth without crevices where contamination could become trapped. Various surface treatments must be investigated to reduce the incidence of surface 'weeping' which is prevalent on currently licensed casks.
- o Removable parts - All parts which are removed from the cask during handling will be reviewed to determine if they can be captured to preclude loss or damage. In some cases significant time has been spent in locating lost parts which must be replaced on a cask prior to shipment when simple re-engineering could have been provided a capture mechanism.
- o Handling of the cask at the reactor site, at the maintenance facility and at the repository will probably be done by personnel wearing protective clothing, including rubber gloves. The cask is being

designed such that all handling, maintenance, decontamination, preparation for shipment, etc. can be performed when such personnel protection devices are used.

In summary, all aspects of the cask design will be reviewed by experienced cask users to ensure that the cask and all supporting equipment can be operated and maintained in a manner which will allow rapid turnaround time and simplified maintenance.

Appendix 1.3.5
Technical Certification Issues Requiring NRC Resolution

1.3.5 Technical Certification Issues Requiring NRC Resolution

1.3.5.1 Basket Neutron Absorber Material

At the present time the neutron absorbing material proposed for use in the fuel baskets has not been approved by the NRC. The proposed neutron absorber is a copper clad aluminum material which has boron carbide dispersed throughout the inner aluminum core. Essentially the material is envisioned as modified BORAL, which is a material manufactured by Brooks and Perkins in Livonia, Michigan. Boral has been successfully used in nuclear applications for many years and has been accepted by the NRC. The material proposed for use in the NuPac 140-B cask uses the Boral core matrix with a copper outer skin rather than the current aluminum cover.

1.3.5.2 Cask Neutron Absorber Material

The neutron shield currently shown on the design of the NuPac 140-B cask is a borated silicone material which may require additional evaluation and testing before submittal of the SAR to the NRC. Additional test data will be obtained and evaluated after which a determination will be made regarding the need for additional testing and evaluation. Nuclear Packaging considers this. We believe our approach is sound and that we have addressed all of the known material concerns in our design.

1.3.5.3 Fuel Basket Design

The design of both the PWR and the BWR baskets will be reviewed with the NRC to determine specific issues which must be addressed during the final design and testing phase of the program. The structural analyses which have been performed show the basic design is sound. In addition we have been assured that the basket can be fabricated using existing technology. Concerns regarding potential decontamination problems are being addressed.

1.3.5.4 Testing After O-Ring Seal Replacement

The current design approach is that one of the tests which must be performed to verify cask integrity is a leak test performed at the time the cask is manufactured, then on an annual basis, or whenever significant repairs are made to one or more of the sealing surfaces. The acceptance criteria for this test is more stringent than leak testing which will be performed at a reactor site prior to shipment of a loaded cask. Operating experience indicates that the primary O-ring seal will be considered as a consumable and will be replaced at the reactor site each time a cask is loaded. Further, the acceptance criteria for the leak test performed after the seal replacement and prior to shipment is assumed to be the same as if the seal ring was not replaced. Repair of the seal surface is not a consideration. It appears that NRC and DOE agreement must be obtained on this point.

Appendix 1.3.6
Technical Issues Requiring Experimental Verification

1.3.6 Technical Issues Requiring Experimental Verification

1.3.6.1 Component Testing

It is anticipated that additional testing and/or evaluation will be required on the following items:

- o Seals - Preliminary material selection has been performed. Additional testing of full size samples may be required. Radiation resistance must also be evaluated.
- o Borated Silicone Neutron Absorber - Additional test data will be obtained and evaluated. Test plans are being prepared which will identify any additional evaluations and/or tests which may be required to obtain NRC certification.
- o Puncture Test - The need to perform a puncture test on a full scale section of the cask wall is being evaluated. Puncture testing will be performed on the cask scale model and may provide adequate data to confirm the adequacy of the analyses.
- o Basket Panels - Full scale PWR and BWR basket panels will be fabricated and tested to verify the structural and thermal evaluation results as well as to confirm decontamination assumptions.

1.3.6.2 Scale Model Testing

A scale model of the cask body and the impact limiters will be fabricated and tested. Due to the small size, a dummy basket will be used during these tests. Several drop tests will be performed as well puncture testing of the cask body. A detailed description of the specific tests which will be performed, as well as the anticipated results will be included in the Scale Model Test Plan.

2.0 STRUCTURAL EVALUATION

This section presents structural evaluations demonstrating that the NuPac 140-B Cask design meets all applicable structural criteria. The energy absorbing impact limiters, outer cask shells and closures, and the internal PWR and BWR payload baskets, are evaluated and shown to provide adequate protection for the PWR and BWR fuel assembly payloads. Normal and hypothetical accident condition evaluations using primarily analytical techniques are performed in accordance with 10 CFR 71 requirements, Reference (1). Normal and hypothetical accident condition evaluations using experimental techniques are planned later in accordance with the contract schedule per Reference (2).

Analytic demonstration techniques comply with the criteria presented in Regulatory Guide 7.6, Reference (3) and the load combination requirements of Regulatory Guide 7.8. Experimental verifications and evaluations planned later are of the following six forms:

- o 1/4 Scale Drop Tests of cask with impact limiters and simulated payloads.
- o Full scale cross section, partial scale length PWR and BWR payload baskets structural tests to simulate worst case drop loads.
- o Full scale thermal performance tests on the prototype casks later in the program.
- o Component tests to verify puncture resistance of cask wall and gamma/neutron shield system.
- o Component tests to verify that borated-silicone neutron shield tolerances are adequate.
- o Component tests to verify that the impact limiters perform as designed including actual heated cask conditions.

Experimental verifications and evaluations that have been performed to date are reported in Reference (4), for elastomeric seals, and Reference (5), for polyurethane impact limiter foam at high temperatures.

Analytically determined minimum margins of safety obtained for the major cask components, internal PWR and BWR basket sections, and worst case PWR fuel assembly payload are summarized in Tables 2.0-1 and 2.0-2. Table 2.0-1 presents margins associated with normal conditions of transport and Table 2.0-2 presents margins of safety for hypothetical accident conditions.

TABLE 2.0-1
Margins of Safety - Normal Conditions of Transport

COMPONENT	<u>Minimum Margin of Safety</u>				PEAK (Fatigue)
	PRIMARY Tensile	MEMBRANE Compressive (Buckling)	PRIMARY MEMBRANE + BENDING	PRIMARY + SECONDARY (Thermal)	
Outer Shell	MS=3.24 (2.6.7.3(2))	MS=3.24 (2.6.7.3(2))	(B)		(C)
Inner Shell	MS=1.61 (2.6.7.3(3))	MS=3.63 (2.6.7.1(6))	(B)		(C)
Outer Cask					
Lid	(A)	(A)	MS=1.42 (2.6.7.1(3))		(C)
Lid Bolts	MS=0.09 (2.6.7.1(4))	N/A	(B)		MS=2.3 (2.1.2.2.2.1)
Outer Shell	(D)	MS=Large (2.6.8.1.2)	MS=1.80 (2.6.8.1.1)		N/A
Basket					
Truss		MS=2.37	MS=1.81		
Core Panel	(D)	(2.6.8.1.2)	(2.6.8.1.1)		N/A
PWR		MS=2.22			
Fuel Rods		(E)			N/A

Notes: Continued on next page

- Notes:
- A. Lids are governed by membrane plus bending allowables rather than membrane allowables.
 - B. Shells and bolts are governed by membrane allowables rather than membrane plus bending.
 - C. Non-Fatigue allowables govern per Section 2.1.2.2.2.1.
 - D. Baskets are governed by membrane plus bending and buckling allowables.
 - E. Ratio of allowable inertia load (from Reference (9)) to actual predicted for NuPac 140-B Cask per Appendix 2.10.2 minus 1.0.

TABLE 2.0-2
Margins of Safety - Hypothetical Accident Conditions

COMPONENT	<u>Minimum Margin of Safety</u>			EXTREME STRESS RANGE
	PRIMARY Tensile	MEMBRANE Compressive (Buckling)	PRIMARY MEMBRANE + BENDING	
Outer Cask	Outer Shell	MS=1.69 (2.7.1.3(2))	MS=1.37 (2.7.1.2(4))	(B)
	Inner Shell	MS=1.53 (2.7.1.3(3))	MS=1.21 (2.7.1.2(4))	(B)
	Load Collar	N/A	N/A	MS=0.83 (2.7.3.2)
	Lid	(A)	(A)	MS=1.50 (2.7.1.1(3))
	Lid Bolts	MS=0.10 (2.7.1.2(8))	N/A	(B)
Basket	Outer Shell	(C)	MS=large (2.7.7.1)	MS=1.99 (2.7.7.2)
	Truss		MS=0.69 (2.7.7.1)	MS=1.35 (2.7.7.2)
	Core	(C)		
	Panel			
PWR		MS=0.12		
Fuel		(D)		
Rods				

Notes: Continued on next page.

- Notes:
- A. Lids are governed by membrane plus bending allowables rather than membrane allowables.
 - B. Shells and bolts are governed by membrane allowables rather than membrane plus bending.
 - C. Baskets are governed by membrane plus bending and buckling allowables.
 - D. Ratio of allowable inertia load (from Reference (9)) to actual predicted for NuPac 140-B Cask per Appendix 2.10.2 minus 1.0.

2.1 Structural Design

This section describes the structural configuration of the NuPac 140-B Cask; states design criteria primarily based upon NRC Regulatory Guide 7.6; discusses brittle fracture considerations; discusses fatigue considerations; and mentions vibration, extreme total stress intensity range, and buckling. Miscellaneous structural design criteria are listed concerning maximum weight limits, maximum life requirements, and payload fuel assembly inertia load limits.

2.1.1 Discussion

The NuPac 140-B Cask consists of three basic components as depicted in the drawings of Appendix 1.3.2: (1) the energy absorbing impact limiters, drawings 2111-202 & 203, which protect the cask, (2) the cask itself, which provides the primary containment (the fuel rods themselves provide their primary level of containment), drawing 2111-201 and (3) the basket assemblies, drawings 2111-210 & 212. The polyurethane foam-filled impact limiters serve the dual purposes of limiting 'g' loads acting on the cask under drop conditions and providing thermal insulation for a hypothetical accident thermal event. By selecting a proper density foam, 'g' loads acting upon the cask are limited to acceptable levels.

The basic structure of the outer cask consists of austenitic stainless steel (Type XM-19) materials and is comprised of an 8.50 inch thick bottom closure plate which welds directly to an outer shell (1.50 inches thick, 69.25 inch OD) and an inner shell (1.13 inch thick, 59.25 inch OD). The annular region between these shells is filled with lead shielding. The top closure consists of a thick walled forging (ring) with transitions which weld directly to the inner and outer shells and an 8.50 inch thick separate top cover lid which bolts to the body forging via thirty-two (32), ASTM A-320, Grade L43, 1 5/8 - 8 UN bolts. The bottom closure plate, inner shell, top forging and top closure plate form the primary containment boundary. Lifting and handling trunnions made of 17-4 PH (Type 630) stainless steel are bolted to the top forging and bottom closure plate. The outer shell is surrounded by a borated-silicone neutron shield which varies in thickness from 3.75 inches at the

cask ends to 5.875 at the cask center. Borated-silicone neutron shield segments are sandwiched between 24, 0.50 inch copper (SB-152) heat transfer fins and a circumferential 0.125 inch copper shell. In turn, the copper shell is surrounded by a stainless steel fire shield made up of 0.125 and 0.25 inch plate. The 0.25 inch sections are located at each end and are termed 'load collars' since they contain four flanges, two on each side of cask, utilized for tiedown of the cask on the transport structure. The stainless steel employed for the fire shield/load collar shell is Type XM-19.

The NuPac 140-B cask contains two independent basket designs (see drawings 2111-210 & 212 of Appendix 1.3.2). The independent baskets are used for PWR or BWR Fuel. The basic structure, of both basket assemblies, consists of a stainless steel (Type XM-19) shell and the basket web assembly connected to it by a series of gusset supports. The basket web assemblies are plate assemblies fabricated of stainless steel (Type XM-19), copper (SB-152), and a neutron absorbing material in a sandwich assembly.

The support gussets are fabricated of stainless steel (Type XM-19) and copper (SB-152). The basket shells for both PWR and BWR assemblies are 0.25 inch thick stainless steel (Type XM-19) with outer diameter of 56.75 inches and length of 168.0 inches with provision for extension to 180.0 inches. The support gussets for both PWR and BWR baskets are similar in design. The support gussets are 0.375 inches wide, 0.19 inches thick and welded to the shell at 12 inches intervals. The gussets form hollow support cells which are lined on the interior with 0.06 inch thick copper sheets.

The basket web assemblies are connected to the shell by welded connections to the gusset supports. The PWR Basket Web Assembly provides 21, 8.75 x 8.75 inch fuel cells. The web plate ligaments are 1.11 inches thick and 9.13 inches long. Each ligament consists of two, 0.06 inch thick stainless steel plates, connected 0.59 inches apart by a 0.035 inch thick stainless steel web. This assembly acts as a truss type mechanism. The stainless steel plates are covered with 0.165 inches of neutron absorbing material and copper with a 0.03 inch thick stainless steel

cover plate. Each ligament contains a 0.70 inch by 0.59 inch stainless steel connection block in each end.

The basket web is assembled by connecting the individual ligaments together with a stainless steel (Type XM-19) key. The key serves as a pinned end connection between the individual ligaments. The BWR web assembly is similar in design to the PWR web assembly. The web plate ligaments are 0.34 inches thick and 6.01 inches long. Each ligament consists of a 0.226 inch thick stainless steel poison panel (Type 304) covered with a 0.03 inch thick copper sheet (SB 152) and a 0.025 inch thick stainless steel cover plate (Type XM-19). Each ligament contains a 0.226 inch by 0.34 inch stainless steel connection block in each end. The basket web is assembled similarly to the PWR design with a stainless steel (Type XM-19) key to serve as a pinned connection between the individual ligaments. The completed assembly will provide 52, 5.92 inch by 5.92 inch BWR fuel cell openings, which will allow shipment of the fuel channels.

A rupture disc is provided in the top closure plate of the cask. This rupture disc is sized for over pressurization greater than that for structural testing as discussed in Section 8.1.1. The disc will rupture before gross failure of the containment structure in the event of a gross overpressurization event.

All welds are designed and detailed in accordance with ASME Section III NB standards. All containment joint designs, shell transitions, tolerances, weld inspections, and tolerance requirements conform with those imposed by ASME Section III NB.

2.1.2 Design Criteria

2.1.2.1 Basic Design Criteria (Allowable Stresses)

This section defines the stress allowable for primary membrane, primary bending, secondary, peak, bearing, shear, and buckling stresses for

containment structures and fasteners, and non-containment structures and fasteners.

Regulatory Guide 7.6, Reference (3), is used in conjunction with Regulatory Guide 7.8, Reference (7), to evaluate the integrity of the NuPac 140-B Cask. Where the loads specified by Regulatory Guide 7.8 conflict with those given in the current version 10 CFR 71, the latter is used. Material properties and design stress intensity values, S_m , used in the analyses can be found in Tables 2.1-1 and 2.1-2.

2.1.2.1.1 Containment Structures

Regulatory Guide 7.6 was used for all package containment boundaries for both the normal conditions of transport and the hypothetical accident conditions. Material data used in the evaluation correspond to the design stress intensity values, S_m , yield strengths, S_y , and ultimate strengths, S_u , given in the ASME Code, Section III, Class I, Reference (6).

The primary containment boundary is considered to be the 57.00 inch inside diameter, 1.13 inch thick inner shell of the cask, the forging at the top end of the cask, the bolted closure lid, the elastomeric seals in between, and the 8.50 inch thick cask bottom plate. A summary of allowable stresses used for containment structures, non-containment structures, and fasteners is presented in Tables 2.1-1 and 2.1-2. These data are consistent with Regulatory Guide 7.6 and Sections NB-3000 and Appendix F of ASME Section III, Reference (6). In addition to the criteria of regulatory guide 7.6, the following requirements are also followed in the cask design: 1) No yielding of materials is allowed in vicinity of seals, 2) No bolt yielding in normal or accident conditions, and 3) imposition of IAEA 200 meter immersion event (addressed in detail in Section 2.7.5).

2.1.2.1.2 Non-Containment Structures

Structural evaluations of non-containment boundaries, such as the basket assembly, the cask outer shell, and lifting and tie-down trunnions use allowable stresses for normal and accident conditions as presented in Table 2.1-2. The impact limiter is allowed to plastically deform for all conditions. The acceptance criterion for all impact related loads within the impact limiter is that no cask 'hard points' directly come into contact with the impact surface. For lifting and handling loads, the 'non-containment' normal condition allowables of Table 2.1-2 are utilized in conjunction with a safety factor of three (3), per 10 CFR 71.45.

TABLE 2.1-1
Allowable Stress Limits

Stress Category	CONTAINMENT STRUCTURE ALLOWABLE STRESSES	
	Normal Conditions	Accident Conditions
Primary Membrane Stress Intensity	S_m	Lesser of: $2.4S_m$ $0.7S_u$
Primary Membrane + Bending Stress Intensity	$1.5S_m$	Lesser of: $3.6S_m$ S_u
Range of Primary + Secondary Stress Intensity	$3.0S_m$	Not Applicable
Bearing Stress	S_y	S_y for seal surfaces S_u elsewhere
Pure Shear Stress	$0.6S_m$	$0.42S_u$
Peak	Per Section 2.1.2.2.2	
Buckling	Per Section 2.1.2.2.3	
Stress Category	CONTAINMENT FASTENER ALLOWABLE STRESSES	
	Normal Conditions	Accident Conditions
Membrane Stress*	lesser of: $2.0S_m$ S_y	S_y
Membrane + Bending Stress *	lesser of: $3.0S_m$ S_y	S_y

*Not Considering Stress Concentrations (Stress concentration is considered in Section 2.1.2.2.2)

TABLE 2.1-2
Allowable Stress Limits

Stress Category	NON-CONTAINMENT STRUCTURE ALLOWABLE STRESSES	
	Normal Conditions	Accident Conditions
Primary Membrane Stress Intensity	Greater of: S_m S_y	$0.7S_u$
Primary Membrane + Bending Stress Intensity	Greater of: $1.5S_m$ S_y	S_u
Range of Primary + Secondary Stress Intensity	Greater of: $3.0S_m$ S_y	Not Applicable
Bearing Stress	S_y	S_u
Pure Shear Stress	Greater of: $0.6S_m$ $0.6S_y$	$0.42S_u$
Peak	Per Section 2.1.2.2.2	
Buckling	Per Section 2.1.2.2.3	
Stress Category	NON-CONTAINMENT FASTENER ALLOWABLE STRESSES	
	Normal Conditions	Accident Conditions
Membrane Stress *	Greater of: $2.0S_m$ S_y	Greater of: S_y $0.7S_u$
Membrane + Bending Stress *	Greater of: $3.0S_m$ S_y	S_u

* Not Considering Stress Concentrations (Stress concentration is considered in Section 2.1.2.2.2)

2.1.2.2 Miscellaneous Structural Failure Modes

2.1.2.2.1 Brittle Fracture

With exception of the closure bolts, all containment, and nearly all non-containment structural components, are fabricated of Type XM-19 austenitic stainless steel. Since this material does not undergo a ductile-to-brittle transition in the temperature range of interest (down to -40° F), it is safe from brittle fracture.

The closure bolts are fabricated of ASTM A-320, Grade L43, alloy steel. Per Section 5 of NUREG/CR-1815 Reference (8), bolts are generally not considered as fracture-critical components because multiple load paths exist and because bolted systems are designed to be redundant. However, for purposes of comparison, the nil-ductility transition (NDT) temperature of the closure bolts will be calculated and compared with the requirements of Reference (8).

According to Section 6.2.1.1 of the ASTM A-320, Grade L43, specification, the minimum impact energy absorption is 20 ft-lbs at -150 °F. The Charpy impact measurement may be transformed into a fracture toughness value by using the empirical relation developed in Section 4.2 of Reference (8), as follows:

$$K_{ID} = [5E(C_v)]^{0.5}$$

Where:

K_{ID} = dynamic fracture toughness, psi-in^{0.5}

E = Modulus of Elasticity, psi

= 28.8(10)⁶ psi at -150° F, Table I-5.0 of Ref. (6)

C_v = Charpy impact measurement, ft-lbs

= 20 ft-lbs

Then,

$$K_{ID} = \{5[28.8(10)^6](20)\}^{0.5} = 53,665 \text{ psi-in}^{0.5}$$

The dynamic fracture toughness is translated to an equivalent nil ductility transition (NDT) temperature by using the Pellini reference curve given as Figure 2 in Reference (8).

By interpolation, the temperature relative to NDT is found as:

$$(T - \text{NDT}) = 30^{\circ}\text{F}$$

Accordingly, the nil-ductility transition temperature is:

$$\begin{aligned}\text{NDT} &= -150 - (+30) \\ &= -180^{\circ}\text{F}\end{aligned}$$

For Category I fracture critical components, and in section thicknesses of 1.5 inches (bolt diameter), Figure 3, Reference (8), gives the minimum offset, 'A' as approximately 60°F. Thus, the maximum NDT temperature value is:

$$T_{\text{NDT}} = \text{LST} - A = -40 - 60 = -100^{\circ}\text{F}$$

Where:

$$\begin{aligned}T_{\text{NDT}} &= \text{maximum NDT temperature per Reference (8)} \\ \text{LST} &= \text{lowest service temperature} \\ &= -40^{\circ}\text{F (Reg Guide 7.8)} \\ A &= 60^{\circ}\text{F, per Figure 3, Reference (8)}\end{aligned}$$

The ASTM A-320, Grade L43, closure bolts experience a ductile to brittle transition temperature at -180°F whereas the criterion of Reference (8) prescribes a maximum NDT temperature of -100°F. The 80°F margin between criteria requirements and material capability provide conservative assurance that brittle fracture failures will not occur in these ferritic closure bolt materials.

2.1.2.2.2 Fatigue

2.1.2.2.2.1 Normal Operating Cycles

Normal operating cycles do not present a fatigue concern for the NuPac 140-B Cask components which have no significant stress concentrations.

This is because the allowable stress for normal conditions ($3.0S_u$) will not exceed the allowable fatigue stress limit for the expected number of operating cycles. From Table I-1.2 of ASME Section III, Reference (6), Appendix I, the maximum value of the stress intensity (S_m) at the maximum normal operating temperature established herein, 300° F, is 31,400 psi for the Type XM-19 stainless steel used in the NuPac 140-B Cask. The expected number of operating cycles (defined as the process of going from an empty cask, to one with maximum heat load, at the maximum normal operating temperature, and back again) for the NuPac 140-B Cask is assumed to be below 1,000. From ASME Section III, Appendix I, Figure I-9.2.1, the fatigue allowable stress intensity amplitude, S_a , of the alternating stress component (one-half of the alternating stress range) for 1,000 cycles is 119,000 psi. This value, when corrected for temperature by multiplying by the ratio of elastic modulus, $27.0(10)^6/28.3(10)^6$, gives a fatigue allowable alternating stress intensity amplitude of 113,534 psi. The non-fatigue allowable stress intensity range, from ASME Section III, Division 1, Section NB-3222.2, however, is 94,200 psi ($3.0S_u$). Since one-half of this range is less than the fatigue allowable alternating stress intensity, the non-fatigue allowable stress criterion will govern.

Areas of stress concentrations for the package occur at the inner vessel and outer cask closure bolts. The maximum cyclic stress in the outer cask closure bolts is due to the preload, since the normal conditions of transport loads are less than the preload. From Equation 5.4 of Reference (9), the preload, P , is given by:

$$P = T/KD$$

Where:

T = torque, in-lb
 K = torque coefficient
 D = nominal bolt diameter, in

Outer Cask Closure Bolts:

For these bolts, the preload torque is 1,300 to 1,400 ft-lbs, or a maximum of 16,800 in-lb. The nominal bolt diameter is 1 5/8 inches, hence $D = 1.625$ in. The torque coefficient for cadmium plated bolts is generally taken as $K = 0.13$. Therefore, maximum bolt load is:

$$\begin{aligned} P &= 16,800 / (0.13) (1.625) \\ &= 79,527 \text{ lbs} \end{aligned}$$

From Reference (32), the tensile stress area of the bolt is 1.7521 in². Stress in the bolts is:

$$\begin{aligned} S &= 79,527 / 1.7521 \\ &= 45,389 \text{ psi} \end{aligned}$$

From ASME NB-3232.3(c), the fatigue strength reduction factor is assumed to be 4.0. In addition, the modulus of elasticity at 300° F is $26.7(10)^6$ psi (ASME Section III, Table I-6.0). Since Figure 2.1-1 is based upon a modulus of elasticity of $30(10)^6$ psi, the equivalent stress range is given by:

$$\begin{aligned} S_{\text{range}} &= (45,389) (4) [30(10)^6 / 26.7(10)^6] \\ &= 203,996 \text{ psi} \end{aligned}$$

The alternating component is one-half of the stress range, or 101,998 psi ($203,996/2$). From Figure 2.1-1, using the $3.0S_n$ curve, the allowable number of operating cycles is greater than 600. Assuming a maximum of 52 load/unload cycles per (one per week) year for the bolts, a 11.5 year life is projected. Operational procedures will probably specify replacement of these bolts every 5 years, thus assuring a fatigue failure margin of safety in excess of two.

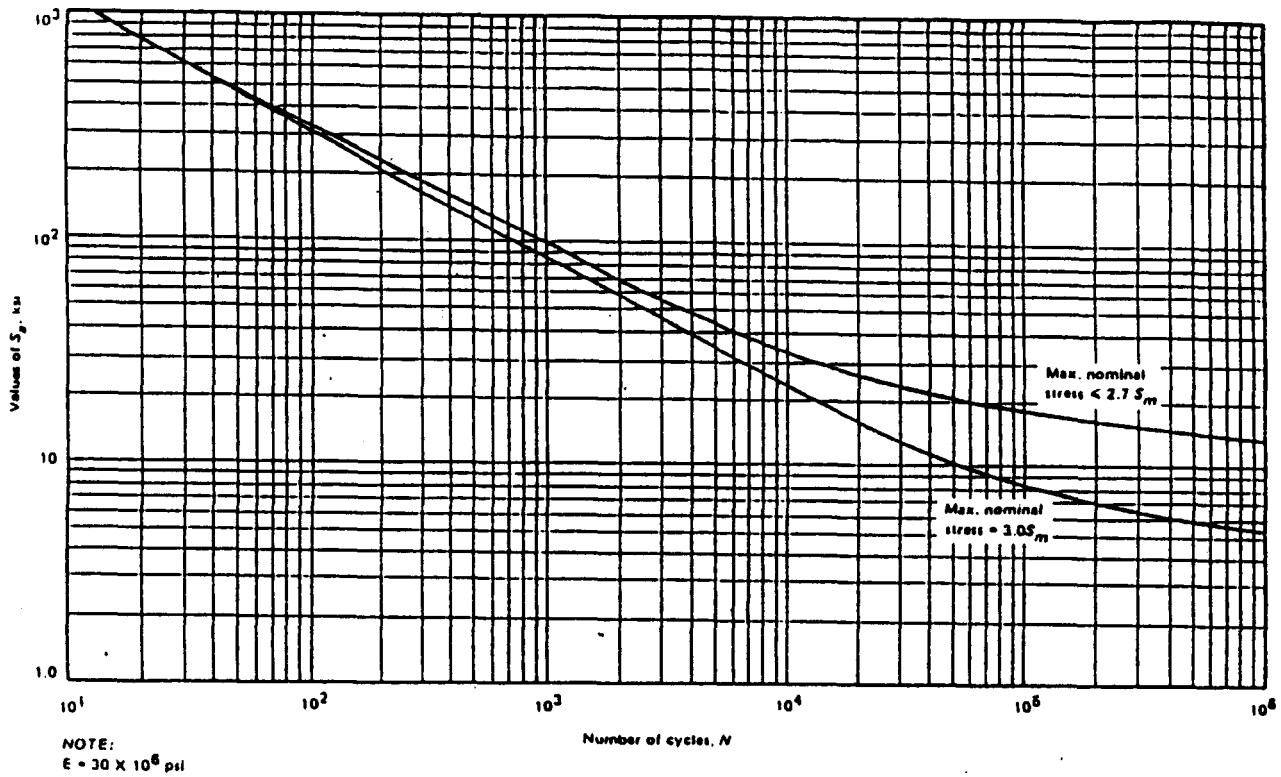


FIG. I-9.4 DESIGN FATIGUE CURVE FOR HIGH STRENGTH STEEL BOLTING
 FOR TEMPERATURES NOT EXCEEDING 700°F
 Table I-9.1 Contains Tabulated Values and a Formula for Accurate
 Interpolation of These Curves

FIGURE 2.1-1
 ASME Section III, Figure I-9.4

2.1.2.2.2.2 Normal Vibration over the Rail

Fatigue concerns associated with normal vibrations over the rail utilize input information from the recent report of Reference (43) and are addressed in Section 2.6.5.

2.1.2.2.2.3 Extreme Total Stress Intensity Range

Per paragraph C.7 of Regulatory Guide 7.6, "the extreme total stress intensity range (including stress concentrations) between the initial state, the fabrication state, the normal operating conditions, and the accident conditions should be less than twice the adjusted value (adjusted to account for modulus of elasticity at the highest temperature) of S_u at 10 cycles given by the appropriate design fatigue curves." Evaluation of stresses per this criteria is addressed in Section 2.7.3.

2.1.2.2.3 Buckling

Buckling, per paragraph C.5 of Regulatory Guide 7.6, is an unacceptable failure mode for the containment vessels. The intent of this provision is to preclude large deformations which would compromise the validity of linear analysis assumptions and quasi-linear stress allowables, as given in Paragraph C.6 of Regulatory Guide 7.6.

Buckling criteria per Regulatory Guide 7.6 is applicable to the 1.13 inch stainless steel inner containment shell of the NuPac 140-B Cask. However, per Nuclear Packaging general design practice, buckling criteria per Regulatory Guide 7.6 is also applied to the 1.50 inch stainless steel outer shell. For reference purposes the principal geometric features of these shells are as follows:

<u>Shell</u>	<u>Dimension (inches)</u>		
	Mean		
	Radius	Thickness	Length
	<u>(R)</u>	<u>(t)</u>	<u>(L)</u>
Inner Shell	29.065	1.13	177
Outer Shell	33.875	1.5	177

To demonstrate that buckling will not occur for these two shell geometries, the methodology of ASME Code Case N-284, Reference (10) is directly applied. The basic steps involved are summarized as follows. Details of the buckling assessments are presented in Appendix 2.10.3.

1. Theoretical elastic buckling stresses are determined for hoop and axial compression and in-plane shear loadings using classical theory.
2. Capacity reduction factors are applied which account for the difference between classical theory and predicted instability stresses for fabricated shells.
3. Plasticity reduction factors are applied for those cases where elastically determined buckling stresses are above the proportional limit.
4. Elastic and inelastic buckling checks which employ appropriate factors of safety and appropriate interaction equations are made using worst case applied compressive and in-plane shear stresses.

Consistent with Regulatory Guide 7.6 philosophy, Factors of Safety corresponding to ASME Code Level A and D Service conditions are employed respectively for normal and accident condition loadings. The applicable factors of safety are 2.0 for normal conditions and 1.34 for accident conditions as specified in Code Case N-284.

In addition to the two primary cask shells mentioned above, buckling considerations are also incorporated into the payload basket analyses of Section 2.6.8 and 2.7.7. Furthermore, buckling considerations are incorporated into the payload fuel assembly analysis of Section 2.9 as demonstrated in Reference (11).

2.1.2.3 Miscellaneous Design Criteria

Other structural criteria exist for the NuPac 140-B Cask which are not specified in 10 CFR 71 or any of the NRC Regulatory Guides. For example, cask criteria includes a limit on maximum weight of 200,000 pounds, design life of 25 years, both from Reference (1), and maximum drop impact 'g' load limits of 83 g's axially and 63 g's laterally on fuel rods per Reference (11).

2.2 Weights and Centers of Gravity

The total weight of the NuPac 140-B Cask and Fuel Package, including a maximum payload of 52 BWR fuel assemblies is 180,783 lbs. This value increases to 190,108 lbs "on the hook" which includes 7000 lbs of water and 2325 lbs for the lift fixture. Thus, the 200,000 lb maximum "on the hook" weight specified in Reference (1) has been satisfied. Furthermore, the cask configuration with or without payload is nearly symmetrical. Therefore, the center of gravity (C.G.) is very near the geometric center of the cask.

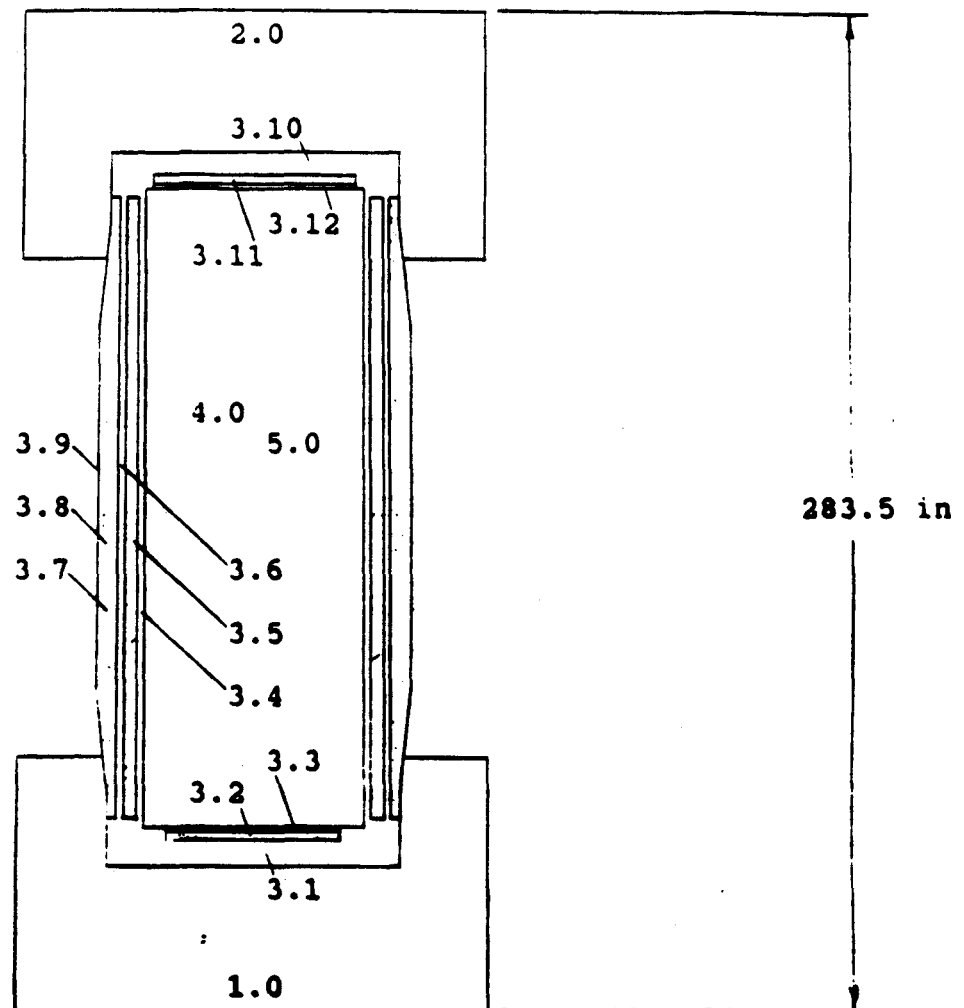
It should be mentioned that cask weight with impact limiters is needed for structural analysis, where the water and lift fixture weights mentioned above are not applicable. Thus, with each impact limiter weighing 11,188 lbs, the maximum total weight of the cask for structural analysis is $180,783 + 2 \times 11,188 = 203,159$ lbs. A detailed weight break down is presented in Table 2.2-1, which refers to Figure 2.2-1 for a component numbering schematic.

TABLE 2.2-1
Weights and Centers of Gravity

COMPONENT	WEIGHT (lbs)	DISTANCE TO C.G. (inch) (1)
1.0 Bottom Impact Limiter	11,188	27.94
2.0 Top Impact Limiter	11,188	255.56
3.0 Cask	132,087	141.75
3.1 Bottom Closure Plate	15,105	45.24
3.2 Bottom Boro-silicone Neutron Shield Disk	272	49.75
3.3 Bottom Inner Steel Disk	281	51.25
3.4 Inner Containment Shell	10,541	141.75
3.5 Lead Gamma Shield Shell	50,037	141.75
3.6 Outer Shell (1.5 in)	16,338	141.75
3.7 Heat Transfer Fins	9,429	141.75
3.8 Boro-silicone neutron shield segments	10,835	141.75
3.9 Thermal Shield and Load Collars	3,810	141.75
3.10 Top Closure Plate	14,784	238.26
3.11 Top Boro-silicone Neutron Shield Disk	324	233.75
3.12 Top Inner Steel Disk	331	232.25
4.0 Baskets		
4.1 BWR	16,508	141.75
4.2 FWR	14,225	141.75
5.0 Payload		
5.1 52 BWR Fuel Assemblies (2)	32,188	141.75
5.2 21 FWR Fuel Assemblies	31,815	141.75
TOTAL CASK WEIGHTS & C.G.	203,159 (BWR)	141.75
	200,503 (FWR)	141.75

Notes: See next page

- (1) Reference datum is the base of the bottom impact limiter
- (2) The weight of the fuel channels and corresponding effects on the structural analysis will be included during the final design phase of the contract. The fuel channels will increase this weight to 35,568 lbs.



NOTE: See Table 2.2-1 for identification of components.

FIGURE 2.2-1

NuPac 140-B Cask Schematic for Component Weight Breakdown

2.3 Mechanical Properties of Materials

The NuPac 140-B Cask is fabricated primarily from Type XM-19 austenitic stainless steel, with polyurethane foam energy absorbers, high strength ferritic steel bolts, copper, borated-silicone and lead. The general arrangement drawings presented in Appendix 1.3.2 define the specific material used for each item of the cask. Table 2.3-1 presents material properties for type XM-19 and 17-4 PH, Type 630, stainless steels; ASTM Type A-320 and Inconel 718, Type NO7718 fasteners; SB-152 copper; and lead per QQ-L-171c used in the NuPac 140-B Cask. Material properties are linearly interpolated between the temperature values shown.

Polyurethane foam stress-strain properties are presented in Table 2.3-2. These values are the required limits which the foam must satisfy to be acceptable for use in the impact limiters. Quality Assurance testing will be performed for each pour of the foam as discussed in Section 8.1.4.

Table 2.3-2 presents room temperature data (approximately 70 to 75°F) for foam with density of about 15 pounds per cubic foot (pcf) used in the impact limiters. The nominal, room temperature stress-strain curve is bounded by +/- 15% up to about 65% strain, and up to +/-20% from 65 to 80% strain to cover potential deviations in stress-strain values from the desired nominal stress-strain curve. The mean of the results from a multiple number of foam tests is used as the nominal foam curve. Two standard deviations of stress from the nominal curve is used to determine the +/-15% bounds. The wider bounds of +/- 20% at 80% strain are based on the foam manufacturer's recommendation. In addition, multiple tests were performed upon foam samples to determine stress-strain characteristics at -20°F up to 300°F. Part of the data (-20°F to 180°F) are obtained from Nuclear Packaging proprietary tests and the remainder from tests reported in Reference (5). Utilization of these data for specific NuPac 140-B Cask application is described in Appendix 2.10.2.

These temperatures are the required bounding temperatures for the NuPac 140-B Cask impact limiters. For hot foam, values were interpolated from

the data of Reference (5) for the impact limiter thermal gradients computed in section 3.0. The results of the interpolations are presented in Appendix 2.10.2.

To adequately address fabrication stresses due to lead shrinkage on the inner and outer cask, stress-strain and creep data are required for lead. The lead used in the cask is per Federal Specification QQ-L-171e. The composition of this lead is primarily lead (99.9% minimum) and copper (0.04 to 0.08%). Stress-strain and creep data are therefore extracted from Reference (12) for a copperized lead with a composition of 99.94% lead and 0.0577% copper. Stress-strain curves and creep data from the reference are presented in Figures 2.3-1 through 2.3-6.

Other materials used in the package which are non-structural in nature include the elastomeric O-ring seals and the borated-silicone utilized in the neutron shield.

Table 2.3-1
Mechanical Properties of Materials

Material Spec	Grade or Type	Temp °F	Strength (ksi)			Elastic Modulus 10 ⁶ psi	Coefficient of Thermal Expansion 10 ⁻⁶ in/in°F
			Yield S _y	Ult S _u	Allowable S _a		
A) Austenitic Stainless Steel (2)							
SA-240	XM-19	-100	-	-	-	29.1	-
SA-182		70	-	-	-	28.3	8.24
(22CR-13Ni-5M)		100	55.0	100.0	33.3	28.1	8.30
		200	47.0	99.5	33.2	27.6	8.48
		300	43.4	94.3	31.4	27.0	8.65
		400	40.8	90.7	30.2	26.3	8.79
		500	38.8	89.1	29.7	25.8	8.92
		600	37.3	87.8	29.2	25.3	9.03
Data Source Ref. (6),	Table:	I-2.2	I-3.2	I-1.2	I-6.0	I-5.0	
B) Ferritic Stain-Steel							
SA 564 (17Cr-4Ni-4Cu)	630	-100	-	-	-	29.1	-
		70	-	-	45.0	28.3	5.89
		100	105.0	135.0	45.0	28.1	5.89
		200	97.1	135.0	45.0	27.6	5.90
		300	93.0	135.0	45.0	27.0	5.90
		400	89.8	131.4	43.8	26.3	5.91
		500	87.0	128.3	42.8	25.8	5.91
		600	84.7	126.7	42.1	25.3	5.93
Data Source Ref. (6),	Table:	I-2.1	I-3.1	I-1.1	I-6.0	I-5.0	

Table 2.3-1 - (Continued)
Mechanical Properties of Materials

Material Spec	Grade or Type	Temp °F	Strength (ksi)			Elastic Modulus 10 ⁴ psi	Coefficient of Thermal Expansion 10 ⁻⁶ in/in°F
			Yield S _y	Ult S _u	Allowable S _a		
C.1) Low Alloy Steel							
ASTM A-320 (Fasteners)	L43	70	105.0	125.0	-	27.8	6.20
		100	105.0	125.0	35.0	-	6.27
		200	99.0	-	33.0	27.1	6.34
		300	95.7	-	31.9	26.7	6.78
		400	91.8	-	30.6	26.1	6.98
Data Source Ref. (6),		Table:	I-1.3	I-1.3	I-1.3	I-6.0	I-5.0
C.2) Inconel 718							
ASTM A-637 (Fasteners)	N07718	70	150.0	185.0	-	29.0	7.05
		100	150.0	185.0	50.0	-	7.08
		200	144.0	-	48.0	28.3	7.22
		300	140.7	-	46.9	27.8	7.33
		400	138.3	-	46.1	27.6	7.45
Data Source Ref. (6),		Table:	I-1.3	I-1.3	I-1.3	I-6.0	I-5.0

Table 2.3-1 - (Continued)
Mechanical Properties of Materials

Material Spec	Grade or Type	Temp °F	Strength (ksi)			Elastic Modulus 10 ⁶ psi	Coefficient of Thermal Expansion 10 ⁻⁶ in/in°F
			Yield S _y	Ult S _u	Allowable S _a		
D) Copper Plates SB-152	C10200	-100	-	-	-	17.5	-
		70	10.0	30.0	-	17.0	-
	C10400	100	-	-	6.7	-	-
		150	-	-	5.7	-	-
		200	-	-	5.5	16.6	-
		250	-	-	5.1	-	-
		300	-	-	5.0	16.3	-
		Data Source Ref. (6).		Table:	I-8.4	I-8.4	I-8.4

Material Specification	Grade or Type	Temp °F	Strength (ksi)					Elastic Modulus 10 ⁶ psi	Coefficient of Thermal Expansion 10 ⁻⁶ in/in°F
			Proportional S _p (tons)	Yield S _y (comp) (tons)	Ultimate S _u (comp) (tons)				
E) Lead QQ-L-171c	A or C	-99	-	-	-	-	-	2.50	13.28
		70	-	-	-	-	-	2.34	16.07
		100	.276	.213	.584	.490	1.570	2.30	16.21
		175	.293	.107	.509	.428	1.162	2.20	16.58
		250	.277	.107	.490	.391	.844	2.09	16.95
		325	.189	.093	.311	.320	.642	1.96	17.54
		440	-	-	-	-	-	1.74	18.50
		620	-	-	-	-	-	1.36	20.39

Notes:

1. Other properties

	<u>Density</u>	<u>Poissons Ratio</u>	<u>Melting Point</u>
- Steel	0.283 lb/m ³	0.30	2,600°F
- Lead	0.41 lb/m ³	0.45	620°F
- Copper	0.325 lb/m ³	0.30	1,980°F

2. Data for XM-19 is for plate and is used in the report for XM-19 sheet also. Material specifications SA-240 requires 60 KSI minimum yield for XM-19 sheet but this higher strength is not used in computing margins of safety.

TABLE 2.3-2

Polyurethane Foam Crush Stress vs. Strain

(Room Temperature 75°F)

15 PCF maximum density

Crush Stress (psi)

Strain (in/in)	Upper	Nominal	Lower
	Bound		Bound
.10 (1)	739	642	546
.20	744	647	550
.30	798	694	590
.40	902	785	667
.50	1098	954	811
.60	1538	1337	1132
.65 (1)	1983	1724	1466
.70 (2)	2634	2251	1868
.75 (3)	4047	3430	2812
.80 (4)	6818	5682	4545

Notes: (1) Up to and including 65% strain the variation is +/- 15%

(2) +/- 17% variation

(3) +/- 18% variation

(4) +/- 20% variation

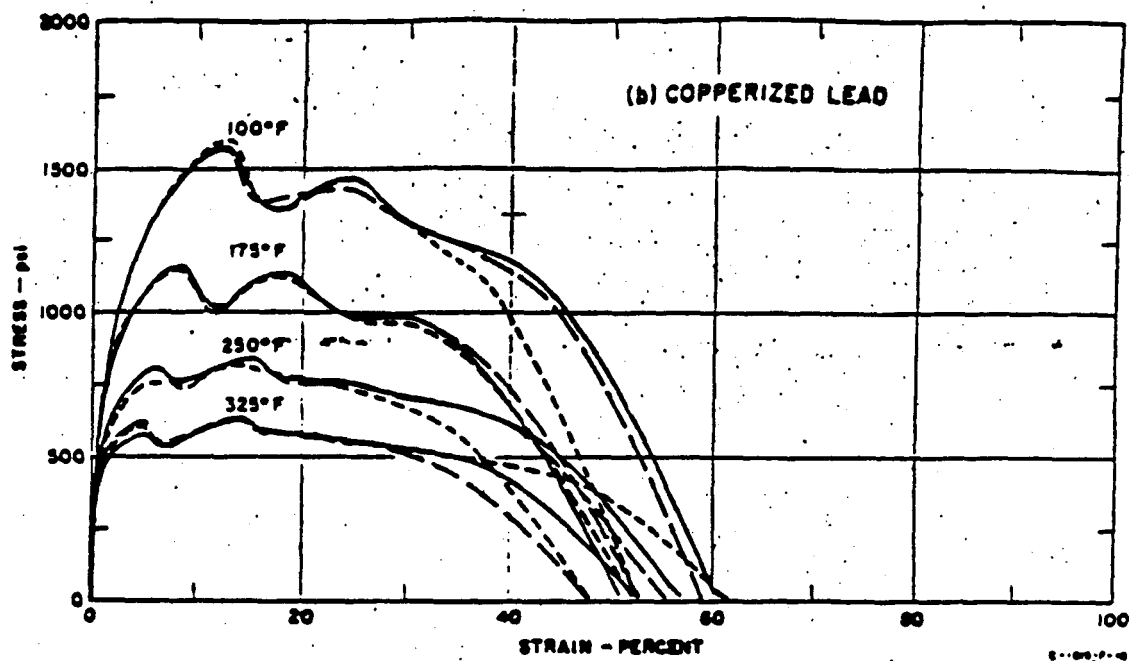


FIGURE 2.3-1

Tensile Stress-Strain Curves to Failure at a Strain Rate of 0.05 In/In/Min

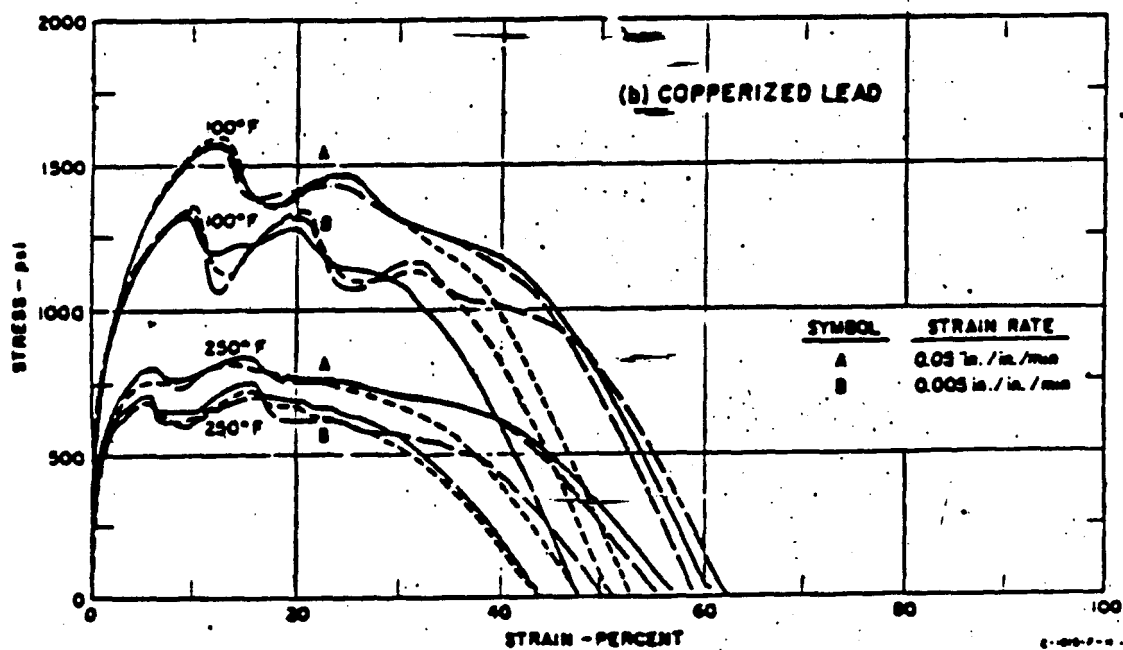


FIGURE 2.3-2

Effect of Strain Rate on the Tensile Stress-Strain Curves at 100 and 250°F

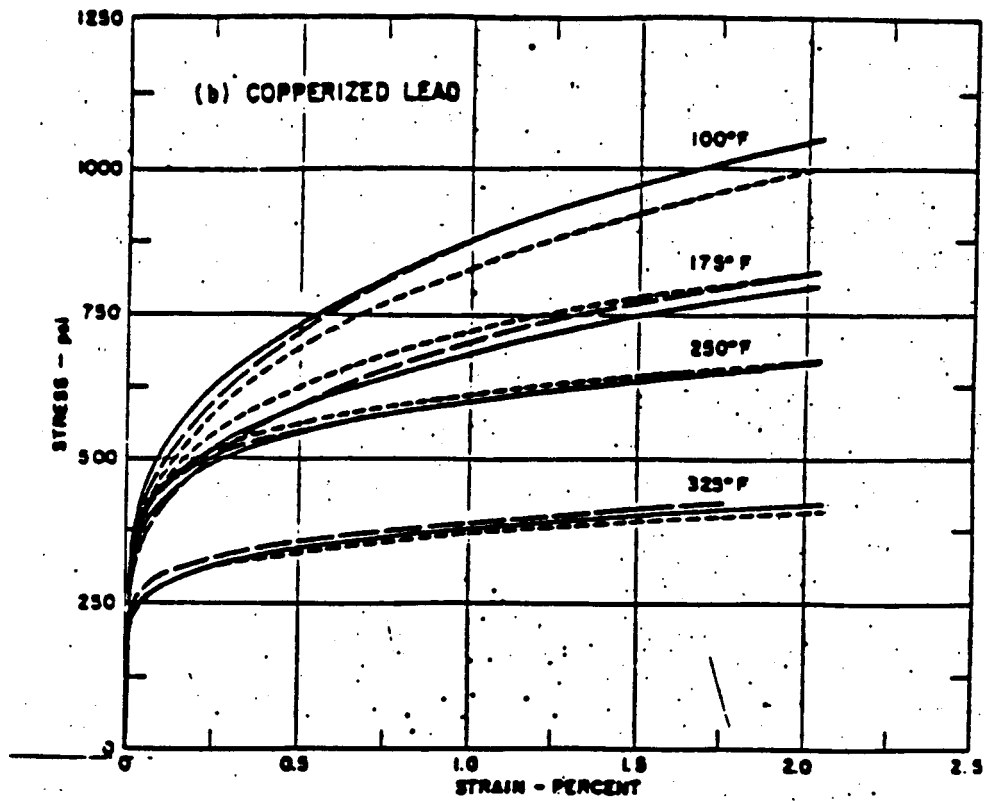


FIGURE 2.3-3

Tensile Stress-Strain Curves to 2% Strain at a Strain Rate of 0.005 In/In/Min

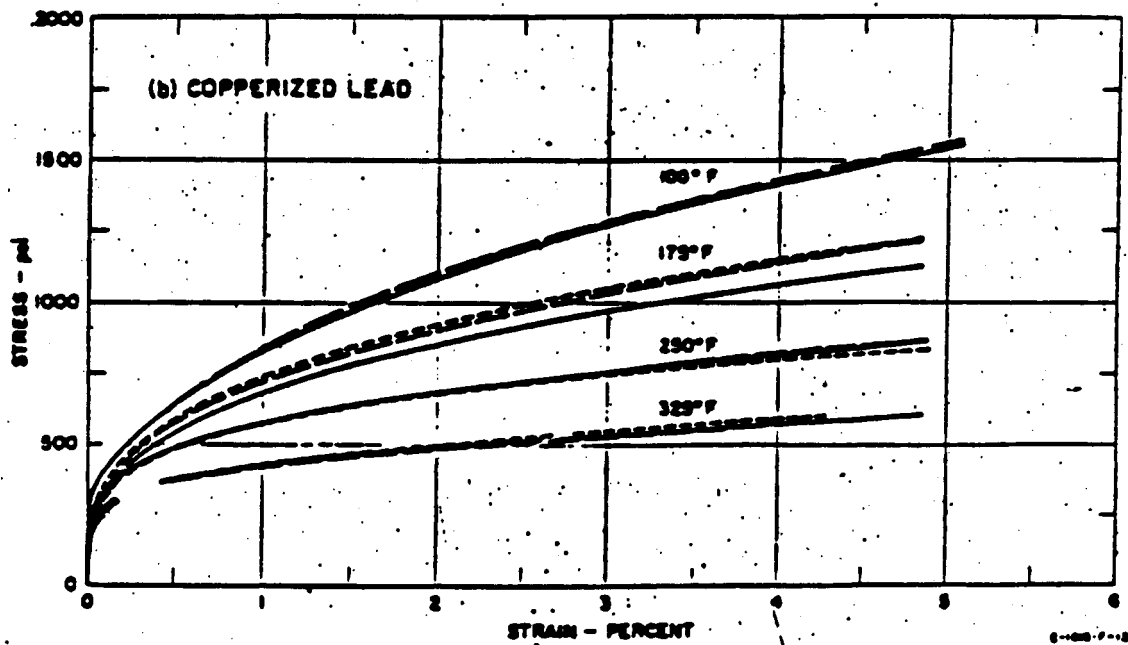


FIGURE 2.3-4

Compression Stress-Strain Curves to 5% Strain at a Strain Rate of 0.005 In/In/Min

FIGURE 2.3-5

Total Strain vs. Creep Time for Copperized Lead

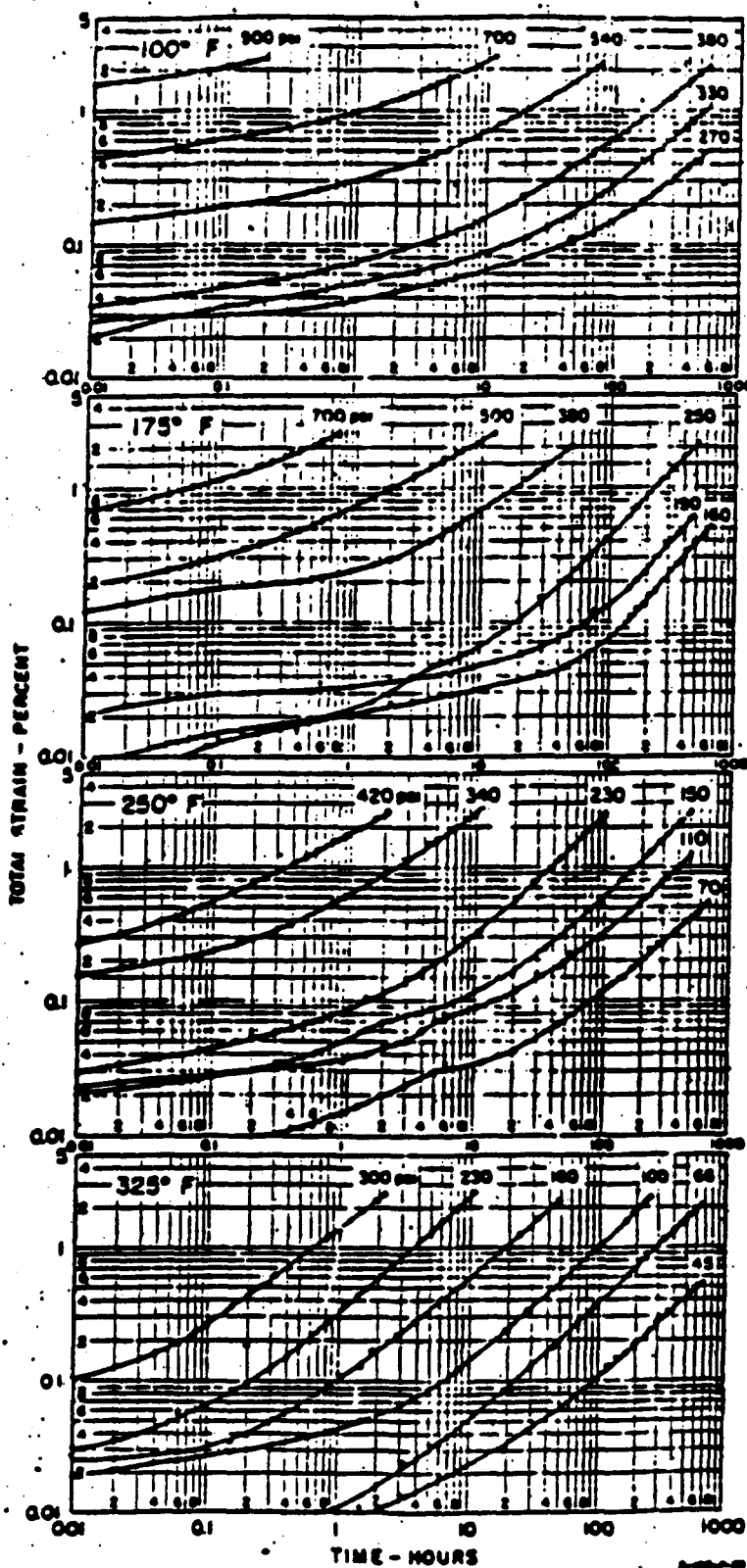
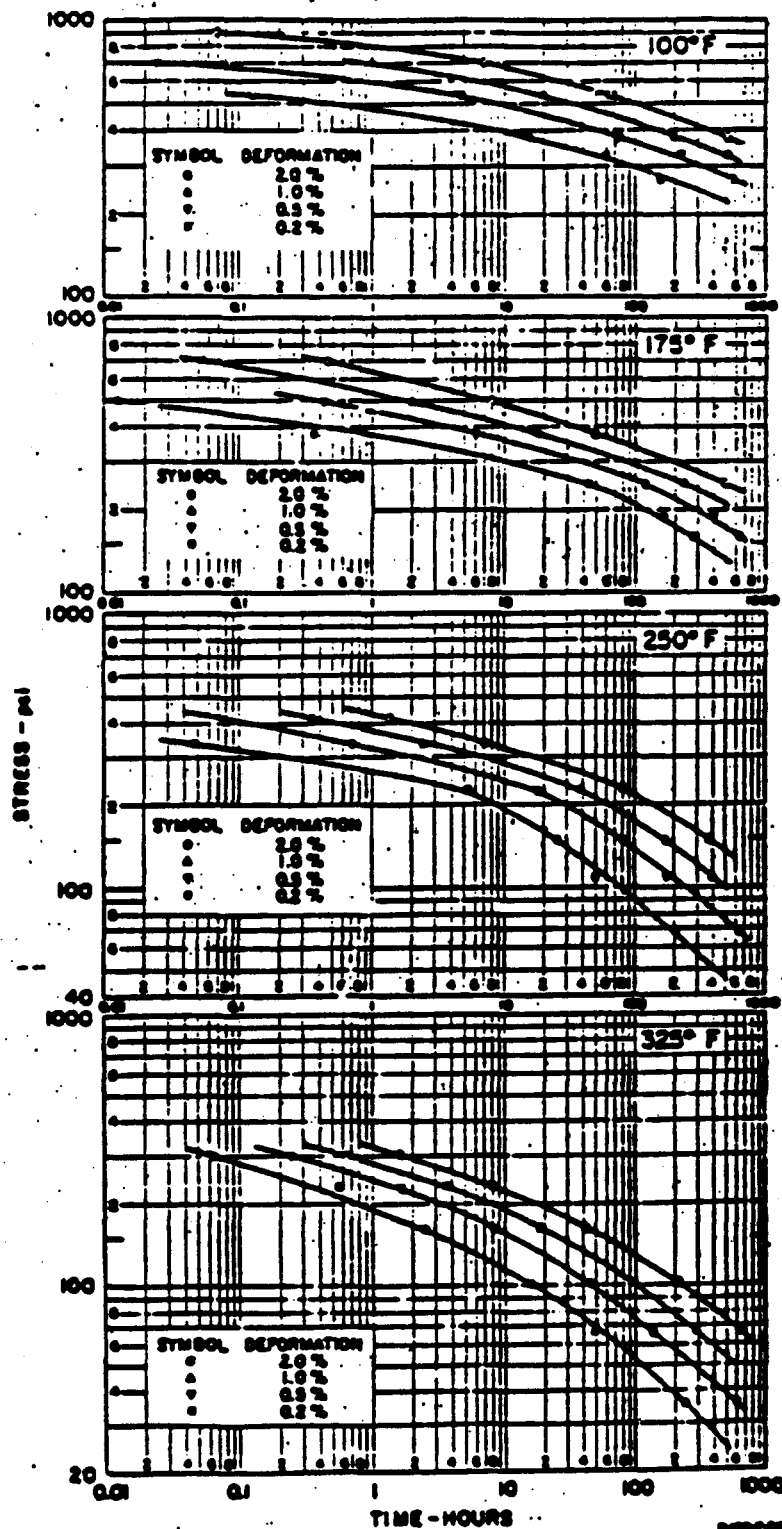


FIGURE 2.3-6

Stress vs. Creep Time at Constant Strain Values of 0.2, 0.5, 1.0, and 2.0% for Copperized Lead



2.4 General Standards For All Packages

This section demonstrates that the general standards for all packages are met.

2.4.1 Minimum Package Size

The minimum transverse dimension of the package (without impact limiter) is 81.77 inches, and the minimum longitudinal dimension is 203.50 inches.

2.4.2 Tamper-Indicating Feature

A 'lock wire', or equivalent, will be utilized between two tie points on the closure-end of the package during a loaded shipment. Specifically, one tie point will be affixed to the outer cask thermal shield, and one tie point will be affixed to the closure-end impact limiter, as illustrated on the general arrangement drawings in Appendix 1.3.2. Failure of said device will indicate purposeful tampering in accordance with 10 CFR 71.43(b).

2.4.3 Positive Closure

Inadvertent opening of the cask closures cannot occur for the NuPac 140-B Cask. Following installation of the cask payload, the outer cask closure lid is secured via thirty-two (32), 1 5/8 - 8 UN bolts. Impact limiters are then installed on each end of the cask via eight (8), 1 1/2 - 6 UNC bolts. When installed, the impact limiters eliminate access to the outer cask closure. With this double containment closure and the presence of the impact limiters, inadvertent opening of the cask cannot occur.

2.4.4 Chemical and Galvanic Reactions

The materials from which this package is fabricated (i.e., stainless steel, carbon steel, lead, copper, polyurethane foam, and borated-Silicone criticality moderator) will not cause significant chemical, galvanic, or other reactions in air, nitrogen, or water environments. The inner surfaces of the impact limiter skins are treated with a bond release agent so that bonding of polyurethane foam and steel does not occur.

Galvanic reactions may be avoided between metals by designing for 'similar' metal contact as defined in Reference (44). Similar metals are those which fall in the same group (I, II, III or IV) as depicted in Table 2.4-1, which was taken from Reference (44). As shown, copper and stainless steel both fall within

TABLE 2.4-1
Groupings of Similar Metals

I	II	III	IV	
magnesium and its alloys	aluminum and its alloys, zinc, cadmium	iron, lead, tin and their alloys (except stainless steel)	copper and its alloys, nickel and its alloys, chromium, stainless steel, gold, silver	titanium and its alloys, platinum, rhodium, cobalt, graphite tungsten

Note: (1) Contact between a member of any one group and another member of the same group shall be considered as similar.

(2) Where reference is made to a metal in a particular group, the reference applies to the metal on the surface of the part; that is, aluminum means aluminum castings as well as aluminum coating such as aluminum metal spray or hot dip aluminum.

group IV. Thus, they are similar, and galvanic corrosion will not be a problem. It should be mentioned that the more cathodic metals are towards group IV, and the more anodic are towards group I. Where materials cannot readily be inspected, Nuclear Packaging structural design practice is to make the more critical material cathodic.

2.5 Lifting and Tie Down Standards For All Packages

2.5.1 Lifting Devices

The lifting devices structural analysis considers the trunnions and their attachment area, and the basket lift points.

2.5.1.1 Trunnions Lifting Devices

The trunnions and their attachments in the cask are designed for the maximum permissible lift load of 200,000 lbs. The appropriate safety factor for lift is 3.0 on yield as specified in 10 CFR 71.45. With this safety factor, the total structural design load is:

$$(200,000) (3.0) = 600,000 \text{ lbs.}$$

Assuming the load is evenly divided between two opposite trunnions, the yield load per trunnion is 300,000. lbs. The structural design temperature is 350°F.

Minimum margins of safety for the trunnions, bolts and cask socket and threads for the above loads for lift are:

Component	M.S.
Trunnion body	+ 0.43
Bolts	+ 0.40
Cask socket	+ 0.12
Cask threads	+ 0.40

Stress and margin calculations for the above components are described in the following paragraphs.

Bolt head bearing stress is calculated using the maximum bolt load. The maximum bolt load is 59,710 lbs, determined in the Bolt Load Section which follows. The bearing stress (σ_b) is calculated as:

$$\sigma_b = P/A$$

where,

$$\begin{aligned} P &= \text{Maximum Bolt Load} \\ &= 59,710 \text{ lbs} \end{aligned}$$

$$\begin{aligned} A &= \text{Bolt head bearing area} \\ &= \pi(D_o^2 - D_i^2)/4 \\ &= 0.933 \text{ in}^2 \end{aligned}$$

Then

$$\begin{aligned} D_o &= 1.479 \text{ in.} \\ D_i &= 1.00 \text{ in.} \end{aligned}$$

$$\begin{aligned} \sigma_b &= 59,710/0.933 \\ &= 63,998 \text{ psi} \end{aligned}$$

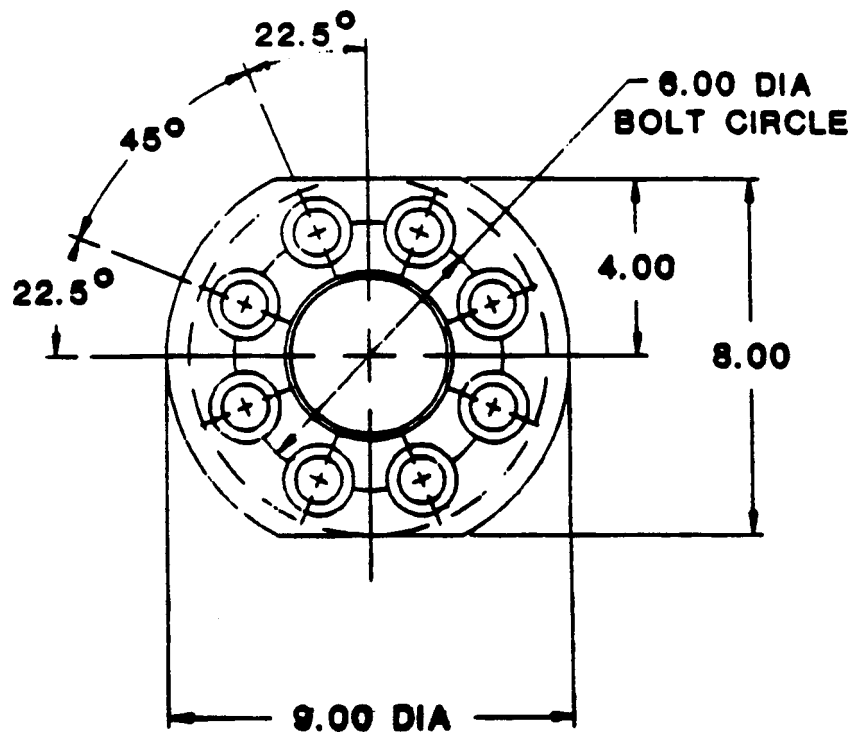
The trunnion loads and dimensions are shown in Figures 2.5.1-1 and 2.5.1-2. The trunnions bearing diameter is 8.0"; the material is 17-4 PH, Spec. SA-564, Grade 630. The trunnions are located in a 1.0" deep socket in the end rings, secured with 8 bolts each. Trunnion bolts are 1-8UNC bolts. The base material of the bolts is IN-718 Nickel Spec SB-637, Type NO7718, yield strength is 138ksi at 350°F.

TRUNNION BODY STRESSES: The primary loads on the trunnion body are a) bearing on the socket walls, b) bearing of the bolt head and c) shear at the minimum cross section. Because of the small length to diameter ratio of the trunnion, flexural stresses in the trunnion body itself can be neglected.

As the 17-4-PH material of the trunnion is considerably stronger than the XM-19 of the socket in the cask end ring, the socket bearing stresses will govern, and socket bearing on the trunnion can be neglected.

The minimum Margin of Safety (M.S.) is based on a bearing stress allowable (σ_b) equal to yield strength at the design temperature:

FIGURE 2.5.1-1
Lift Loads on Trunnion, Socket, and Bolts



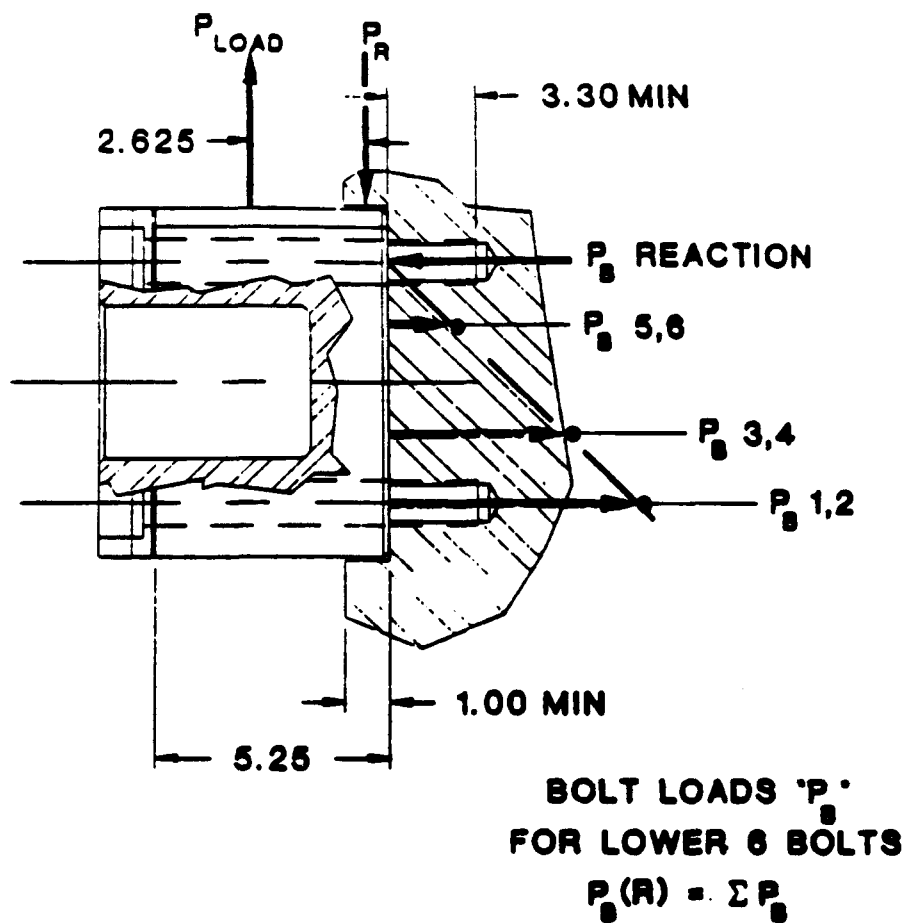


FIGURE 2.5.1-2

Lift Loads on Trunnion, Socket, and Bolts

$$\begin{aligned}
 \text{M.S.} &= (\sigma_s / \sigma_b) - 1 \\
 &= (91,400 / 63,998) - 1 \\
 &= + 0.43
 \end{aligned}$$

The trunnion shear stress (σ_s) is defined as the load over the minimum cross sectional area. The critical cross section is the area of the 8" O.D. body, less the 3.5" bore, less the 8 holes, of approximately 1.1" diameter.

$$\sigma_s = V / A$$

where,

$$\begin{aligned}
 V &= \text{Shear Load} \\
 &= 300,000 \text{ lbs.}
 \end{aligned}$$

$$\begin{aligned}
 A &= \text{Critical Cross Section Area} \\
 &= \pi [8^2 - 3.5^2 - 8(1.1^2)] / 4 \\
 &= 33.04 \text{ in}^2
 \end{aligned}$$

Therefore

$$\begin{aligned}
 \sigma_s &= 300,000 / 33.04 \\
 &= 9,780 \text{ psi}
 \end{aligned}$$

The allowable shear stress (σ_a) is defined as:

$$\begin{aligned}
 \sigma_a &= 0.6(S_y) \\
 &= 0.6(91,400) \text{ at } 350^\circ\text{F} \\
 &= 54,840 \text{ psi}
 \end{aligned}$$

The trunnion body Margin of Safety (M.S.) on shear is:

$$\begin{aligned}
 \text{M.S.} &= (\sigma_a / \sigma_s) - 1 \\
 &= (54,840 / 9,780) - 1 \\
 &= + 4.61
 \end{aligned}$$

BOLTS LOADS: The applied moment (M) on the bolted connection is:

$$\begin{aligned}
 M &= 300,000 \text{ (2.625)} \\
 &= 787,500 \text{ in-lbs}
 \end{aligned}$$

Assuming the bolt loads vary linearly as shown in Figure 2.5.1-2, each bolt load can be expressed as a function of the load in bolt 1 (P_{B1}). The bolt forces are dependant on their distance from the upper most bolts where a hinge point is assumed. The bolt distances are calculated as:

$$\begin{aligned}
 d_1 = d_2 &= 2[R \cos(22.5^\circ)] \\
 &= 2[3 \cos(22.5^\circ)] \\
 &= 5.543 \text{ in.}
 \end{aligned}$$

$$\begin{aligned}
 d_3 = d_4 &= R \cos(22.5^\circ) + R \cos(67.7^\circ) \\
 &= 3 \cos(22.5^\circ) + 3 \cos(67.5^\circ) \\
 &= 3.920 \text{ in.}
 \end{aligned}$$

$$\begin{aligned}
 d_5 = d_6 &= R \cos(22.5^\circ) - R \sin(22.5^\circ) \\
 &= 3 \cos(22.5^\circ) - 3 \sin(22.5^\circ) \\
 &= 1.624 \text{ in.}
 \end{aligned}$$

The bolt forces are expressed in terms of P_{B1} (Bolt force in bolt 1) below:

$$\begin{aligned}
 P_{B2} &= P_{B1} \\
 P_{B3} = P_{B4} &= (3.920/5.543) P_{B1} = 0.7072 P_{B1} \\
 P_{B5} = P_{B6} &= (1.624/5.543) P_{B1} = 0.2930 P_{B1}
 \end{aligned}$$

By equating the moments, the maximum bolt load, P_{B1} , can be calculated as shown:

$$787,500 = 2(5.543)P_{B1} + 2(3.920)(0.7072)P_{B1} + 2(1.624)(0.2930)P_{B1}$$

Then

$$P_{B1} = 44,790 \text{ lbs}$$

A bolt load of 46,000 lbs is conservatively used for design purposes.

Bolt preload stresses are calculated considering the relative flexibilities of the trunnion and the bolts. When the flexibilities are considered, only a portion of the live load is added to the preload.

The preload torque (T), for the preliminary design, is called out as 440 ft-lbs, ± 15 ft-lbs, giving a maximum preload of 455 ft-lbs. For the 1.0" bolts, the friction factor " K_f " ranges between 0.10 and 0.15. Therefore, the range of the axial preload "F" is:

$$\begin{aligned} F_{\max} &= T / [Dia (K_f)] \\ &= (455) (12) / (1.0) (0.10) \\ &= 54,600 \text{ lbs} \end{aligned}$$

$$\begin{aligned} F_{\min} &= T / [Dia (K_f)] \\ &= (425) (12) / (1.0) (0.15) \\ &= 34,000 \text{ lbs} \end{aligned}$$

The $F_{\min}$ value being less than P_{B1} indicates unclamping at the lower pair of bolts. However, this value includes a design factor of safety of 3.0. Unclamping under actual loads will not occur.

The maximum bolt stress, (σ_t), is the bolt load ($F_{\max}$), divided by the tensile area (A_t). The minimum bolt tensile area is 0.606 in², per Reference (41).

$$\begin{aligned} \sigma_t &= F_{\max} / A_t \\ &= 59,711 / 0.606 \\ &= 98,533 \text{ psi} \end{aligned}$$

The minimum Margin of Safety (M.S.) is the allowable yield stress (σ_y) divided by the maximum bolt stress.

$$\begin{aligned} \text{M.S.} &= \sigma_y / \sigma_t - 1 \\ &= 138,000 / 98,533 - 1 \\ &= + 0.40 \end{aligned}$$

The maximum bolt load is calculated as:

$$\begin{aligned}
 F_{\max} &= (F_{\text{preload}} + \text{Liveload}) [K_{\text{bolt}} / (K_{\text{bolt}} + K_{\text{flange}})] \\
 F_{\max} &= (54,600 + 46,000) (3.8(10^6) / [3.8(10^6) + 30.4(10^6)]) \\
 F_{\max} &= 59,711 \text{ lbs}
 \end{aligned}$$

Joint separation is checked considering the minimum clamping load of the joint.

The margin of +0.40 on yield for the current preliminary design includes the 3.0 safety factor and a conservative analysis. The next size smaller bolt, a 7/8-9 UNC, could carry the load with a margin of approximately +0.07, or a 7/8-14 UNF would give a margin of approximately +0.176. These bolts can be considered in the final design. Lower strength bolts could also be considered in the final design.

BEARING AND THREAD STRESSES IN THE SOCKET: The bearing stresses (σ_b) in the socket are equal to the load, divided by the diameter bearing area (A_b):

$$\begin{aligned}
 \sigma_b &= F/A_b \\
 \sigma_b &= 300,000/8.0 \\
 &= 37,500 \text{ psi}
 \end{aligned}$$

The minimum Margin of Safety (M.S.) is the allowable yield strength (σ_y) divided by the maximum bearing stress (σ_b):

$$\begin{aligned}
 \text{M.S.} &= (\sigma_y / \sigma_b) - 1 \\
 &= 42,750 / 37,500 - 1 \\
 &= + 0.14
 \end{aligned}$$

The thread engagement is designed to provide sufficient strength to develop the full strength of the bolt. The bolt shear Margin of Safety is 0.40. To provide the same margin in the bolt engagement, the allowable shear stress (σ_s) is:

$$\sigma_s = 0.40 \sigma_y$$

where,

$$\begin{aligned}
 \sigma_y &= \text{Yield strength of parent material} \\
 \sigma_s &= 0.40 (42,750) \\
 &= 17,100 \text{ psi}
 \end{aligned}$$

The thread shear stress (σ_v) is calculated as:

$$\sigma_v = F/A_v$$

where,

$$\begin{aligned} F &= \text{Maximum Bolt Load} \\ &= 59,711 \end{aligned}$$

$$\begin{aligned} A_v &= \text{Thread Shear Area} \\ &= 2.3256 \text{ in}^2/\text{in} \quad (\text{Reference (32)}) \end{aligned}$$

The required bolt engagement is calculated by:

$$\sigma_s \leq \sigma_v$$

$$17,100 \leq 59,711/2.3256 \quad (L)$$

where,

$$\begin{aligned} L &= \text{required engagement length} \\ &\geq 1.5 \text{ in.} \end{aligned}$$

Therefore, an engagement length of 1.5 in. will develop the full capacity of the bolt. This provides a Margin of Safety for the threads of :

$$M.S. = + 0.40$$

2.5.1.2 Basket Lifting Points

The basket lifting points have been designed to support the full weight of a dry, loaded basket. Included in the design load is a safety factor on yield of 3.0. This has been done to assure that the basket can be safely lifted if required, although there is at present no formal requirement or operational need to lift a loaded basket. The structural margins above the design load are enough so that most reasonable handling loads, including a water filled basket and frictional forces within the cask, could be safely overcome.

The basket design shown in Figure 2.5.1.2-1 incorporates four reinforced lifting points. For the preliminary analysis, it is assumed that all of the load could be applied to either pair of diametrically opposed points. The PWR and the BWR baskets have similar structure at the lifting points. The stresses and margins are conservatively computed for the heavier BWR basket.

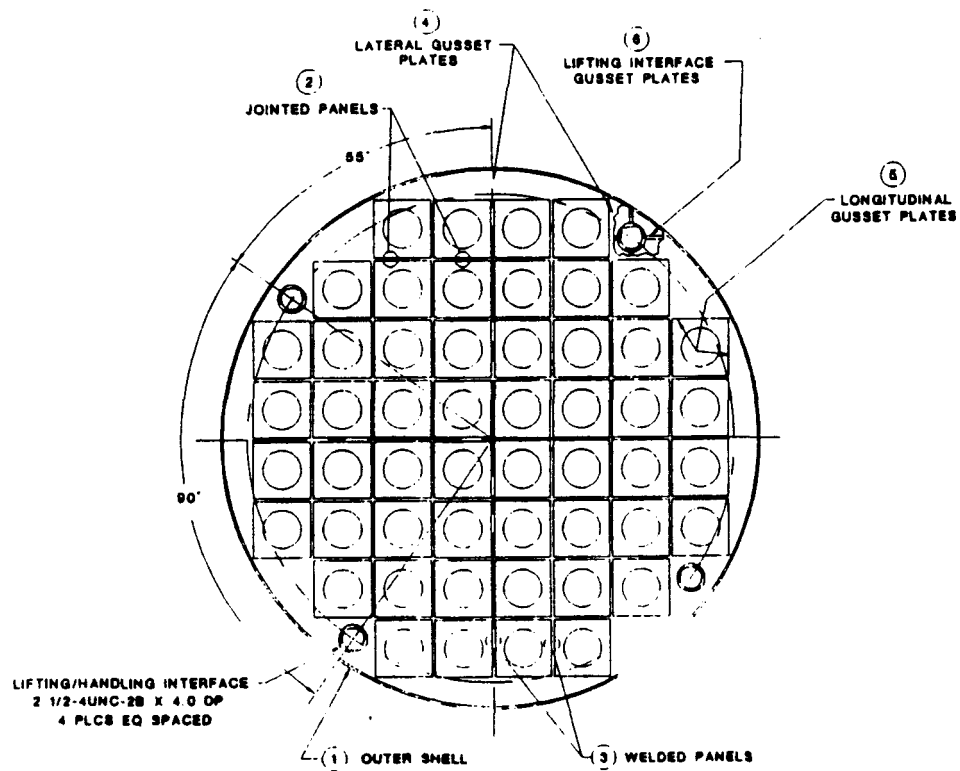


FIGURE 2.5.1.2-1
BWR Basket Structural Components

The BWR basket weighs approximately 50,000 lbs. The design load is therefore:

$$50000 (3.0) = 150,000 \text{ lbs.}$$

Load per lifting point is the design load divided by 2:

$$= 75,000 \text{ lbs/point}$$

Minimum margins of safety for the basket lifting points, including the 3.0 factor of safety on yield are:

BWR Basket lift loads: + 0.49

PWR Basket lift loads: + 0.60

The following paragraphs describe the stress and margin calculations for the lift loads.

The lift load path is from the long threaded fitting into the threaded pipes located at the upper surface of the basket. The load goes through the threads, into the body of the pipe, then through shear into the gusset plate weld joints (Item 6 on Figure 2.5.1.2-1) at each lifting point pipe. These gusset plates are welded to the 1/4 inch cylindrical outer shell of the basket. The load in the shell is transmitted to the major longitudinal and lateral gussets which make up most of the basket structure. Also welded to the basket are the outer set of truss core panels. At the bottom of the basket, fittings attached to the panels support the fuel elements. The jointed panels themselves are keyed together with transverse keys and to the outer welded panels to prevent any longitudinal motion or slippage when the loaded basket is lifted.

Preliminary calculations show that the critical area in the load path is the weld joining the gusset plate to the pipe and the outer shell.

The gusset plates are 3/8 inch thick XM-19, 6 inches long, chamfered to give 3/8 minimum weld root thickness.

Weld joint shear stresses (σ_v) are:

$$\sigma_v = 75,000 \text{ lbs}/[(3/8") (6") (2)] = 16,667 \text{ psi}$$

The shear stress allowable (σ_s) equals 0.6 times the yield stress:

$$\begin{aligned}\sigma_s &= 0.6 (41,500) \text{ (conservatively at } 400^\circ\text{F)} \\ &= 24,900 \text{ psi}\end{aligned}$$

Margin of safety on shear is calculated as:

$$\text{M.S.} = (24,900/16,667) - 1 = +0.49$$

This is the margin for the BWR basket, for the PWR basket, the margin is approximately + 0.60 considering the reduction in basket weight.

2.5.2 Tiedown Devices

The tiedown components of the cask assembly include the integral mounting lugs on the thermal shield and the reinforced areas of the thermal shield. These members transmit loads to the thermal shield itself and to the end rings.

In addition, steel reinforcing plates back up one pair of the longitudinal copper fins on each side of the cask. These steel plates carry the lateral tie down loads to the 1.5 inch thick outer structural containment shell.

This structural arrangement of the tie down lugs on a reinforced thermal shield presents several fundamental advantages. First, the arrangement does not compromise the borated-silicone shielding as would a structural member through the shielding to the outer containment shell. Secondly, in the unlikely event of loads well in excess of design, failure of the tie down components would not result in failure of the containment boundary. Thirdly, this configuration provides a direct load path from the lugs to the solid end rings for the axial load, which is the greatest tie down load. Thus, little moment is introduced, which reduces loads in the tie down components and especially in the containment structure. The vertical shear forces can also be carried in the thermal shield to the end ring directly, with little moment introduced to the structure. The side loads are carried by 3/8 inch steel plates, which run parallel to a set of copper fins. This is the only direct load path to the containment structure, and produces a distributed load on the containment shell. In the event of a severe overload on the side, the thin steel and copper plates would tend to buckle and yield before penetrating the 1.5 inch containment shell.

From a structural viewpoint, the current design meets all design criteria in an efficient and reliable manner.

The computer model used to analyze the tiedown structure is shown on Figure 2.5.2-1. The individual Computer Run solution printouts are included in Appendix 2.10.4.3

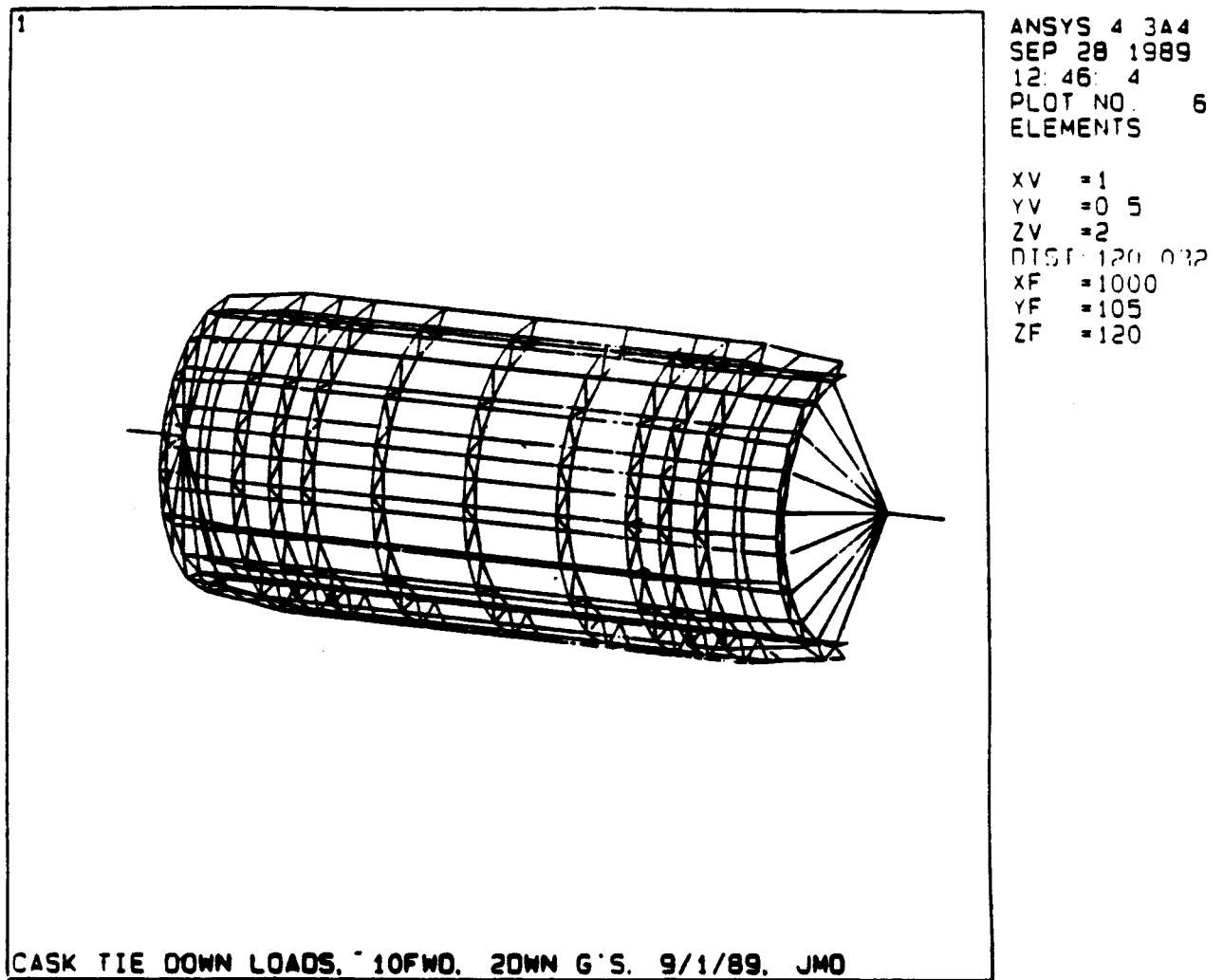


FIGURE 2.5.2-1
Cask Half Model for Tie Down Loads

Analysis for the primary tie down loads of 10, 2, and 5 g's are presented in Section 2.5.2.1. A separate analysis for vertical loads alone, which is needed to evaluate transportation induced vibration, is given in Section 2.5.2.2.

Analysis for the tie down cradle, and for the cradle's interface with the rail car are presented in Section B.

Tiedown loads are specified in 10 CFR 71.45, which states that tie down devices must be capable of withstanding, without generating stresses in excess of yield, the following combined loads applied through the center of gravity:

2 g's Vertical
10 g's Longitudinal
5 g's Lateral

Where longitudinal direction is in the direction of travel.

For the structural analysis the cask weight is 203,000 lbs. A design temperature of 300 °F is conservatively used for all tiedown devices.

Vertical loads, including overturning moment forces, are carried by the four tiedown lugs. Longitudinal forces are conservatively distributed to only two lugs.

The assumptions made for the preliminary analysis are described in the following sections. The resulting margins of safety for the tie down structure are:

Component	M.S.
Tie Down Lug	+ 0.33
Thermal shield	+ 0.56
Copper fin reinforcing plates	+ 0.66 (buckling)
	+ 1.76 (yield)

2.5.2.1 Analysis for Tie Down Loads

For use in the preliminary design and analysis phase, the computer model shown in the following figures was used, to aid in the development of a light weight structure which could meet all required structural tie down requirements.

The model shown represents one half of the cask. The cask internal areas are simplified in the model and do not directly affect the tie down stress and deflection. This model gives a good representation of the cask for longitudinal and vertical loads, but does not precisely handle side loads. Side load results are calculated separately and independently combined, where applicable, with the longitudinal and vertical load results.

The model is relatively complex, in order to accurately represent the load paths and stiffnesses of the major elements. The mass distribution includes lumped masses for end lids and impact limiters, as well as steel and copper densities for the thermal shield and the copper fins. The containment structural shell, 1.5 inches thick, contains its own mass, plus the mass of all internals, adjusted to make the total mass agree with the estimated weight of 203,000 lbs for the fully loaded BWR configuration. While the stiffness and mass are accurately represented for symmetrical loadings, the local stress distributions are only good

approximations. The final analysis will use a more refined model to more precisely calculate the local stresses. However, in these areas, sufficient margin of safety is shown to ensure the results are more than adequate for the preliminary design. The preliminary analysis indicates local thickening of the 1/2 inch plate may be required to reduce local stresses. Figure 2.5.2-4 shows the small local area of high stress. Based on the calculated stresses a local thickening of the 1/2 inch plate is indicated. A thickening of 1/4 inch, over an area of about 20 x 40 inches, on two sides at both ends, at a density of 0.283 lbs/inch³, would add approximately 226 lbs to the cask. The 1/4 inch plate thickening would lower local stresses to well within acceptable limits.

The approach taken for the preliminary analysis is that the combined vertical and longitudinal loads could be analyzed by the computer model. Side loads of 5 G's are applied to the cradle's longitudinal side beams, as well as the cradle's four lug attachment points. The side load will be distributed between the steel plates and the thin shell based on their relative rigidities. The rigidity of the thin shell is very small compared to the steel plates. It is assumed, for this preliminary analysis, that the steel plates will carry all the applied side load. Analysis of these plates shows that they have a margin of +1.76 on yield, and a margin of +0.66 on buckling. For the final design, the half model will be revised to more precisely model the high stressed areas and doubled to represent the full cask. This will allow a three dimensional load analysis without the simplifying assumptions of the preliminary analysis.

Using this approach, the margins of safety are found for the cask shell, lugs, and joint of thermal shield to end ring due to longitudinal and vertical loads.

The side loads are applied separately to the fin reinforcing steel plates to find structural margins, and potential additional stresses from the vertical and longitudinal loads are reviewed in Section 2.5.2.1.2.

2.5.2.1.1 Computer Model Results for Combined Longitudinal and Vertical Loads

The model is a three dimensional model of one-half of the cask, shown in Figure 2.5.2-1. The thermal shield is 1/8 inch thick, reinforced to a total thickness of 1/4 inch at each end around the lug pads, which taper from 1-1/8 inches to 1/2 inch thick. There is a 1/2 inch plate between the reinforced lug pad and the end ring forging. These different thicknesses are approximately modeled by elements of different thicknesses in the model. Figure 2.5.2-2 shows a side view of the deflection shape, under the g loads of 10 and 2 g's as noted on the figure. The displacements are greatly exaggerated for clarity. The computed maximum deflection is 0.130256 inches, labeled DMX on the figure. Most of this deflection is in the deflection of the springs representing the cradle. Two nodes are longitudinally restrained by grounded spring elements. The reaction forces of these springs on their ground are indicated by the two arrows at the right end of the cask. The force on the cask is opposite in direction to the direction of the arrows. Four vertical springs restrain the cask at four nodal points. The arrows for these reactions, being to scale, cannot be clearly seen in this figure. From the computer printout, Appendix 2.10.4, sheet 3, the reactions for one side of the cask at 10 g's longitudinal and 2 g's vertical are:

Lug location		Reaction
Vertical:	fwd lug, fwd corner	= 74,648 lbs
	fwd lug, aft corner	= 71,631
	aft lug, fwd corner	= 35,213
	aft lug, aft corner	= 21,838
	Total Vertical	= <u>203,329 lbs</u>
Longitudinal:		
	aft lug, upper corner	= 514,314
	aft lug, lower corner	= 502,596
	Total Longitudinal	= <u>1,016,910 lbs</u>

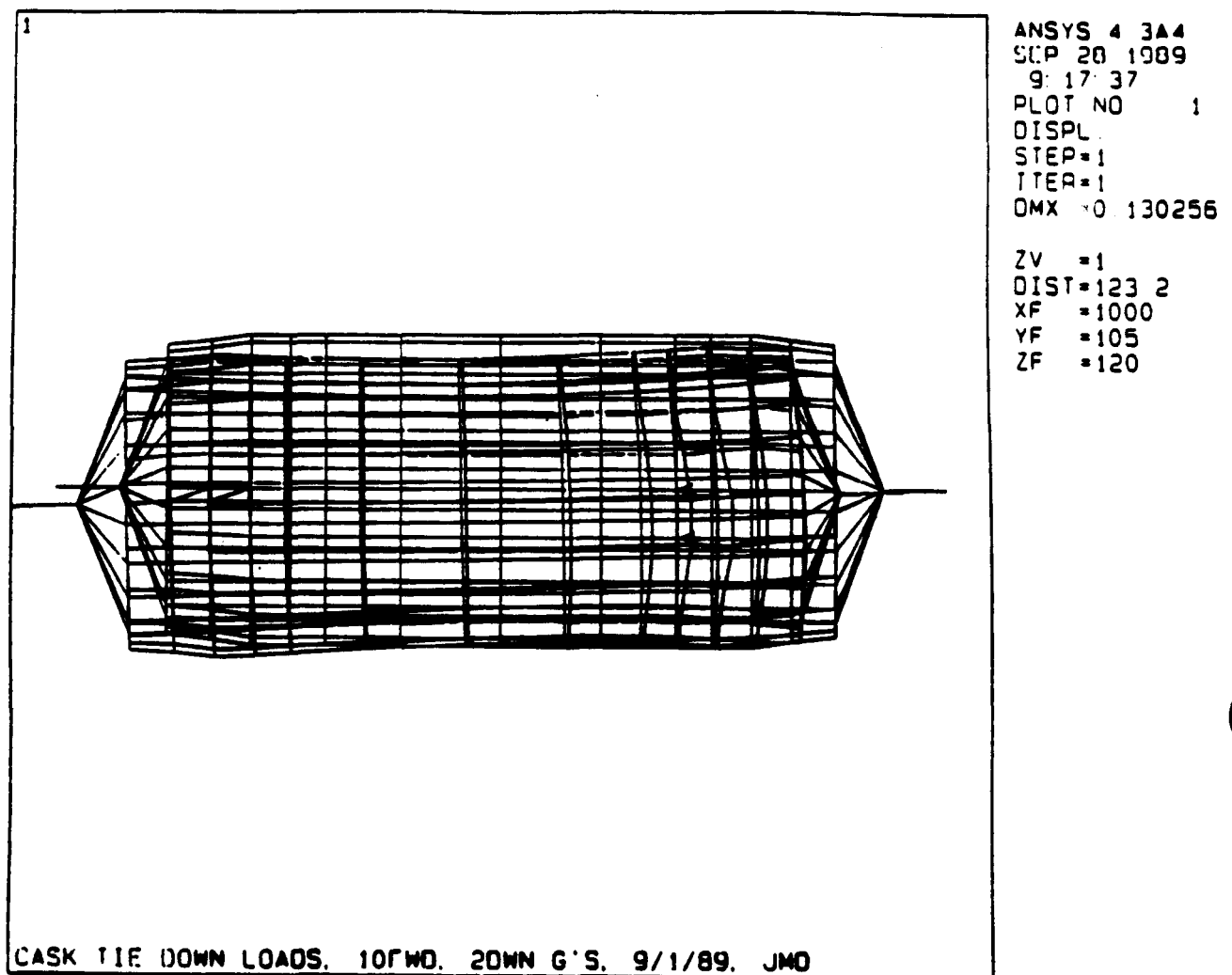


FIGURE 2.5.2-2
Cask Half Model, Original and Displaced Shape

The maximum stress intensity for the outer surface of the thermal shield is 46.31 ksi at the end of the shell in the model, which represents the transition from the thin flange on the ring forging to the thick section. Figure 2.5.2-3 breaks out the thermal shield for clarity, showing its displaced shapes. The arrows at the right end represent forces between the shell and the very stiff beam elements representing the end ring. Stress intensity contours of the thermal shield are shown in Figure 2.5.2-4. These values are for the mid surface stress of the shell elements. The analysis shows a maximum local stress in the heat shield of 46,019 psi. The copper fin arrangement is broken out of the model and displayed in Figure 2.5.2-5. The many arrows represent joint forces from elements which have been removed for this display. The highest stress shown is a stress intensity on the corner of the fin at mid height, which is reinforced. As will be shown in the following section, this stress could be added to the lateral load stresses in the steel plate without exceeding yield or buckling stresses. This model was not designed to investigate in detail the fin stresses. A more refined model would reduce stresses further. Figure 2.5.2-6 shows the stresses in the outer structural shell, which includes all the mass of the lead, basket etc. These stresses, though low, are very conservative and would be reduced by a more refined model.

As stated previously, a local thickening of the thermal shield will reduce the local high stresses to within allowable stress limits. The final thermal shield design will incorporate the local thickening.

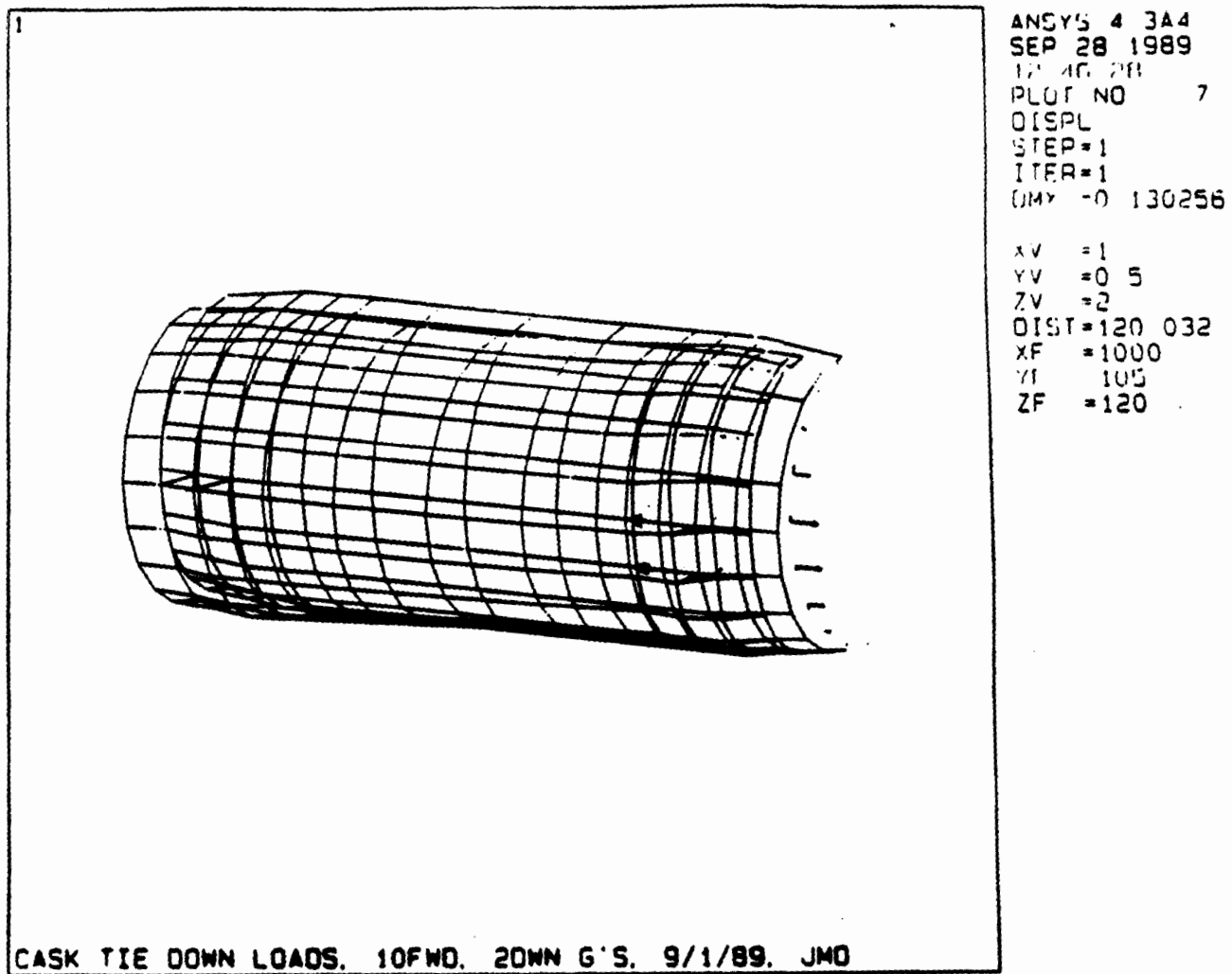


FIGURE 2.5.2-3
Original and Displaced Shape of the Thermal Shield

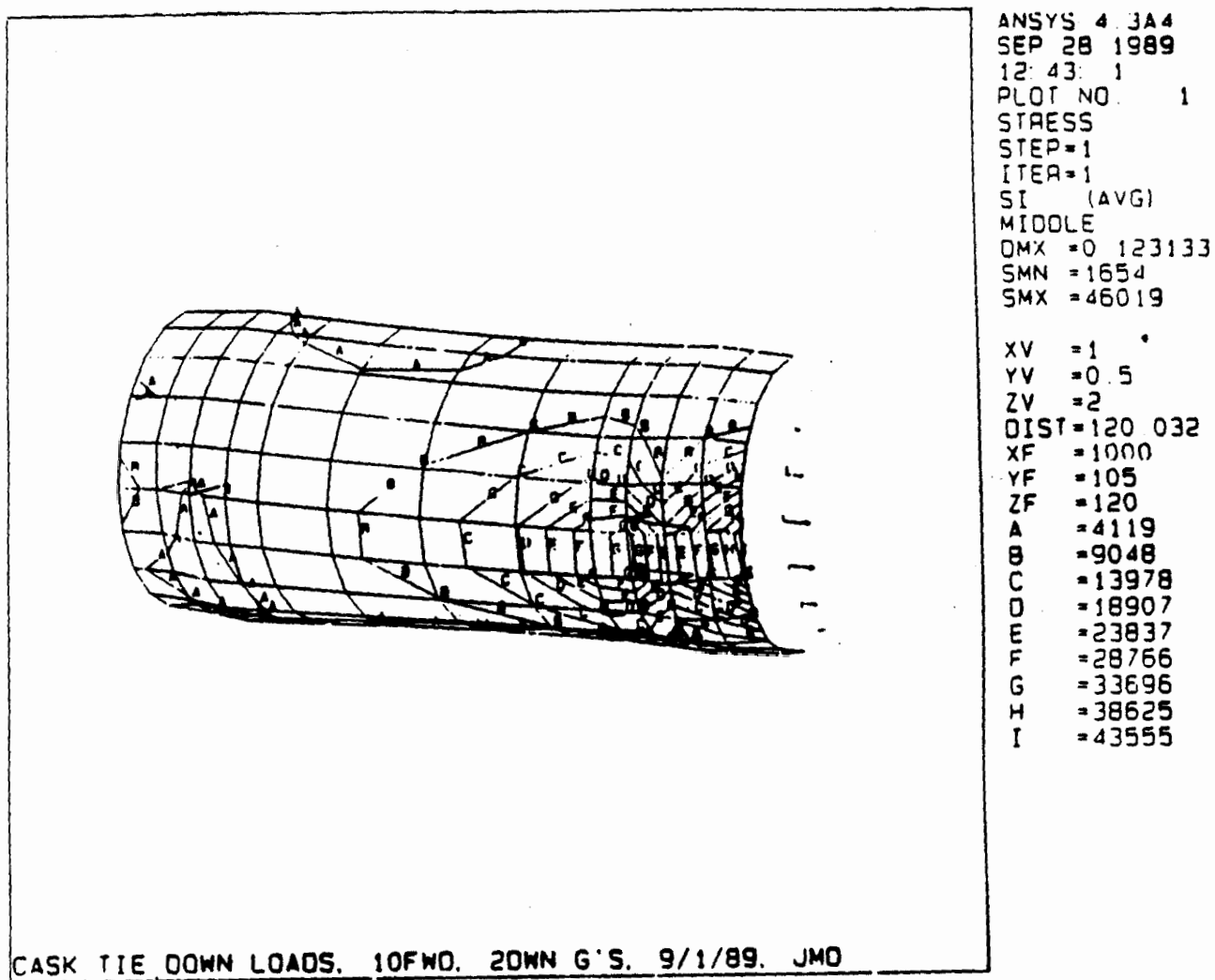


FIGURE 2.5.2-4
Stress Intensity for the Thermal Shield
with reinforced Lug Areas

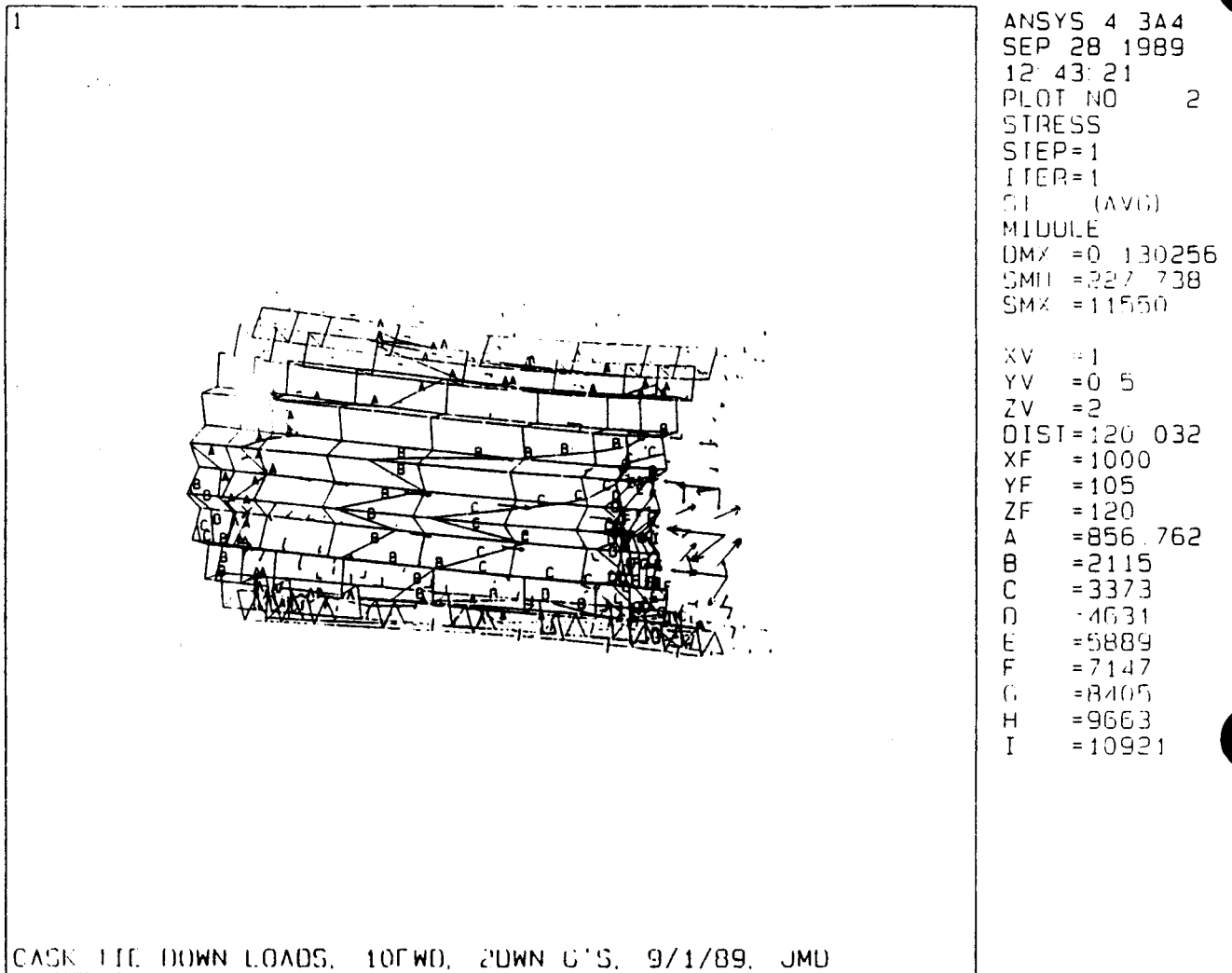


FIGURE 2.5.2-5

Stress Intensity on Copper and Reinforced fins

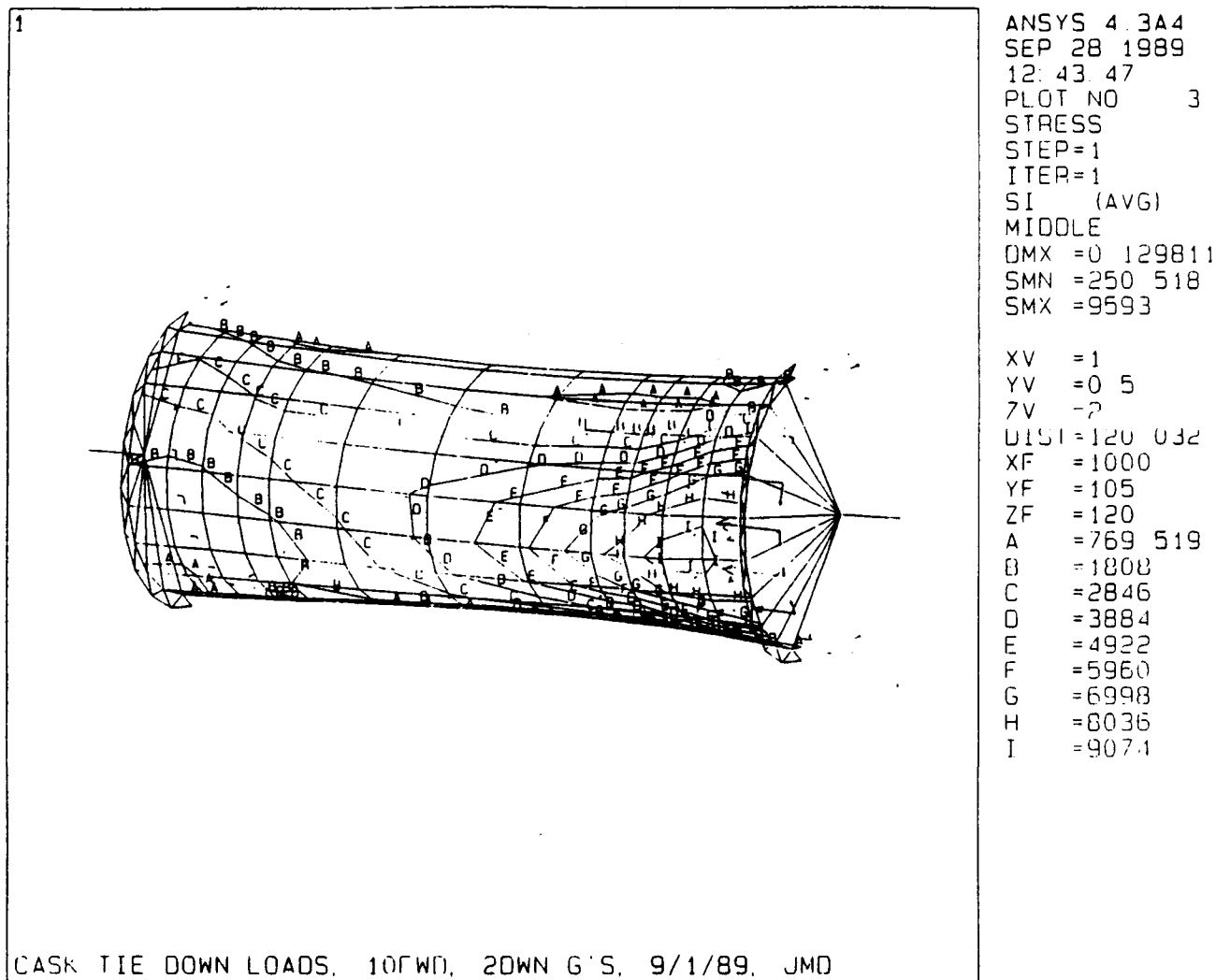


FIGURE 2.5.2-6
 Stress Intensity of Structural Shell

2.5.2.1.2 Results of Calculations for Tie Down Side Loads

As discussed above, the side load is assumed to be carried by the reinforced fins adjacent to the cradle interface. Stresses and margins are found with the entire 5 g side load carried only by the 3/8 inch steel reinforcing plates. The strength and stiffness of the copper fins are neglected, although at 300 °F the 1/2 inch copper has an allowable S_m of 5.0 ksi, and an elastic modulus of $16.3 (10^6)$ psi as presented in Table 2.3-1. The total side load (F_s) is calculated as:

$$\begin{aligned} F_s &= 203,000 (5) \\ &= 1,015,000 \text{ lbs} \end{aligned}$$

The distributed load (F_d) over a 140 inch length is:

$$\begin{aligned} F_d &= 1,015,000 / 140 \\ &= 7,250 \text{ lbs / inch} \end{aligned}$$

The resultant load in the steel plates is 5,900 lbs / inch, as shown in Figure 2.5.2-7. The shear load at the connection to the containment shell is 4,640 lbs / inch.

This margin neglects all support from the copper as well as from the 3/4 inch diameter studs.

The reinforcing plates themselves carry a load of 5,900 lbs / inch. The resulting stress (σ_s) is calculated as:

$$\sigma_s = F / A$$

Where,

$$\begin{aligned} F &= \text{Distributed plate load} \\ &= 5,900 \text{ lbs / inch} \end{aligned}$$

$$\begin{aligned} A &= \text{plate stress area} \\ &= 0.375 \text{ inches}^2 \end{aligned}$$

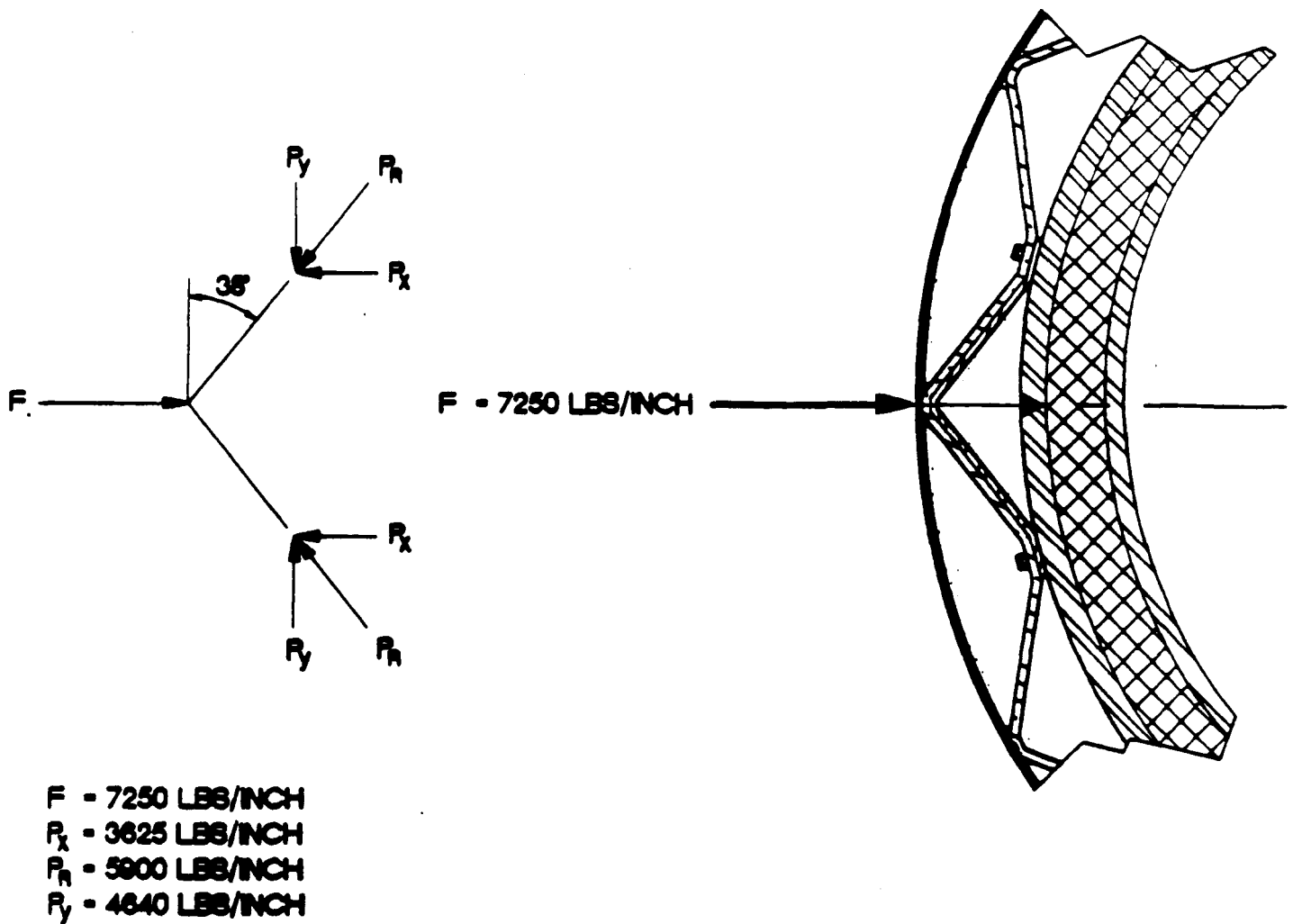


FIGURE 2.5.2-7
Side Loads on Fin Reinforcement Plates

Then,

$$\begin{aligned}\sigma_c &= 5,900/0.375 \\ &= 15,733 \text{ psi}\end{aligned}$$

To determine buckling margins, the critical (allowable) buckling stress is found. The V-shaped steel reinforcement can be considered as a pair of long, fixed edge plates. With a compressive load of 5,900 lbs / inch, and approximating each plate as a fixed end beam, the Euler equation with a safety factor of 2.0 gives a critical stress capability (σ_{cr}) of:

$$\begin{aligned}\sigma_{cr} &= (1/2) \{ 4 [(\pi^2) (E) (I)] / L^2 \} \\ &= (1/2) \{ 4 [(\pi^2) (28.3(10^6) (3.75^3)] / 12) / 9.8^2 \} \\ &= (1/2) (51.1 \text{ ksi}) \\ \sigma_{cr} &= 26,200 \text{ psi}\end{aligned}$$

The resulting buckling Margin of Safety (M.S.) is:

$$\begin{aligned}\text{M.S.} &= (\sigma_{cr}/\sigma_c) - 1 \\ &= (26,200/15,733) - 1 \\ &= + 0.67\end{aligned}$$

The shear load is carried by a 3/8 inch fillet weld. The shear stress (σ_v) in the weld is calculated as:

$$\sigma_v = F / A_v$$

Where,

$$\begin{aligned}F &= \text{distributed load} \\ &= 4,640 \text{ lbs/inch} \\ A_v &= \text{shear area of weld} \\ &= 0.375 (.707) \\ &= 0.265 \text{ inch}^2/\text{inch}\end{aligned}$$

Then,

$$\begin{aligned}\sigma_v &= 4,640/0.265 \\ &= 17,509 \text{ psi}\end{aligned}$$

The allowable shear stress (σ_s) on the weld is:

$$\begin{aligned}\sigma_s &= 0.6(43,400) \\ &= 26,040 \text{ psi}\end{aligned}$$

The Margin of Safety (M.S.) is therefore:

$$\begin{aligned}\text{M.S.} &= (\sigma_s / \sigma_v) - 1 \\ &= (26,040/17,509) - 1 \\ &= + 0.49\end{aligned}$$

2.5.2.2 Vertical Transportation Loads in the Tie Down Structure

The same model used for combined longitudinal and vertical tie down loads is also used for vertical transportation loads. The transportation load stresses are used in Section 2.6.5, where margins of safety during vibration are presented for all the major cask components.

Vertical loads of 1.4 g's were applied to the cask half model to provide an estimate of transportation stresses for long term vibration. The vibration loading is taken from the Reference (43) report, prepared by the Association of American Railroads for EG&G Idaho, to identify shock and vibration environments of rail transport for a 100 ton load. This report indicates that 2.0 g's would be the maximum shock loading, and that less than 0.75 g's could be expected in continuous vibration 99% of the time with peaking to 1.0 g's less than 1% of the time. Thus, the 1.4 g's applied above is conservative and accounts for any unknown potential amplification that may occur due to the specialized railcar planned for the NuPac 140-B cask.

Stresses for the vertical 1.4 g load are summarized on the stress intensity plots of Figures 2.5.2-8 through 2.5.2-11. The maximum stress intensities for the tie down structural components are:

Component	Stress Intensity (psi)
Tie down lug	3,476
Thermal shield to end ring weld joint	5,304
Thermal shield	1,950
Fin reinforcing plates	1,311
Outer structural containment shell	675

Computer print out for the vertical transportation loads are in Appendix 2.10.4.3.2.

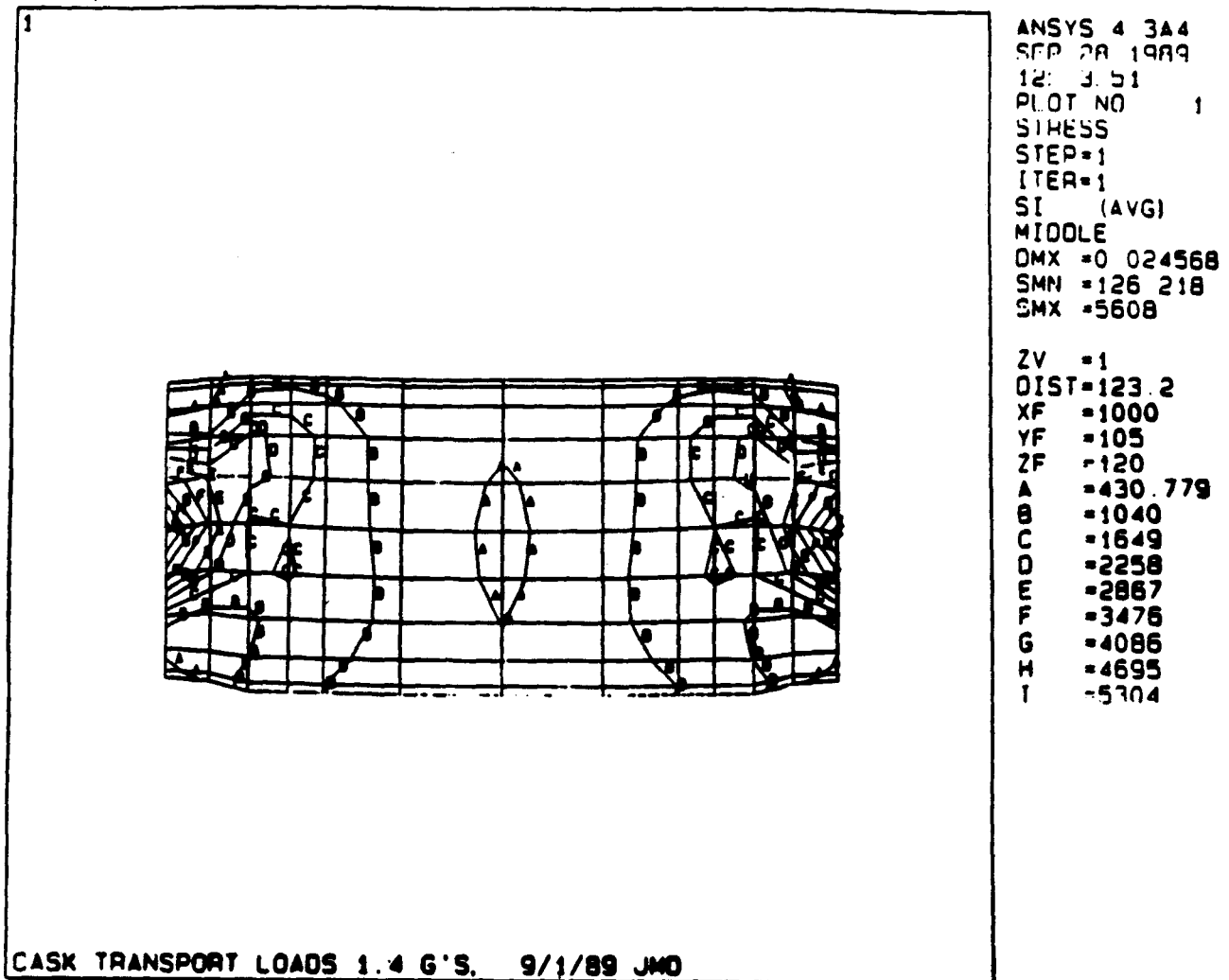


FIGURE 2.5.2-8
Thermal Shield Stress Intensity

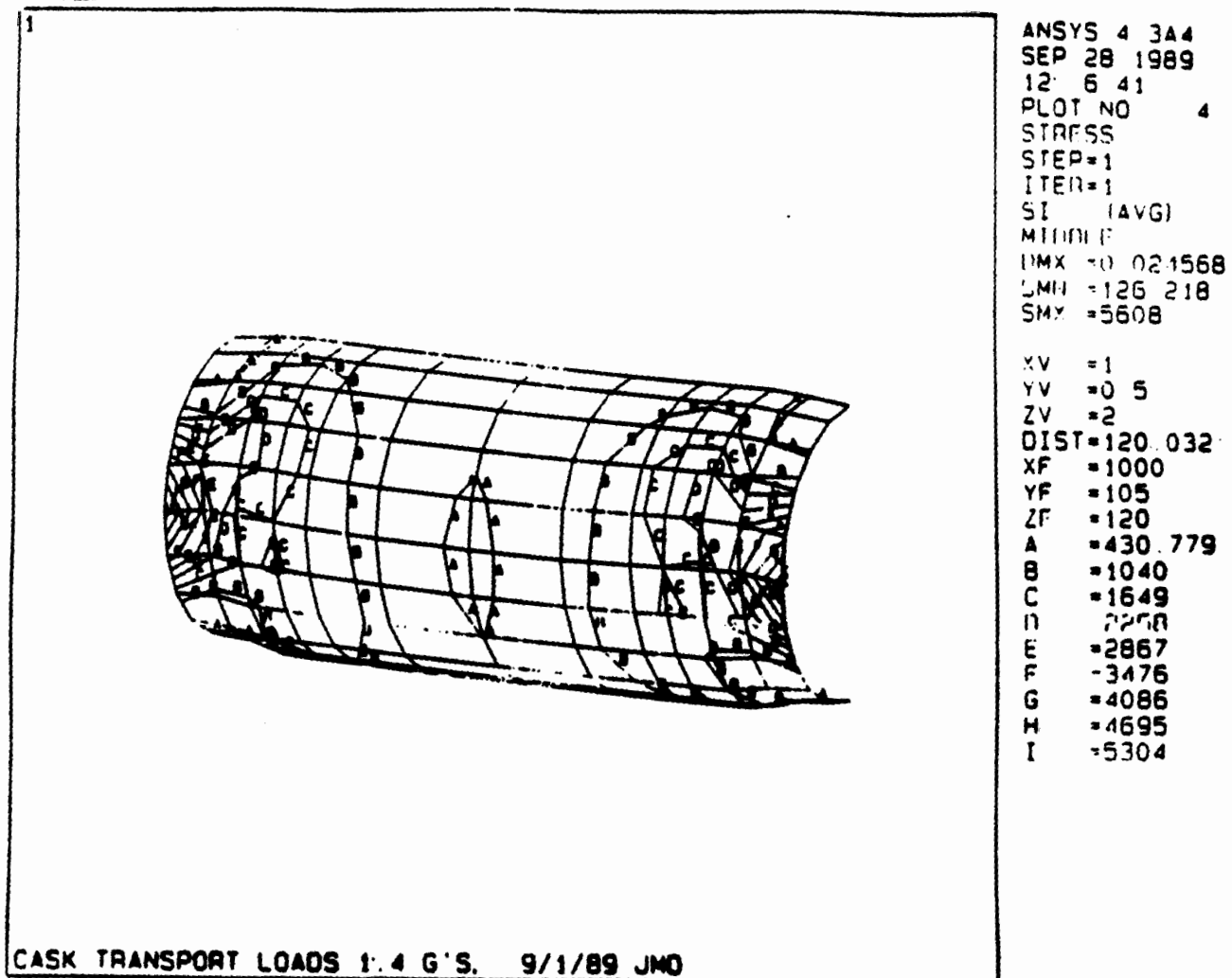


FIGURE 2.5.2-9
Rotated View of Stress Intensity in Outer Shell

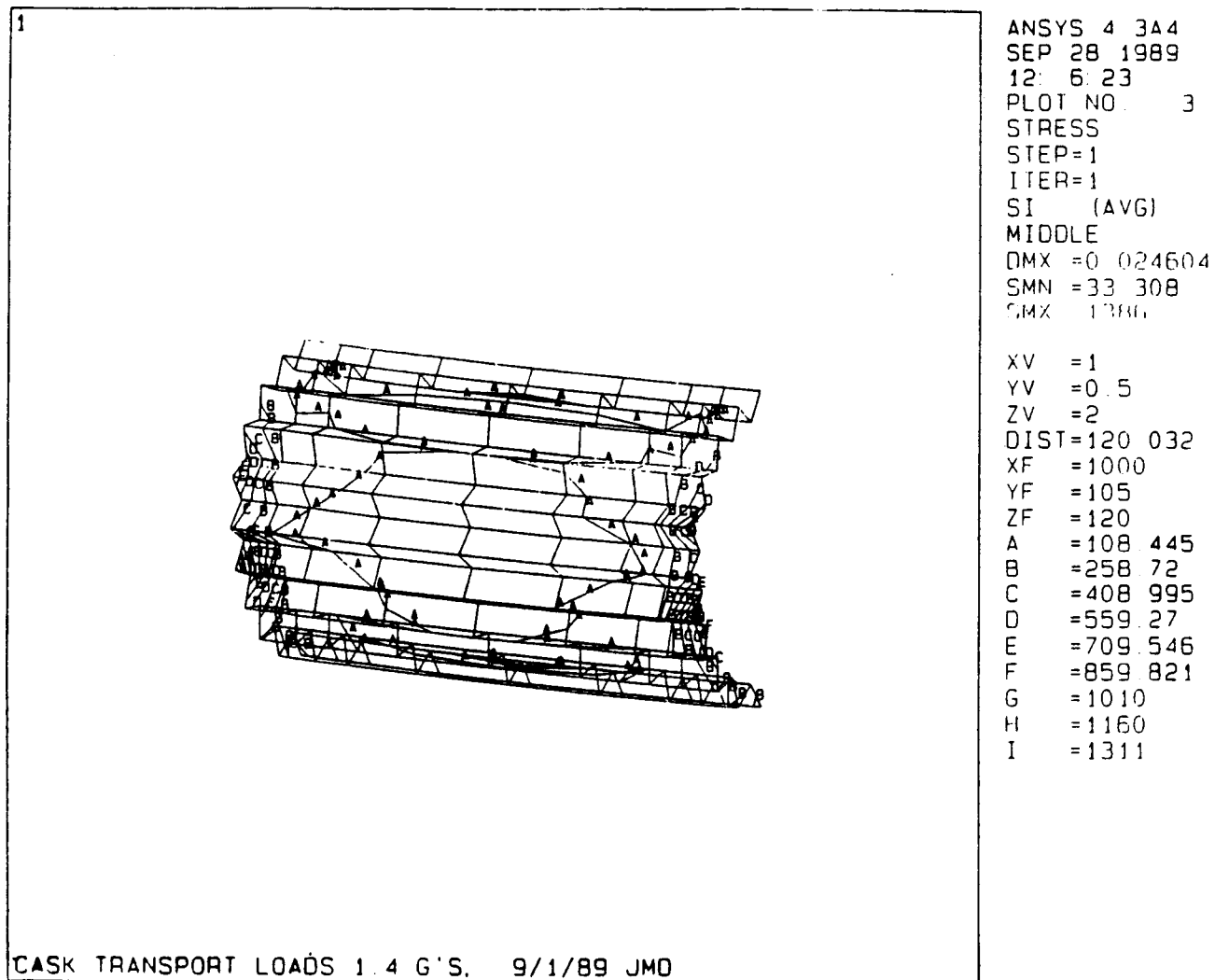


FIGURE 2.5.2-10
Stress Intensity of Copper Fins

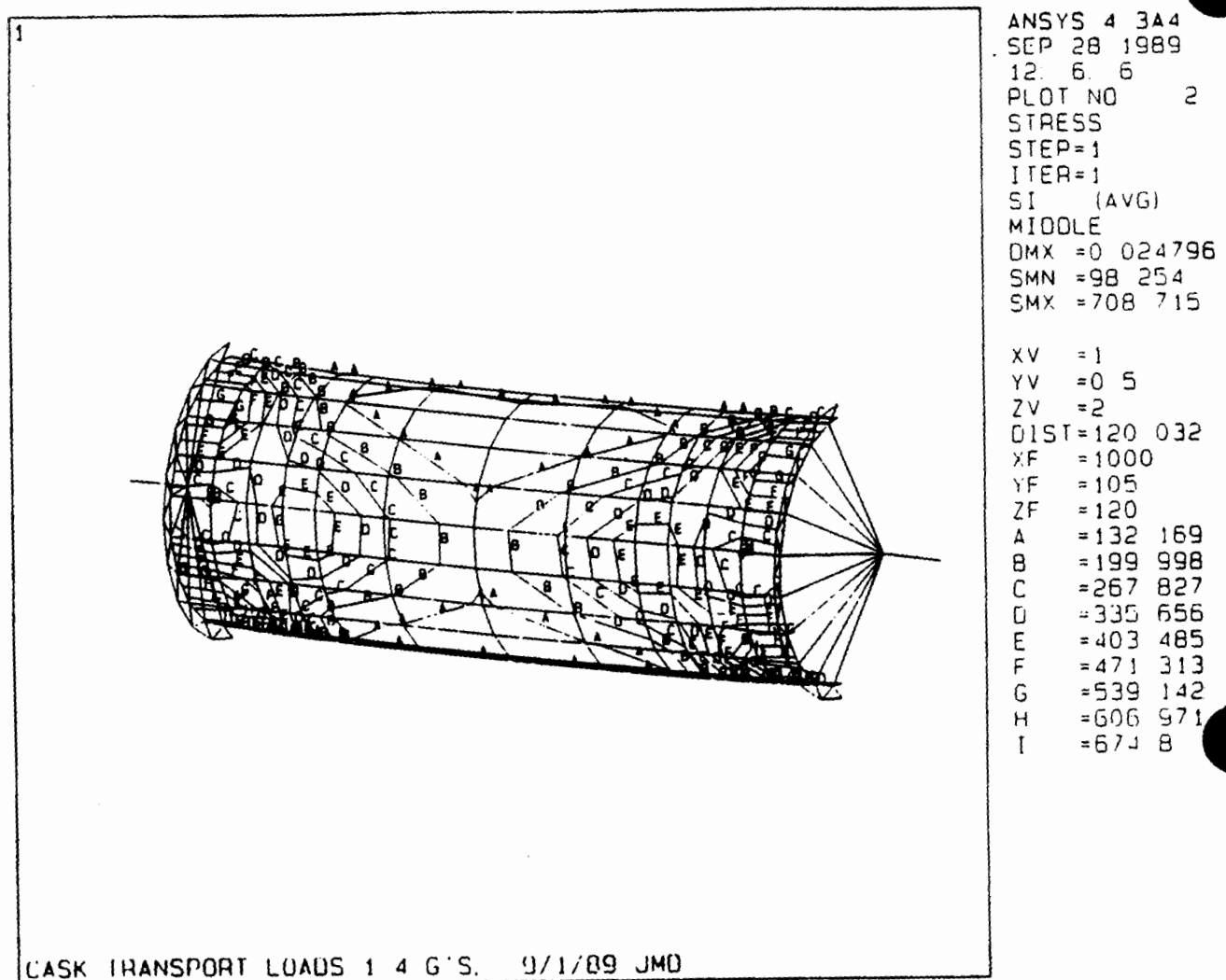


FIGURE 2.5.2-11
Stress Intensity of Outer Shell

2.6 Normal Conditions of Transport

The NuPac 140-B Cask, when subjected to normal conditions of transport as specified in 10 CFR 71.71, meets the performance requirements specified in Subpart E of 10 CFR 71. This is demonstrated in the following subsections where each normal condition is addressed and shown to meet the applicable design criteria (except buckling) previously discussed in Section 2.1.2. Buckling evaluations for all buckling sensitive cask components are presented in Appendix 2.10.3. Detailed structural evaluation of the payload (PWR and BWR) baskets is included as Section 2.6.8.

2.6.1 Heat

The thermal evaluation for the normal heat condition is presented in Section 3.4. The normal heat condition consists of exposing the cask to direct sunlight and 100 °F still air per the requirements of 10 CFR 71.71(c)(1), with insolation. A maximum internal heat load of 11,000 watts (52 BWR or 21 PWR fuel assemblies) is used for the evaluation.

2.6.1.1 Summary of Pressures and Temperatures

The conditions of normal heat result in rather modest temperatures throughout the cask. Maximum temperatures for the various cask components are presented in Table 2.6.1-1. As indicated, most cask structural component temperatures remain below 300 °F. Thus, for all of the cask structural analyses herein, a 300 °F maximum temperature is employed. Basket temperatures are higher and all structural analyses of the baskets is presented in Section 2.6.8.

Experience from Reference (27) for a similar cask shows that the primary regions of concern for thermal stress are at the cask cylindrical shell to end closure interfaces. For the NuPac 140-B Cask the load collar also terminates at this interface. In addition, lid bolt stress and the copper fin to outer cask wall attachment interface are also of concern. Thus, these cask regions are investigated in detail for thermal stress in Section 3.4.5.

Differential thermal expansion between the cask and the baskets is also considered briefly in Section 3.4.5. Differential expansion between the borated silicone neutron shield and its surrounding walls is considered in detail in Section 3.4.5.

It is noted that fabrication induced stresses due to lead shrinkage on the inner shell of the outer cask will be relieved under the normal heat condition which results in lead and steel shell temperatures very near 300°F.

TABLE 2.6.1-1
Normal Conditions of Transport Temperatures (°F)

<u>Component</u>	<u>Maximum</u>	<u>Minimum</u>
Fuel rods	660	440
BWR baskets	594	403
PWR baskets	580	356
Fuel assy spacers	387	312
Cask Inner Shell	298	288
Cask Lead Shell	296	283
Cask Outer Shell	294	280
Cask Lid	333	294
Cask O-ring Seals	300	295
Cask End Closure Forgings	296	273
Copper Fins	293	271
Borated Silicone Neutron Shield	293	271
Load Collar	290	245
Thermal Shield	246	241
Impact Limiter	278	110

Initial pressure in the NuPac 140-B Cask is one (1) atmosphere (14.7 psia). Per Section 3.4.4, the maximum normal operating pressure (MNOF) is 284 psig for any package component.

2.6.1.2 Thermal Stress Due to Differential Thermal Expansion

Section 3.4.5 presents detailed differential thermal expansion and resulting thermal stress calculations for the cask shells, copper fins, and lid closure bolts. Differential thermal expansion between the cask and baskets is also considered. Thermal stress induced by the borated silicone neutron shield material is negated by design with adequate tolerances to allow for the corresponding growth with temperature.

Table 2.6.1-2 summarizes the results of the thermal stress calculations for the cask shells.

TABLE 2.6.1-2
Thermal Stress Summary for the Cask Shells and Load Collar (psi)

Cask Component	Wall Thickness (in)	<u>Meridional (Axial)</u>		<u>Circumferential (Hoop)</u>	
		Membrane (psi)	Bending (psi)	Membrane (psi)	Bending (psi)
Inner Shell	1.13	-234	-1,445	-704	-433
Outer Shell	1.50	234	-2,109	-1,164	-633
Load Collar	0.25	4,671	6,384	3,524	1,915

Since these thermal stresses are deformation controlled, they are classified as secondary stresses. The normal condition allowable for the range of primary plus secondary stress intensity, S_s , is $(3.0)S_u$, or 94,200 psi at 300 °F for the Type XM-19 stainless steel (see Table 2.3-1) used within the NuPac 140-B Cask. Conservatively combining the highest normal condition secondary stress intensity determined above (6,384 psi) with the highest normal condition primary membrane plus bending stress intensity anywhere in the cask, excluding bolts (15,473 psi at center of outer cask lid for internal pressure plus end drop loading, per Table 2.6.7.1-1 of Section 2.6.7.1) results in a primary plus secondary stress intensity of 21,857 psi. By conservatively assuming fully reversing stress states, the range of primary plus secondary stress intensity becomes:

$$SI_{\text{range}} = 2(21,857) = 43,714 \text{ psi}$$

The corresponding Margin of Safety is:

$$\begin{aligned} \text{M.S.} &= (S_u/SI_{\text{range}}) - 1 \\ &= (94,200/43,714) - 1 = \underline{+1.15} \end{aligned}$$

Per the above discussion, the remainder of Section 2.6 does not specifically address thermal stresses when load combinations are made. In other words, the design is governed by primary, load controlled stresses and the corresponding limits specified in Section C.4 of Regulatory Guide 7.6 rather than by primary plus secondary stresses and the $(3.0)S_u$ limit specified in Section C.4. However, bucking evaluations presented in Appendix 2.10.3 do consider secondary thermal stresses.

The thermal stress calculation in Section 3.4.5 for the lid closure bolts resulted in 11,484 psi for 300 °F. By itself, this value of thermal stress is insignificant since the normal condition allowable from Table 2.1-1 is $(2.0)S_u$, or 62,800 psi at 300 °F for the Type XM-19 stainless steel (see Table 2.3-1) used within the NuPac 140-B Cask. The real significance of the lid bolt thermal stress is its inclusion in the closure bolt stress evaluations of Appendix 2.10.5. These evaluations show positive margins for the lid closure bolts.

The thermal stress calculation in Section 3.4.5 for the copper fins resulted in maximum component stresses of 2,560 psi compression and 621 psi bending, and 2,362 psi compression locally at the bolted attachment interface. With a yield strength in excess of 5,000 psi at 300 °F, the copper fins will stay well within their elastic limit. Thus, the heat dissipation function of the copper fins will not be degraded due to loosening of the stud-welded attachment bolts.

As mentioned above, there will not be any stress induced in the surrounding cask structure due to thermal expansion of the borated silicone neutron shield segments or discs. As described in Section 3.4.5, all borated silicone neutron shield components are designed with adequate tolerances to accommodate normal thermal expansion to temperatures of 300 °F.

2.6.1.3 Stress Calculations for Pressure Induced Stresses

This section presents calculations of pressure induced stresses on the various cask components. Unit pressure (1 psi) is first considered, and appropriate factors are then applied to arrive at stresses for particular loading cases of interest. Locations of interest for the NuPac 140-B Cask are shown in Figure 2.6.1-1.

The only significant mechanical loading on the NuPac 140-B Cask is dead weight. As free drop cases impose higher mechanical loadings on the cask than does dead weight alone, combinations of pressure loads and mechanical loadings due to free drop will govern the design and are presented in Section 2.6.7. Fatigue is not a concern as discussed in Section 2.1.2.2.2.

2.6.1.3.1 Stresses Due to Unit Pressures

Referring to Figure 2.6.1-1, geometries and temperatures for the various locations (P1 through P6) are as follows:

<u>Location</u>	<u>Temperature</u> <u>(°F)</u>	<u>Cylinders</u>		<u>End Plates</u>	
		<u>Radius</u> <u>(in)</u>	<u>Thickness</u> <u>(in)</u>	<u>Diameter</u> <u>(in)</u>	<u>Thickness</u> <u>(in)</u>
P1	300	33.875	1.50	---	---
P2	300	29.063	1.13	---	---
P3	300	29.063	1.13	---	---
P4	300	33.875	1.50	---	---
P5	300	---	---	77.40	8.50
P6	300	---	---	77.40	8.50

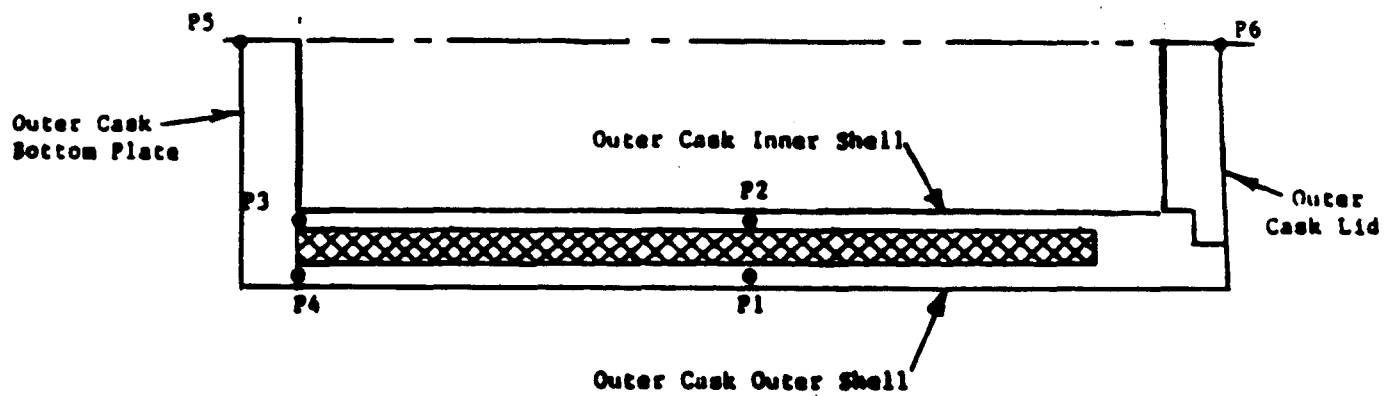


FIGURE 2.6.1-1
Cask Locations for Pressure Stress Calculations

For locations P1 and P2 (away from the cylinder ends),

$$\sigma_1 = pr/2t \quad (\text{axial membrane})$$

$$\sigma_2 = pr/t \quad (\text{hoop membrane})$$

$$\sigma_1' = \sigma_2' = 0 \quad (\text{axial and hoop bending, respectively})$$

For locations P4 and P5 (at the cask cylinder ends), stresses are determined from Table 29, Case 1c, and Table 30, Cases 8 and 10, of Reference (15):

$$\sigma_1 = pr/2t \quad (\text{meridional membrane})$$

$$\sigma_2 = pr/t - 2V_o\lambda r/t + 2M_o\lambda^2 r/t \quad (\text{hoop membrane})$$

$$\sigma_1' = -6M_o/t^2 \quad (\text{meridional bending})$$

$$\sigma_2' = -\mu 6M_o/t^2 \quad (\text{hoop bending})$$

$$\lambda = [3(1 - \mu^2)/r^2t^2]^{0.25}$$

$$D = Et^3/12(1 - \mu^2)$$

$$\delta = 0 = (pr^2/Et)(1 - \mu/2) - V_o/2D\lambda^3 + M_o/2D\lambda^2 \quad (\text{radial displacement})$$

$$\theta = 0 = V_o/2D\lambda^2 - M_o/D\lambda \quad (\text{meridional slope})$$

$$E = 27.0(10)^6 \text{ psi at } 300^\circ \text{ F}$$

$$\mu = 0.3$$

$$p = 1.0 \text{ psi (unit internal pressure)}$$

The following table summarizes the results for locations P1 through P4:

	Temp	λ	D	V_x	M_x	σ_x	σ_y	σ_x'	σ_y'
<u>Location</u>	<u>(°F)</u>	<u>(in⁻¹)</u>	<u>(lbs-in)</u>	<u>(lbs/in)</u>	<u>(in-lbs/in)</u>	<u>(psi)</u>	<u>(psi)</u>	<u>(psi)</u>	<u>(psi)</u>
P1	300	---	---	---	---	11.29	22.58	0.0	0.0
P2	300	---	---	---	---	12.86	25.72	0.0	0.0
P3	300	0.224	3.568(10) ⁴	3.808	8.532	12.86	3.86	-40.09	-12.03
P4	300	0.180	8.345(10) ⁴	5.540	15.350	11.29	0.0	-46.16	-13.85

Locations P5 and P6 are considered utilizing Table 24, Case 10a, of Reference (15). Assuming a simply supported plate with a uniform loading, the bending moment and stress are:

$$M_o = pa^2(3 + \mu)/16$$

$$\sigma = 6M_o/t^2 = 3pa^2(3 + \mu)/8t^2$$

Where:

a = plate radius

$\mu = 0.3$

t = plate thickness

p = 1.0 psi (unit internal pressure)

The following table summarizes the results for locations P5 and P6:

	Temperature	M_o	σ
<u>Location</u>	<u>(°F)</u>	<u>(in-lbs/in)</u>	<u>(psi)</u>
P5	300	308.9	25.65
P6	300	308.9	25.65

2.6.1.3.2 Stresses Due to Maximum Pressures

The maximum internal pressure acting on any component in the NuPac 140-B Cask is 284 psig. Utilizing the unit stresses developed in Section 2.6.1.3.1, the following maximum pressure stresses result:

	Meridional	Hoop	Meridional	Hoop
	Membrane	Membrane	Bending'	Bending'
<u>Location</u>	<u>σ_1, psi</u>	<u>σ_2, psi</u>	<u>σ_1', psi</u>	<u>σ_2', psi</u>
P1	3,206	6,413	0	0
P2	3,652	7,304	0	0
P3	3,652	1,096	-11,386	-3,416
P4	3,206	0	-13,109	-3,933

* σ_1' and σ_2' are positive when tensile on the shell outside diameter.

	Radial
End Plate	Bending
<u>Location</u>	<u>σ, psi</u>
P5	7,285
P6	7,285

2.6.1.4 Comparison with Allowable Stresses

As discussed in Section 2.6.1.3, load combinations and comparisons with allowable stress limits are presented in Section 2.6.7 where worst case combined load cases are addressed.

2.6.2 Cold

For the cold condition, a -40 °F steady state ambient temperature and no internal heat generation are assumed. This will result in a uniform temperature throughout the cask of -40 °F. The materials of construction for the cask are not adversely affected by the -40 °F condition. In particular, brittle fracture is not a concern, as discussed in Section 2.1.2.2.1.

The only concern identified with the cold condition is with shrinkage of the lead onto the inner shell of the cask. As shown by the following calculations, a hoop stress of -3,230 psi and an axial stress of -3,344 psi can develop in the inner shell when cooled to -40 °F. This case is independent of other load cases and will not, therefore, limit the cask design. However, a -20 °F case must be considered as a possible initial condition for other load cases per 10 CFR 71.71(b). The hoop stress will be approximately -3,099 psi and the axial stress approximately -3,174 psi at -20 °F. These stresses are determined by conservatively neglecting the beneficial effects of lead creep.

An additional discussion of fabrication induced stresses due to lead shrinkage onto the outer cask inner shell is contained in Section 2.6.7.1. The analysis therein assesses the consequences of a flat end drop of the package. In order to arrive at appropriate initial conditions for the drop, the process of lead cool-down is initially followed.

Fabrication Stresses Due to Lead Pour:

Assume a uniform steel and lead temperature of 620 °F. The static head (pressure) due to a column of lead is simply:

$$p = \rho h$$

Where:

$$\rho = 0.386 \text{ lbs/in}^3 \text{ (liquid lead)}$$

$$h = 177.0 + 22.0 = 199.0 \text{ in (22.0 inches for overflow pipe)}$$

Then,

$$p = (0.386)(199.0) = 77 \text{ psi}$$

The physical properties of ASTM A-240, Type XM-19, stainless steel used for the cask inner and outer shells are extracted from Table 2.3-1, as:

Temperature (°F)	E (10 ⁶ psi)	α (10 ⁻⁶ in/in/°F)	μ
70	28.3	8.24	0.3
100	28.1	8.30	0.3
200	27.6	8.48	0.3
300	27.0	8.65	0.3
400	26.5	8.79	0.3
500	25.8	8.92	0.3
600	25.3	9.03	0.3
620	25.2	9.05	0.3

Where:

E = Young's (Elastic) Modulus

α = coefficient of thermal expansion (mean from 70 °F)

μ = Poisson's Ratio

The physical properties of lead (copperized) are also taken from Section 2.3, as follows:

Temperature	E	α	μ
(°F)	(10^6 psi)	(10^{-6} in/in/°F)	
70	2.34	16.07	0.45
100	2.30	16.22	0.45
200	2.17	16.70	0.45
300	2.00	17.33	0.45
400	1.82	18.16	0.45
500	1.61	19.12	0.45
600	1.40	20.17	0.45
620	1.36	20.38	0.45

At 70 °F, the steel shell geometry is as follows:

Geometry	Inner Shell	Outer Shell
(at 70 °F)	(in)	(in)
Inside Diameter, D_i	57.000	66.250
Outside Diameter, D_o	59.250	69.250
Shell Thickness, t	1.125	1.500
Mean Shell Radius, R	29.063	33.875

At 620 °F, without lead, the shells will grow as follows:

$$R' = R(1 + \alpha\Delta T)$$

$$t' = t(1 + \alpha\Delta T)$$

$$\Delta T = 620 - 70 = 550 \text{ °F}$$

$$\alpha = 9.05(10)^{-6} \text{ in/in/°F}$$

Then,

$$R_1' = 29.063[1 + 9.05(10)^{-6}(550)] = 29.208 \text{ in}$$

$$t_1' = 1.125[1 + 9.05(10)^{-6}(550)] = 1.1306 \text{ in}$$

$$R_o' = 33.875[1 + 9.05(10)^{-6}(550)] = 34.044 \text{ in}$$

$$t_o' = 1.50[1 + 9.05(10)^{-6}(550)] = 1.5075 \text{ in}$$

When filled with lead, the inner and outer shells of the cask will be subjected to the 77 psi pressure head. This will decrease the radius of the inner shell and increase the radius of outer shell. Utilizing Table 29, Case 1b, of Reference (15), the change in radius, ΔR , for each shell is:

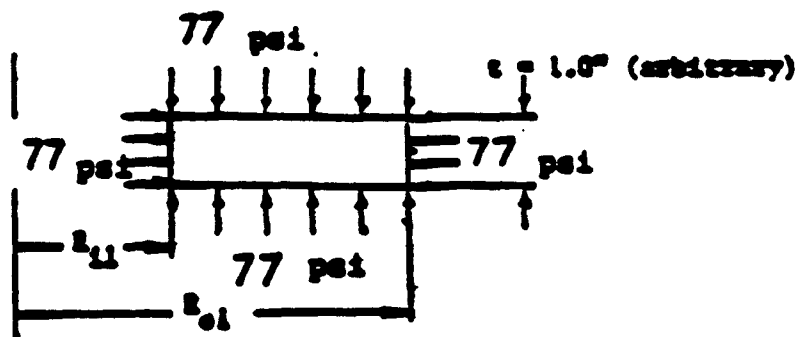
$$\begin{aligned}\Delta R_i' &= q(R_i')^2/Et_i' \\ &= (-77)(29.208)^2/25.2(10)^6(1.1306) = -0.00231 \text{ in}\end{aligned}$$

$$\begin{aligned}\Delta R_o' &= q(R_o')^2/Et_o' \\ &= (77)(34.044)^2/25.2(10)^6(1.5075) = 0.00235 \text{ in}\end{aligned}$$

In summary, the initial condition of the steel shells just before lead solidification at 620 °F, $(R' + \Delta' \pm t'/2)$ is:

	Inner Radius	Outer Radius
<u>Shell</u>	<u>(in)</u>	<u>(in)</u>
Inner	28.640	29.771
Outer	33.293	34.800

Lead experiences a decrease in volume of approximately 3.85% upon solidification. As the lead solidifies, it will shrink and liquid lead from above will fill in between the solidifying lead and the cask inner and outer shells, thus maintaining a 77 psi pressure on the shells. Eventually, the full annular region between the stainless shells will be filled with lead, subjected to a loading as illustrated below:



Note: The 77 psi pressure actually only exists at the base of the lead column and will linearly decrease to 12 psi at the top of the column.

Under this loading, $R_{i1} = 29.771$ inches and $R_{o1} = 33.293$ inches, the outer radius of the inner shell and inner radius of the outer shell, respectively. Geometry of the unloaded lead shell is determined as follows:

a = outer radius

b = inner radius

L = thickness

$$R_{o1} = a + \Delta a$$

$$R_{i1} = b + \Delta b$$

$$t = L + \Delta L$$

Solve for a , b , and L by superimposing Cases 1b and 1d, Table 32, of Reference (15), with $q = 77$ psi:

$$\begin{aligned} \Delta a &= qab^2(2 - \mu)/E(a^2 - b^2) - qa[a^2(1 - 2\mu) + b^2(1 + \mu)]/E(a^2 - b^2) \\ &= R_{o1} - a \end{aligned}$$

$$\begin{aligned} \Delta b &= qb[a^2(1 + \mu) + b^2(1 - 2\mu)]/E(a^2 - b^2) - qba^2(2 - \mu)/E(a^2 - b^2) \\ &= R_{i1} - b \end{aligned}$$

$$\begin{aligned} \Delta L &= qLb^2(1 - 2\mu)/E(a^2 - b^2) - qLa^2(1 - 2\mu)/E(a^2 - b^2) \\ &= t - L \end{aligned}$$

This results in three equations with three unknowns to be solved:

$$\begin{aligned} [1] \quad & 8.78(10)^{-5}ab^2/(a^2 - b^2) - 5.6618(10)^{-5}a[(0.1)a^2 + (1.45)b^2]/(a^2 - b^2) \\ &= 33.293 - a \end{aligned}$$

$$\begin{aligned} [2] \quad & 5.6618(10)^{-5}b[(1.45)a^2 + (0.1)b^2]/(a^2 - b^2) - 8.78(10)^{-5}ba^2/(a^2 - b^2) \\ &= 29.771 - b \end{aligned}$$

$$\begin{aligned} [3] \quad & 5.6618(10)^{-6} L b^2 / (a^2 - b^2) - 5.6618(10)^{-6} L a^2 / (a^2 - b^2) \\ & = 1.125 - L = [-5.6618(10)^{-6}] L \end{aligned}$$

Where:

$$E = 1.36(10)^6 \text{ psi}$$

$$\mu = 0.45$$

$$R_{o1} = 33.293 \text{ in}$$

$$R_{i1} = 29.771 \text{ in}$$

$$t = 1.125 \text{ in}$$

Then, solving for the unknowns:

$$L = 1.1250064 \text{ in, or } \Delta L = 6.4(10)^{-4} \text{ in}$$

$$a = 33.293132 \text{ in, or } \Delta a = 1.32(10)^{-4} \text{ in}$$

$$b = 29.771119 \text{ in, or } \Delta b = 1.19(10)^{-4} \text{ in}$$

These displacements are negligible, and will be ignored. At this point, the hoop stress in the inner and outer steel shells of the outer cask is:

$$\begin{aligned} \sigma_i &= pR_i/t_i \\ &= (-77) [(29.771 + 28.640)/2] / (29.771 - 28.640) = -1,988 \text{ psi} \end{aligned}$$

$$\begin{aligned} \sigma_o &= pR_o/t_o \\ &= (77) [(34.800 + 33.293)/2] / (34.800 - 33.293) = 1,740 \text{ psi} \end{aligned}$$

Again, utilizing Table 32, Cases 1b and 1d, of Reference (15), the stress in the lead shell, σ_1 = axial, σ_2 = hoop, σ_3 = radial, upon solidification, may be determined as:

$$\sigma_1 = \sigma_2 = \sigma_3 = -77 \text{ psi}$$

In cooling from lead solidification at 620 °F down to a nominal ambient temperature of 70 °F, the lead shell, with its greater coefficient of thermal expansion/contraction, will tend to shrink onto the inner steel shell. If there were no interference, the cooling lead shell would assume the following geometry:

$$R_q' = R_q (1 + \alpha \Delta T)$$

$$t_i' = t_i (1 + \alpha \Delta T)$$

$$R_q = (R_{q0} + R_{q1})/2 = (33.293 + 29.771)/2 = 31.532 \text{ in}$$

$$t_i = 33.293 - 29.771 = 3.522$$

$$\Delta T = 70 - 620 = -550 \text{ °F}$$

$$\alpha = 20.38 (10)^{-6} \text{ in/in/°F}$$

$$R_q' = 31.532 [1 + 20.38 (10)^{-6} (-550)] = 31.179$$

$$t_i' = 3.522 [1 + 20.38 (10)^{-6} (-550)] = 3.483$$

$$R_{q0} = R_q' + t_i'/2 = 32.921 \text{ in}$$

$$R_{q1} = R_q' - t_i'/2 = 29.438 \text{ in}$$

Stress-free dimensions of the cask inner shell have been previously determined.

<u>Item</u>	<u>620 °F</u>	<u>70 °F</u>	<u>ΔR</u>
Steel inner shell inner radius	28.640	28.500	0.140
Steel inner shell outer radius	29.771	29.625	0.146
Lead shell inner radius	29.771	29.438	0.333
Lead shell outer radius	33.293	32.921	0.372

The interference between the lead and the steel inner shell is thus the difference between the unstressed lead inner radius and the unstressed inner shell outer radius at 70 °F.

$$\delta = 29.625 - 29.438 = 0.187 \text{ in}$$

Utilizing the press fit equation (2-63) from Reference (28), the interface pressure, p , at a temperature of 70 °F, is:

$$\delta = pb_i \left(\left[\frac{c^2 + b_i^2}{c^2 - b_i^2} \right] + \mu_i \right) / E_i + b_o p \left(\left[\frac{b_o^2 + a^2}{b_o^2 - a^2} \right] - \mu_o \right) / E_o$$

The geometry and properties are, at 70 °F, as follows:

$$a = 28.500 \text{ in}$$

$$b_o = 29.625 \text{ in}$$

$$b_i = 29.438 \text{ in}$$

$$c = 32.921 \text{ in}$$

$$E_i = 28.3(10)^6 \text{ psi}$$

$$\mu_s = 0.3$$

$$E_s = 2.34(10)^4 \text{ psi}$$

$$\mu_l = 0.45$$

and,

$$\delta = b_s - b_l = 0.187 \text{ in}$$

Utilizing the press-fit equation above, the interface pressure is:

$$\delta = p[1.176(10)^{-4} + 2.674(10)^{-3}] = p[1.443(10)^{-4}] = 0.187 \text{ in}$$

Then,

$$p = 0.187/1.443(10)^{-4} = 1,296 \text{ psi interface pressure}$$

This corresponds to a lead shell hoop stress of:

$$\begin{aligned}\sigma_l &= p[(c^2 + b_l^2)/(c^2 - b_l^2)] \\ &= 11,538 \text{ psi}\end{aligned}$$

which, obviously, cannot be sustained.

To fully accommodate the 0.187 inch interference, the lead would be required to see a strain of $0.187/29.438 = 0.00635 \text{ in/in}$, or 0.635%. Extrapolating Figure 2.3-3 from Section 2.3, a hoop stress of approximately 900 psi will exist in the lead for this strain. The effective interface pressure would be:

$$\begin{aligned}p &= \sigma_l/[(c^2 + b_l^2)/(c^2 - b_l^2)] \\ &= 100 \text{ psi interface pressure}\end{aligned}$$

The δ of the cask inner steel shell with -100 psi pressure is:

$$\begin{aligned}\delta &= (b_i \{ [(b_i^2 + a^2) / (b_i^2 - a^2)] - \mu_i \} / E_i) (p) \\ &= 2.674(10)^{-5} (-100) = -0.00267 \text{ in}\end{aligned}$$

which can be conservatively neglected in the inner steel shell hoop stress calculations.

The hoop stress in the inner steel shell would be:

$$\begin{aligned}\sigma_i &= p [(b_i^2 + a^2) / (b_i^2 - a^2)] \\ &= -2,584 \text{ psi}\end{aligned}$$

From Figure 2.3-6 of Section 2.3, it is concluded that in 2 to 3 weeks, at 70 to 100 °F, the lead hoop stress would creep to a value of approximately 300 psi (essentially holding a constant lead strain of 0.635%). The corresponding interface pressure and inner steel shell hoop stress would be:

$$\begin{aligned}p &= \sigma_i / [(c^2 + b_i^2) / (c^2 - b_i^2)] \\ &= 33.7 \text{ psi interface pressure}\end{aligned}$$

$$\begin{aligned}\sigma_i &= p [(b_i^2 + a^2) / (b_i^2 - a^2)] \\ &= -871 \text{ psi}\end{aligned}$$

To determine a stress-free temperature, T_0 , assuming loading with heat-up will be elastic, thus:

$$\begin{aligned}\delta &= R(\alpha_i - \alpha_s) \Delta T \\ &= [1.443(10)^{-4}] p\end{aligned}$$

$$p = 33.7 \text{ psi}$$

$$R = 26.625 \text{ in}$$

$$\alpha_i = 16.07(10)^{-6} \text{ in/in/ } ^\circ\text{F}$$

$$\alpha_s = 8.24(10)^{-6} \text{ in/in/ } ^\circ\text{F}$$

$$\Delta T = T_o - 70$$

Then,

$$(26.625) [16.07(10)^{-6} - 8.24(10)^{-6}] (T_o - 70) = 1.443(10)^{-4} (33.7)$$

$$T_o = 93.3 ^\circ\text{F} \quad (\text{stress-free temperature})$$

Next, consider cooling to $-20 ^\circ\text{F}$ ($\Delta T = -90 ^\circ\text{F}$) for the worst hoop stress on the inner steel shell during all regulatory drop events. The steel and lead properties may be extrapolated from Table 2.3-1 as:

$$\alpha_s = 8.06(10)^{-6} \text{ in/in/ } ^\circ\text{F}$$

$$E_s = 29.0(10)^6 \text{ psi}$$

$$\alpha_l = 15.7(10)^{-6} \text{ in/in/ } ^\circ\text{F}$$

$$E_l = 2.43(10)^6 \text{ psi}$$

The steel and lead shell initial shell conditions are determined as:

$$a = (28.50) [1 + 8.06(10)^{-6}(-90)] = 28.479 \text{ in}$$

$$b_s = (29.625) [1 + 8.06(10)^{-6}(-90)] = 29.604 \text{ in}$$

$$b_l = (29.438) [1 + 15.7(10)^{-6}(-90)] = 29.396 \text{ in}$$

$$c = (32.921) [1 + 15.7(10)^{-6}(-90)] = 32.874 \text{ in}$$

and,

$$\delta = b_s - b_l = 0.208 \text{ in}$$

To fully accommodate the 0.208 inch interference, the lead would be required to see a strain of $0.208/29.396 = 0.00708 \text{ in/in}$, or 0.708%. Extrapolating from Figure 2.3-3 of Section 2.3, a hoop stress of approximately 1,075 psi will exist in the lead for this strain. The effective interface pressure would be:

$$p = \sigma_i / [(c^2 + b_i^2) / (c^2 - b_i^2)]$$

$$= 120 \text{ psi interface pressure}$$

The hoop stress in the inner steel shell, conservatively neglecting the beneficial effects of lead creep, would be:

$$\sigma_i = p[(b_s^2 + a^2) / (b_s^2 - a^2)]$$

$$= -3,099 \text{ psi}$$

Finally, consider cooling to -40°F ($\Delta T = -110^\circ\text{F}$) for the worst hoop stress on the inner steel shell. The steel and lead properties, extrapolated from Table 2.3-1, are:

$$\alpha_s = 8.00(10)^{-6} \text{ in/in/}^\circ\text{F}$$

$$E_s = 29.0(10)^6 \text{ psi}$$

$$\alpha_l = 15.6(10)^{-6} \text{ in/in/}^\circ\text{F}$$

$$E_l = 2.43(10)^6 \text{ psi}$$

The steel and lead shell initial shell conditions are determined as:

$$a = (28.50) [1 + 8.00(10)^{-6}(-110)] = 28.475 \text{ in}$$

$$b_s = (29.625) [1 + 8.00(10)^{-6}(-110)] = 29.599 \text{ in}$$

$$b_l = (29.438) [1 + 15.7(10)^{-6}(-110)] = 29.387 \text{ in}$$

$$c = (32.921) [1 + 15.7(10)^{-6}(-110)] = 32.865 \text{ in}$$

and,

$$\delta = b_s - b_l = 0.212 \text{ in}$$

To fully accommodate the 0.212 inch interference, the lead would be required to see a strain of $0.212/29.387 = 0.00721 \text{ in/in}$, or 0.721%. Extrapolating from Figure 2.3-3 of Section 2.3, a hoop stress of approximately 1,125 psi will exist in the lead for this strain. The effective interface pressure would be:

$$p = \sigma_i / [(c^2 + b_i^2) / (c^2 - b_i^2)]$$

$$= 125 \text{ psi interface pressure}$$

The hoop stress in the inner steel shell, conservatively neglecting the beneficial effects of lead creep, would be:

$$\sigma_s = p[(b_s^2 + a^2) / (b_s^2 - a^2)]$$

$$= -3,230 \text{ psi}$$

The preceding calculations deal only with the calculation of hoop stresses. Axial stress will also develop in the inner steel and lead shells due to axial shrinkage of the lead. In cooling to -20°F , axial strain in the lead, assuming bonding of the lead to the inner steel shell, is:

$$\epsilon = [(\alpha_s - \alpha_l)\Delta T]_{620} + [(\alpha_s - \alpha_l)\Delta T]_{-20}$$

$$= [(20.38 - 9.05)(10)^{-6}(620 - 70)] - [(15.7 - 8.06)(10)^{-6}(-20 - 70)]$$

$$= 0.00692 = 0.692\%$$

Extrapolating from Figure 2.3-3 of Section 2.3, an axial stress of approximately 1,015 psi will exist in the lead for this strain. The effective force in the lead shell would be:

$$P_s = pA_s$$

$$= (1,015)\pi[(32.874)^2 - (29.604)^2] = 651,465 \text{ lbs}$$

From equilibrium, this same force can develop in the inner steel shell. Thus, the compressive axial stress in the inner steel shell is:

$$\sigma_s = P/A_s$$

$$= -651,465/\pi[(29.604)^2 - (28.479)^2] = -3,174 \text{ psi}$$

This is a conservative estimate in that it assumes the ends of the outer cask inner shell are incompressible under axial load and that no load develops in the outer shell (i.e., any compressibility of the inner shell will reduce the lead strain and resultant axial forces and stresses).

In cooling to -40 °F, axial strain in the lead, assuming bonding of the lead to the inner steel shell, is:

$$\begin{aligned}\epsilon_a &= [(\alpha_l - \alpha_s)\Delta]_{620} + [(\alpha_l - \alpha_s)\Delta T]_{-40} \\ &= [(20.38 - 9.05)(10)^{-6}(620 - 70)] + [(15.6 - 8.00)(10)^{-6}(-40 - 70)] \\ &= 0.00707 = 0.707\% \text{ strain (tensile)}\end{aligned}$$

Extrapolating from Figure 2.3-3 of Section 2.3, an axial stress of approximately 1,070 psi will exist in the lead for this strain. The effective force in the lead shell would be:

$$\begin{aligned}P_a &= pA_l \\ &= (1,070)\pi[(32.865)^2 - (29.599)^2] = 685,772 \text{ lbs}\end{aligned}$$

From equilibrium, this same force can develop in the inner steel shell. Thus, the compressive axial stress in the inner steel shell is:

$$\begin{aligned}\sigma_s &= P/A_s \\ &= -685,772/\pi[(29.599)^2 - (28.475)^2] = -3,344 \text{ psi}\end{aligned}$$

Again, this calculation conservatively assumed that the cask inner shell is incompressible under axial load and that no load develops in the outer shell.

The preceding calculations in this section are based on the assumption that the cask inner and outer shells and the lead between them are always at the same temperature during cool down. Although this ideal condition can never be fully achieved in practice, the procedures used for lead pour, see Reference (27) are set up to minimize the temperature differences which do occur. To control the cool down process, thermocouples are used to monitor temperatures of the inner and outer shells of the outer cask. Thermocouples on the outer shell are located directly opposite those on the inner shell and the maximum acceptable temperature difference indicated by any opposing pair of thermocouples is limited to 300 °F. Temperature differences are not expected to ever be this great in practice, but even if they were, the following calculations indicate that they would be acceptable. An additional control placed on the thermocouple readings is that no single thermocouple reading shall change by more than 100 °F in any 30 minute

period. Recognizing the good conductivity exhibited by stainless steel and lead, this control essentially eliminates the consequences of transient effects.

Temperature differences between the inner and outer shells are of little consequence since the axial tie between the shells is made after lead cool down when a uniform temperature exists throughout the outer cask. By welding the outer shell to the base after the cool down process, temperature differences between the two shells will not cause axial stresses to develop in the shells during cool down. It can be shown that if the weld were made prior to lead pour, a condition where the inner shell was 300 °F hotter than the outer shell would result in elastically calculated axial stresses which would be unacceptable. Thus the need for final welding after cool down has been established.

The remaining issue to be addressed is that of temperature differences between the inner shell and the lead. The 300 °F maximum difference in temperature between inner and outer shells can conservatively be used as the maximum temperature difference between the inner shell and the lead. Further, if the inner shell is cooler than the lead, the previous calculations remain conservative. Specifically, with lower temperatures for the inner shell, than for the lead, the interference and associated contact pressure between the shell and lead is reduced thus reducing stress within the inner shell.

When a uniform temperature is finally achieved throughout the outer cask, the lead will have "caught up" with the inner shell and the previous calculations are directly applicable.

The remaining case of the inner shell being 300 °F hotter than the lead is addressed as follows by considering the lead as being at 70 °F and the inner shell at 370 °F. The analytic method is identical to that previously used in this section.

Per the previous calculations, the lead inner radius at 70 °F, assuming no interference with the inner shell is, $R_i = 29.438$ inches. The inner shell outside radius at 370 °F (assuming no interference from the lead) is as follows:

$$\begin{aligned} R_o &= (29.625) (1 + \alpha \Delta T) \\ &= (29.625) [1 + 8.75(10)^{-6}(300)] = 29.703 \text{ in} \end{aligned}$$

The resultant interference, δ , is:

$$\delta = R_2 - R_1 = 29.703 - 29.438 = 0.265 \text{ in}$$

The maximum strain in the lead, ϵ , therefore becomes:

$$\epsilon = \delta/R_1 = 0.265/29.438 = 0.009 \text{ in/in}$$

From Figure 2.3-3 of Section 2.3, a hoop stress, σ_1 , of approximately 940 psi will exist in the lead for this strain. The effective interface pressure, p , is:

$$\begin{aligned} p &= \sigma_1 / [(c^2 + b_1^2) / (c^2 - b_1^2)] \\ &= 105 \text{ psi} \end{aligned}$$

Where:

$$c = 32.921 \text{ inches per previous calc's (stress free lead at } 70^\circ\text{F)}$$

$$b_1 = 29.438 \text{ inches per previous calc's (stress free lead at } 70^\circ\text{F)}$$

$$a = (28.500) [1 + 8.75(10)^{-4}(300)] = 28.575 \text{ inches}$$

Thus, the hoop stress in the inner steel shell becomes:

$$\begin{aligned} \sigma_1 &= p[(b_2^2 + a^2) / (b_2^2 - a^2)] \\ &= -2,713 \text{ psi} \end{aligned}$$

Where:

$$a = 28.575 \text{ inches (stress free steel at } 370^\circ\text{F)}$$

$$b_2 = 29.703 \text{ inches (stress free steel at } 370^\circ\text{F)}$$

This compares with a previously determined hoop stress in the inner shell (with both lead and shell at 70°F) of -2,584 psi. Recognizing that 1) the -2,713 psi stress state only exists during the cool down process (when the shell is 300°F hotter than the lead) and that 2) when the inner shell continues to cool the hoop stress will reduce, it is concluded that the -2,713 psi hoop stress case is an

independent case which does not combine with other loadings such as those resulting from hypothesized drop events.

Regarding axial stresses, with the lead at 70 °F and the inner shell at 370 °F, the axial strain in the lead, ϵ_a , will be (per previous calculations, assuming bonding between steel and lead):

$$\begin{aligned}\epsilon_a &= [20.38(10)^{-6}](620 - 70) - [9.05(10)^{-6}](620 - 70) \\ &\quad + [8.75(10)^{-6}](370 - 70) \\ &= 0.00886 = 0.886\%\end{aligned}$$

Extrapolating from Figure 2.3-3 of Section 2.3, an axial stress of approximately 920 psi will exist in the lead for this strain.

The corresponding axial stress in the inner shell is -3,040 psi. Continued cooling of the inner shell to 70 °F will reduce this axial compressive stress. Further cooling of the lead and inner shell to -20 °F or -40 °F will result in axial stresses no worse than previously determined for these temperatures.

The preceding calculations indicate that stress levels resulting from lead pour and subsequent cool down will be relatively small. In particular, with compressive stress magnitudes in the range of 3,000 to 3,500 psi as calculated herein, it is concluded that buckling of the inner shell of the outer cask should not occur due to the lead pour procedure (i.e., critical buckling stresses as established in Appendix 2.10.3 for the outer cask inner shell are well above the predicted stress levels).

As additional verification that buckling of the cask inner shell does not occur as the result of lead pour and cool down, diametrical and straightness measurements for the inner shell are made before and after lead pour at various locations along the length of the shell (See Section 8.3.1). To be acceptable, the following two requirements must be satisfied:

- (1) All straightness and all diameter measurements taken before and after lead pour must meet the requirements specified on drawing 2111-201 (i.e., a 0.125 inch straightness requirement and a requirement that

diameters satisfy the tolerance requirements of the ASME Code, Section III, Subsection NE, Article NE-4220).

- (2) Changes observed in individual diameter measurements in going from pre-pour to post-pour conditions must be less than or equal to 0.125 inches.

Requirement (1) is a general tolerance requirement which must be met. Satisfaction of this requirement validates the capacity reduction factors used in the buckling evaluation of the cask inner shell (See Appendix 2.10.3.2).

Requirement (2) is intended to demonstrate that the shell did not buckle due to the lead pour procedure. The 0.125 inch diameter change limit (0.0625 inch radial change) is selected as being less than one-fourth the deflection associated with buckling of the outer cask inner shell.

The deflection associated with initiation of buckling is established as follows using Reference (29). Per Page 193 (Equation 162) of Reference (29), for a tube with an initial ellipticity, the limiting pressure, P_{yp} , is determined from:

$$P_{yp}^2 - [(\sigma_{yp}/m) + (1 + 6mn)P_{cr}]P_{yp} + \sigma_{yp} P_{cr}/m = 0.0$$

Where:

σ_{yp} = yield stress = 55,000 psi at 70 °F

$m = R/h = 25.834$

R = shell radius = 29.063 in

h = shell thickness = 1.125 in

$n = u_0/R$

u_0 = maximum initial radial deviation (inches)

P_{cr} = critical buckling stress = $Eh^3/4(1 - \mu^2)R^3$
= 457 psi (Equation 158, Reference (29))

$$E = 28.3(10)^6 \text{ psi}$$

$$\mu = 0.3$$

Thus,

$$P_{yp}^2 - [2,129 + (1 + 6u_0)457]P_{yp} + 972,943 = 0.0$$

With the NE-4221.1 requirement that the maximum difference in diameters, Δd , at a cross section must be within one percent (1%) of the nominal diameter, a maximum value for maximum initial radial deviation, u_0 , is determined to be $\Delta d/4$ as follows:

$$d_{max} = d_{nom} + 2u_0$$

$$d_{min} = d_{nom} - 2u_0$$

Subtracting these equations yields

$$(4)u_0 = d_{max} - d_{min} = \Delta d$$

or,

$$u_0 = \Delta d/4$$

With $\Delta d = 1\%$ of nominal diameter = $0.01(57.0) = 0.57$ in, the maximum initial radial deviation is:

$$u_0 = 0.1425 \text{ in}$$

In actuality, the initial radial deviation, u_0 , can vary from 0.0 (perfect cylinder) up to the 0.1425 inch value. The following table presents resultant values for the limiting pressure, P_{yp} , as a function of u_0 (i.e., solving the above quadratic equation).

Initial Radial Deviation <u>u_o (in)</u>	Limiting Pressure <u>P_{yp} (psi)</u>
0.001	456.25
0.01	449.65
0.05	423.00
0.10	394.61
0.1425	373.78

Finally, using Equation (160) of Reference (29), values for radial displacement, u_2 , of the inner shell under load corresponding to the limiting pressures, P_{yp} , are obtained as follows:

$$u_2 = [u_o P_{yp} / (P_{or} - P_{yp})] (\cos 2\theta)$$

where the maximum u_2 occurs for $(\cos 2\theta) = \pm 1$, or

$$\text{maximum } u_2 = [u_o P_{yp} / (P_{or} - P_{yp})]$$

With $P_{or} = 457$ psi as previously determined, the following table results:

Initial Radial Deviation <u>u_o (in)</u>	Radial Displacement Under Load P_{yp} <u>u_2 (in)</u>
0.001	0.608
0.01	0.612
0.05	0.622
0.10	0.632
0.1425	0.640

The conclusion is therefore reached that for buckling to occur, a radial displacement of over 0.60 inch would be required. The allowable radial displacement is set to one fourth of this value, or 0.15 inches. The corresponding allowable diametrical change is 0.30 inches. Note that if radial displacements sufficient to cause buckling (i.e., greater than 0.60 inches) did occur, the shell would no longer meet NE-4220 requirements since any radial change greater than 0.1425 inch would force a perfect cylinder outside of the acceptable range. Thus, if the shell meets NE-4220 requirements before and after lead pour, buckling could not have occurred.

As a final comment, much of the change in diameters between before and after lead pour can be related to the stress relieving of the inner shell during the lead pouring procedure. The exact quantification of all movement due to stress relieving is difficult at best due to the variance in fabrication stresses and actual material properties. However, those deflections due to relieving some weld induced stresses can be approximated.

During the process of welding the longitudinal seam in the one inch thick outer cask inner shell of a similar cask, an approximate 0.044 inch shrinkage in the inner circumference occurs due to the solidification of the longitudinal weld, (See Section 7.7-5 of Reference (30)). This shrinkage acts to impose an angular discontinuity and associated moment on the shell. The change in diameter of the shell due to this moment can be approximated by the use of Table 18, Case 5k of Reference (15). This approximation indicates that diametrical changes of up to 0.73 inches can be expected at the time of welding. Since the NuPac 140-B Cask inner shell is nearly identical, similar diametrical changes can be expected.

During the procedure for lead pour, temperatures in the 600 - 800 °F range are experienced by the shell. The shell is held at this temperature range for a sufficient length of time such that partial stress relief takes place. This partial annealing process will cause some movement of the vessel walls. Page 4-73 of Reference (31) indicates that the maximum stress relieving that can be expected at these temperatures is about 40%, or a 0.292 inch diametrical change was calculated for the aforementioned cask of Reference (27). The acceptance criteria of the cask, reported in Reference (27), was 0.125 inches conservatively represents this stress relief distortion. Since the NuPac 140-B Cask shells and

lead thickness are nearly identical, similar stress relief distortion can be expected.

These approximations do not take into consideration the added effects of the localized shrinkage of the circumferential welds or the stresses due to rolling of the shell.

2.6.3 Reduced External Pressure

The effect of reduced external pressure of 3.5 psia (11.2 psig internal pressure), per 10 CFR 71.71(c) (3), is considered negligible to the NuPac 140-B Cask. This conclusion is based upon the calculations presented in Section 2.6.1.3.2, which assesses cask survivability at a maximum normal operating pressure (MNOP) of 284 psig.

2.6.4 Increased External Pressure

The effect of increased external pressure of 20 psia (5.3 psig external pressure), per 10 CFR 71.71(c) (4), is considered negligible to the NuPac 140-B Cask due to the thick outer cask outer shell and end closures. Additionally, Section 2.6.7 addresses a multiplicity of different loading cases in excess of the prescribed requirements.

2.6.5 Vibration

The effect of vibrations normally incident to transport are considered to be negligible. This conclusion is drawn based on the small stresses obtained throughout the cask and payload baskets for the side drop event discussed in Sections 2.6.7 and 2.6.8. Per those sections, the normal side drop results in a maximum of 16.3 g's lateral acceleration on the cask which translates to stresses in the cask as follows:

<u>Component</u>	<u>Membrane Plus Bending Stress (psi)</u>
Cask Outer Shell	7,399
Cask Inner Shell	12,033
Basket Outer Shell	14,800
Basket Truss Core Panel	14,400

As a conservative worst case, it is assumed that normal vibration acceleration loads will equal the normal vertical tiedown loading imposed on tiedowns lugs. Utilizing the specification in 10 CFR 71.45(b) (1) of 2 g's in the vertical direction, the maximum stress, σ_{max} , is found by ratioing (2 g's/16.3 g's) to be 1,816 psi in the basket outer shell. This is well below the endurance limit for Type XM-19, stainless steel at 300 °F. From Figure 2.6.5-1 (Figure I-9.2.2 of Reference (6)), for 10^{11} cycles, and correcting for the ratio of elastic moduli, the allowable alternating stress intensity, σ_a , becomes (13,600) (27.0/28.3) = 12,975 psi.

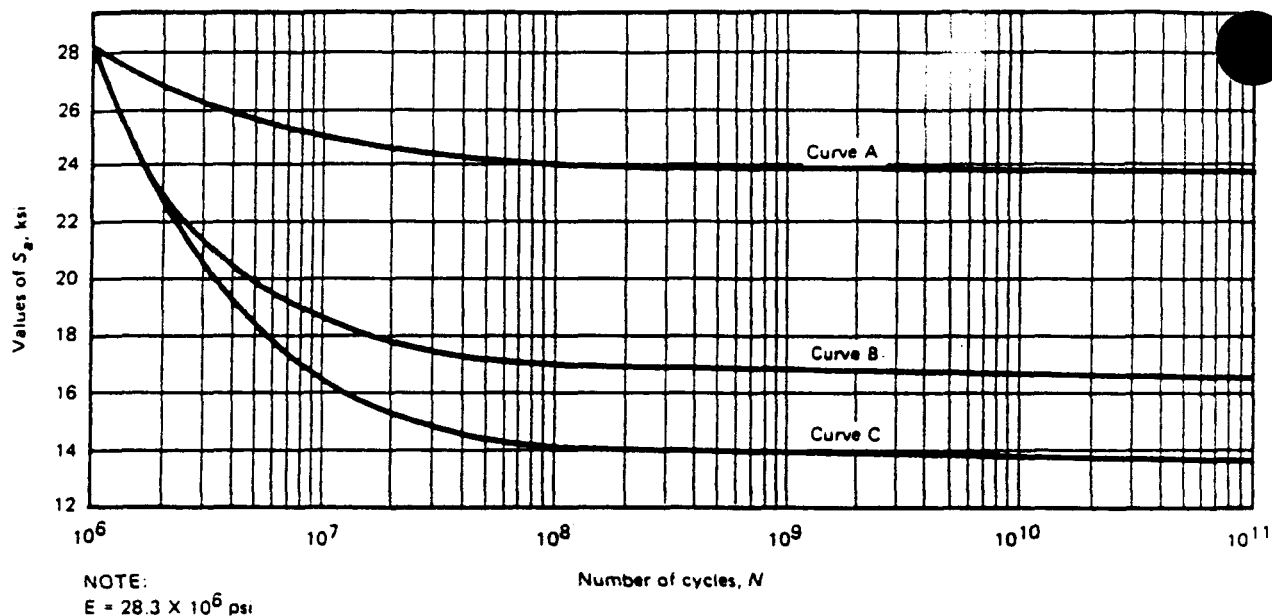
The basket outer shell vibratory stress Margin of Safety is:

$$\begin{aligned}
 M.S. &= (\sigma_a / \sigma_{max}) - 1 \\
 &= (12,975 / 1,816) - 1 = \underline{+6.15}
 \end{aligned}$$

1986 Edition

APPENDIX I

Fig. I-9.2.2



Criteria for the Use of the Curves in This Figure
[Notes (1)–(5)]

Curve	Elastic Analysis of Material Other Than Welds and Adjacent Base Metal	Elastic Analysis of Welds and Adjacent Base Metal
A	$(P_L + P_o + Q)_{\text{Range}} \leq 27.2 \text{ ksi}$	...
B	$(P_L + P_o + Q)_{\text{Range}} > 27.2 \text{ ksi}$ and S_e is corrected for applied mean stress	$(P_L + P_o + Q)_{\text{Range}} \leq 27.2 \text{ ksi}$
C	$(P_L + P_o + Q)_{\text{Range}} > 27.2 \text{ ksi}$	$(P_L + P_o + Q)_{\text{Range}} > 27.2 \text{ ksi}$

NOTES:

- (1) Range applies to the individual quantities P_L , P_o , and Q and applies to the set of cycles under consideration.
- (2) Thermal bending stresses resulting from axial and radial gradients are excluded from Q .
- (3) Curve A is also to be used with inelastic analysis with $S_e = \frac{1}{2} \Delta \epsilon_e E$, where $\Delta \epsilon_e$ is the total effective strain range.
- (4) The maximum effect of retained mean stress is included in Curve C.
- (5) The adjacent base metal is defined as three wall thicknesses from the center line of the weld.

FIG. I-9.2.2 DESIGN FATIGUE CURVE FOR AUSTENITIC STEELS, NICKEL-CHROMIUM-IRON ALLOY, NICKEL-IRON-CHROMIUM ALLOY, AND NICKEL-COPPER ALLOY FOR $S_e \leq 28.2 \text{ ksi}$, FOR TEMPERATURES NOT EXCEEDING 800°F
(For $S_e > 28.2 \text{ ksi}$, use Fig. I-9.2.1.)

Table I-9.2.2 Contains Tabulated Values for Accurate Interpolation of This Curve

FIGURE 2.6.5-1

The tie down lugs must also resist the 2 g vibration loading. In Section 2.5.2 the tie down lug stress intensity for a 1.4 g maximum railcar vibration input to the cask is 3,476 psi. At 2 g's the tie down lugs will experience a stress intensity of $(2.0/1.4) 3,476 = 4,966$ psi.

With a maximum temperature for the tiedown lugs taken at 300 °F, and assuming 10^{11} cycles of stress, the allowable alternating stress intensity, σ_a , becomes $(13,600) (27.0/28.3) = 12,975$ psi.

Therefore, the tiedown lug vibratory stress Margin of Safety is:

$$\begin{aligned} \text{M.S.} &= (12,975/4,966) - 1 \\ &= + \underline{1.61} \end{aligned}$$

The tie down lugs are mounted on the load collar thru which transport vibration loads are transferred to the cask. Thus, the stresses and margins must also be checked for the load collar, or as termed in Section 2.5.2 'the thermal shield to end ring weld joint,'. In Section 2.5.2 the load collar stress intensity for a 1.4 g maximum railcar vibration input to the cask is 5,304 psi. At 2 g's this component will experience a stress intensity of $(2.0/1.4) 5,304 = 7,577$ psi. The corresponding vibratory stress margin of safety is:

$$\begin{aligned} \text{M.S.} &= (12,975/7,577) - 1 \\ &= + \underline{0.71} \end{aligned}$$

The above margin assumes a maximum component temperature of 300°F.

2.6.6 Water Spray

Due to the materials of construction utilized for the NuPac 140-B Cask, the water spray test requirement, per 10 CFR 71.71(c) (6), will have a negligible effect on this package.

2.6.7 Free Drop

The maximum weight of the NuPac 140-B Cask is 203,159 pounds. Subpart F of 10 CFR 71 requires that a package with a weight in excess of 33,000 pounds be dropped one (1) foot onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected. The following subsections address free drops at all angles with respect to horizontal. It is shown that all conditions of 10 CFR 71.71 have been met.

The analyses in this section extract accelerations from Appendix 2.10.2 and statically apply them to the package. Static application is justified since the natural frequencies of the cask are relatively high and the duration of the impact loadings are relatively long. The cask frequencies are high as the result of using relatively thick, stiff shells for the outer cask. The duration of impact loadings are relatively long as the result of using soft (relative to the steel structures) energy absorbing impact limiters to protect the cask during free drop events. In addition, inspection of accelerometer data available from previous cask scale model drop tests indicates that this type of cask responds essentially as a rigid body (e.g., for end and oblique drops, all axial accelerometer traces are nearly identical and, for side drop, the accelerations for all accelerometers oriented in the direction of the drop are nearly identical). This observation further justifies static application of acceleration loads.

2.6.7.1 Flat End Drop

Analysis of the NuPac 140-B Cask behavior during the end drop impact is performed in the following steps:

- (1) Analyze the impact force using the EYDROP-PC computer program.
- (2) Analyze the cask lid for bending assuming the inner basket with payload act as a uniform load upon a simply supported plate (no impact limiter support).

- (3) Analyze the cask lid for bending, as performed in Step 2, but with the impact limiter foam pressure acting in the opposite direction (impact limiter support).
- (4) Analyze the stresses in the outer cask lid bolts.
- (5) Analyze the payload basket stresses.
- (6) Analyze the axial and hoop stresses in the cask shells and lead (maximum fabrication stress condition assumed).
- (7) Analyze the axial and hoop stresses in the cask shells and lead (zero fabrication stress condition assumed).
- (8) Determine the amount of lead slump.
- (9) Address fuel assembly payload behavior.

Material properties and allowable stresses are available respectively from Tables 2.3-1, 2.1-1 and 2.1-2. Material properties and allowables corresponding to maximum enveloping temperature (300 °F) for the outer cask and (300 to 600 °F) for baskets are as follows. Properties and allowables at other temperatures can be readily obtained from Tables 2.3-1, 2.1-1 and 2.1-2. Buckling is addressed in Appendix 2.10.3.

<u>Material Property</u>	<u>Cask</u>	<u>Baskets</u>
Elastic Modulus, E , x 10^6 psi	27.8	25.3 to 27.0
Yield Strength, S_y , ksi	43.4	37.3 to 43.4
Ultimate Strength, S_u , ksi	94.3	87.8 to 94.3
Design Stress, S_d , ksi	31.4	29.2 to 31.4
Allowable Stresses, ksi		
Containment Structures		
Normal Primary Membrane	31.4	29.2 to 31.4
Normal Primary Membrane plus Bending	47.1	43.8 to 47.1
Normal Pure Shear	18.8	17.5 to 18.8
Normal Bearing	43.4	37.3 to 43.4
Non-Containment Structures		
Normal Primary Membrane	43.4	37.3 to 43.4
Normal Primary Membrane plus Bending	47.1	43.8 to 47.1
Normal Pure Shear	26.0	22.4 to 26.0
Normal Bearing	43.4	37.3 to 43.4

Where possible, the analyses in this section are performed on the basis of a unit (1 g) axial acceleration, with actual acceleration numbers being substituted at the final point of calculating stresses. From Appendix 2.10.2, Table 2.10.2-8 the maximum acceleration load for the normal flat end drop case is 24.7 g's (for drop at -20 °F, fully effective foam).

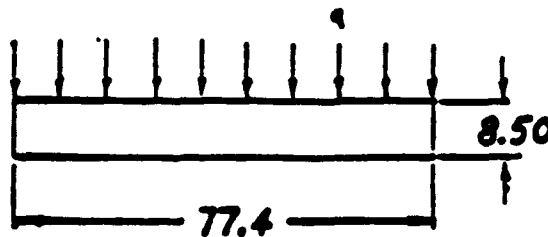
(1) End Drop Computer Analysis (EYDROP-PC)

The end drop analysis was performed utilizing the energy-balance computer program, EYDROP-PC, documented in Appendix 2.10.1. Two cases were run assuming fully effective impact limiters, and considering the variations in foam strength due to temperature effects and the possible variations in foam strength from nominal values (see Section 2.3 and Appendix 2.10.2). Fully effective impact limiter cases consider the entire impact limiter impact area available for crush.

This area does not include that of the relatively soft foam corners (approximately three pound per cubic foot density) extending 12.5 inches inward from the 128 inch outer diameter. Based on the scale test results from a similar cask (i.e., observed impact limiter displacements and cask accelerations), the fully effective impact limiter assumption is realistic. Output for each of the two end drop cases is presented in Appendix 2.10.2. The worst case deflections and accelerations are utilized for all analyses. From Table 2.10.2-9 of Appendix 2.10.2, the maximum normal condition acceleration and maximum impact limiter deflection for a one foot end drop is 24.7 g's and 1.04 inches, respectively.

(2) Cask Lid Bending Analysis Assuming No Impact Limiter Support

The effective thickness of the cask lid and the welded end closure plate is 8.50 inches thick. The effect of the borated silicone cover plate will be conservatively neglected. The welded end closure plate is reasonably treated as a fixed-edge plate. Treating the bolted lid as a simply supported plate is conservative. Therefore, to conservatively bound the problem, a 8.50 inch plate will be analyzed as simply supported. The lid loading is illustrated below:



The unit stress in the cask lid may be calculated by utilizing Table 24, Case 10a, of Reference (15), assuming a uniform load across the entire area:

$$\sigma = 6M/t^2 = 3qa^2(3 + \mu)/8t^2$$

Where:

$$q = W/A = 58,670/4,705 = 12.47 \text{ lb/in}^2$$

$$W = \text{BWR basket weight (16,508) + 52 BWR payload weight (32,188)} \\ + \text{outer cask lid weight (9,974)} = 58,670 \text{ lbs}$$

$$A = \text{inner vessel lid area (exterior)} \\ = \pi(38.70)^2 = 4,705 \text{ in}^2$$

$$a = 38.70$$

$$\mu = 0.3$$

$$t = 8.50 \text{ in}$$

Then:

$$\sigma = 320 \text{ psi per "g"}$$

Applying 24.7 g's acceleration from the normal condition end drop analysis, discussed in Appendix 2.10.3, the bending stress becomes:

$$\sigma = (24.7)(320) = 7,904 \text{ psi}$$

Table 2.6.7.1-1 combines this stress with that due to a 284 psig normal pressure loading. As shown, the resultant margin of safety is +2.04. This margin is somewhat conservative as accelerations applied to the lid are those corresponding to a drop at -20 °F, whereas stress allowables are taken at a temperature of 300 °F.

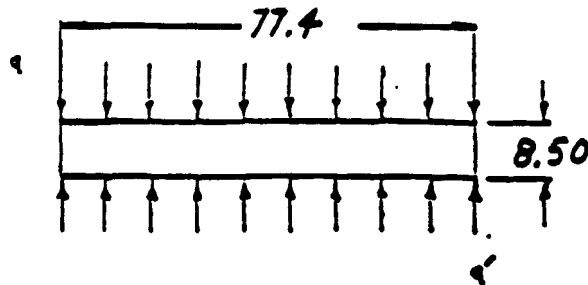
TABLE 2.6.7.1-1
STRESS SUMMARY: Cask Lid - Center of Plate

- Loading Condition: Normal Condition Flat End Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane Plus Bending
- Structural Criterion: Containment Boundary
- Stress Allowable (Section 2.1.2.1): $(1.5)S_u$
- Material Temperature: 300 °F
- Allowable Stress, S_u : 47,100 psi

	Stress Components (psi)				Reference Section
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					
Internal Pressure	-284	7,285	7,285	0	2.6.1.3.2
End Drop	0	7,904	7,904	0	2.6.7.1(2)
Direct Stress Sum:	-284	15,189	15,189	0	
Principal Stresses:	-284	15,189	15,189		
Stress Differences:	15,473	15,473	0		
Maximum Stress Intensity (SI):	15,473 psi				
Allowable Stress Intensity, S_u :	47,100 psi				
Margin of Safety ($[S_u/SI] - 1$):	<u>± 2.04</u>				

(3) Cask Lid Bending Analysis Assuming Full Impact Limiter Support

As analyzed above, assume the cask lid is supported across the impact plane by the foam. The lid loading is illustrated below:



Utilizing the same analysis as above, the lid bending stress is:

$$\sigma' = 3(q' - q)a^2(3 + \mu)/8t^2$$

Where:

$$q' = W'/A = 203,159/4,705 = 43.18 \text{ lb/in}^2$$

$$\begin{aligned} W' &= \text{total cask weight} \\ &= 203,159 \text{ lbs} \end{aligned}$$

$$\begin{aligned} A &= \text{outer cask lid area} \\ &= \pi(38.70)^2 = 4,705 \text{ in}^2 \end{aligned}$$

Then,

$$\sigma = 788 \text{ psi per } g$$

Applying 24.7 g's acceleration, the bending stress becomes:

$$\sigma = (24.7)(788) = 19,464 \text{ psi}$$

Internal pressure would reduce this bending stress. Therefore, the minimum outer cask lid bending stress Margin of Safety is:

$$\begin{aligned} \text{M.S.} &= (S_u/\sigma) - 1 \\ &= (47,100/19,464) - 1 = \underline{+1.42} \end{aligned}$$

(4) Outer Cask Lid Bolt Stresses

Outer cask bolt stresses under end impact conditions for worst case hot conditions, 300°F, are computed and summarized within Table 2.10.5-2 of Appendix 2.10.5. Assuming an impact acceleration of 24.7 g's acting on the full weight of the lid (9,974 lbs), BWR basket (16,508 lbs) and payload (32,188 lbs), the net bolt tensile stress is found as 58,553 psi, assuming equal loading of all 32 lid bolts.

These values include pressure and thermal stress effects. These stresses neglect impact limiter reaction forces which would reduce estimated bolt impact stresses to near zero (if considered). With a maximum total tensile stress of 58,553 psi, the minimum resultant Margin of Safety (versus $2S_u = 63,800$ psi) is computed, in Table 2.10.5-2, as +0.09.

Bolt engagement is checked as follows:

From the preceding calculations, the maximum bolt stress is 58,553 psi which corresponds to a maximum bolt force of 64,176 pounds. The minimum engagement length for these bolts is 2.0 inches. Per Reference (32), the shear area per inch of engagement for a 1-5/8-8NC-2B internal thread is 3.792 in²/in. The resultant shear stress is therefore:

$$\begin{aligned} \tau &= P/A \\ &= 64,176 / (2.0) (3.792) = 8,462 \text{ psi} \end{aligned}$$

With an allowable shear stress, S_s , for the forging of $(0.6)S_u = 18,800$ psi, the Margin of Safety becomes:

$$\begin{aligned} \text{M.S.} &= (S_s/\tau) - 1 \\ &= (18,800/8,462) - 1 = \underline{+1.22} \end{aligned}$$

(5) Payload Basket Stresses

Payload basket stresses and margins of safety are analyzed and presented in Section 2.6.8. Significant worst case results are summarized in Tables 2.6.7.1-2 for outer shells, and 2.6.7.1-3 for the truss core panels.

(6) Stresses in the Cask Shells and Head
(Maximum Fabrication Stress Condition Assumed)

The principal concern for the cask under normal end drop conditions is with buckling of the inner shell. As shown by the following stress calculations, and the detailed buckling evaluations presented in Appendix 2.10.3, buckling will not occur as the result of the hoop and axial compressive stresses which develop in the cask inner shell under normal end drop conditions.

Various initial conditions can be assumed for the normal drop event. In particular, a temperature must be assumed in order to establish an initial fabrication stress for the inner shell. A lower assumed temperature will result in a higher initial hoop stress on the inner shell (see Section 2.6.2) but higher allowable stresses. For purposes of this analysis, drops at 300 °F (conservatively bounding maximum lead and cask shells normal temperatures per Section 3.4.2 or Table 2.6.1-1), and -20 °F are considered. Regardless of temperature, the maximum acceleration load of 24.7 g's (for the -20 °F drop case) is conservatively employed.

To adequately bound the consequences of the drop event at a given temperature, two initial lead conditions are also considered. The first assumes that the lead has shrunk onto the inner shell and away from the outer shell. In addition, due to the combined effects of friction between the lead and inner shell, and axial shrinkage of the lead relative to the stainless shells, axial gaps will develop between the lead and the steel structures at the top and bottom end of the lead column. These axial gaps are important in that, until friction is overcome, under increased axial loading, the lead will impose a direct axial load on the inner shell. Once friction is overcome, the lead will become supported at its base (the bottom of the lead column) and will grow radially outward due to the "Poisson Effect" under increased axial loading. This radial growth will tend

TABLE 2.6.7.1-2
STRESS SUMMARY: Basket Outer Shell

- Loading Condition: Normal Condition Flat End Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Non-Containment Structure
- Stress Allowable (Section 2.1.2.1): S_y
- Material Temperature: 368 °F
- Allowable Stress, S_a : 41,500 psi

	Stress Components (psi)				Reference
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					<u>Section</u>
End Drop	0	0	-1,200	0	2.6.8.2
Direct Stress Sum:	0	0	-1,200	0	
Principal Stresses:	0	0	-1,200		
Stress Differences:	1,200	0	0		

Maximum Stress Intensity (SI): 1,200 psi
 Allowable Stress Intensity, S_a : 41,500 psi
 Margin of Safety ($[S_a/SI] - 1$): +34.6

TABLE 2.6.7.1-3
STRESS SUMMARY: Truss Core Panels

- Loading Condition: Normal Condition Flat End Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Non-Containment Structure
- Stress Allowable (Section 2.1.2.1): S_y
- Material Temperature: 420 °F
- Allowable Stress, S_a : 40,400 psi

	Stress Components (psi)				Reference
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					<u>Section</u>
End Drop	0	0	-1,200	0	2.6.8.2
Direct Stress Sum:	0	0	-1,200	0	
Principal Stresses:	0	0	-1,200		
Stress Differences:	1,200	0	0		

Maximum Stress Intensity (SI): 1,200 psi
 Allowable Stress Intensity, S_a : 40,400 psi
 Margin of Safety ($[S_a/SI] - 1$): +33.7

to relieve the initial fabrication hoop stress as the lead separates from the inner shell. If sufficient axial load develops, the lead would grow out to the outer shell creating tensile hoop stresses therein, and under further loading would eventually flow back inward into the inner shell, thereby developing compressive hoop stresses in the inner shell. From Section 2.6.2, inner shell hoop stress, σ_{it} , due to fabrication is as follows:

Temperature (°F)	σ_{it} (psi)
300	-1,134 (extrapolated)
-20	-3,099

The outer shell hoop stress, σ_{ot} , is considered negligible since the lead separates from the outer shell upon cooling. The equivalent pressure, p , at the lead/inner shell interface is as follows:

$$p = \sigma_{it}t/r$$

Where:

$$t = 1.25 \text{ in}$$

$$r = 29.625 \text{ in}$$

Thus, the interface pressures at the different temperatures are:

Temperature (°F)	Interface Pressure, p (psi)
300	47.8
-20	117.7

With a coefficient of friction, f , for lead on stainless steel assumed to fall in the 0.5 to 1.0 range (Reference (33), 3-26), the load, P , which can be supported by friction at the lead/inner shell interface, may be determined as follows:

$$P = \pi D L p f$$

Where:

$D = 59.25$ in (inner shell outside diameter)

$L = 177.0$ in (lead column height)

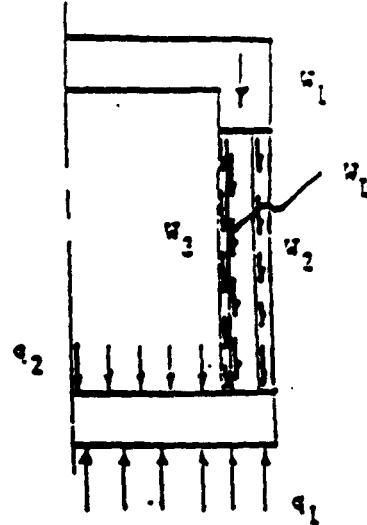
p = interface pressure, psi

$f = 0.5$ to 1.0 (coefficient of friction)

Applying the interface pressures determined earlier, the total load which may be supported is:

Temperature (°F)	Coefficient of Friction, f	Load Supported (lbs)
300	0.5	787,425
	1.0	1,574,850
-20	0.5	1,938,911
	1.0	3,877,823

With the total lead weight equal to 50,037 pounds, per Table 2.2-1, the minimum acceleration load which can be supported by friction is $(787,425/50,037) = 15.7 \text{ g's}$. This acceleration load is that which can be supported with the cask at an assumed (conservatively high) maximum normal operating temperature of 300 °F. At -20 °F, the minimum acceleration load that can be supported is $(1,938,911/50,037) = 38.7 \text{ g's}$. Although these acceleration loads bracket the maximum normal condition end drop of 24.7 g's at -20 °F, the lead will not be assumed to slip for the following analytical considerations. Slippage of the lead will be considered subsequently under item (7). Axial stress in the inner shell is determined, with reference to Table 2.2-1, as follows:



$$\begin{aligned}
 W_1 &= (\text{weight of top impact limiter, outer cask lid, and flange}) \\
 &\quad \times (24.7 \text{ g's}) \\
 &= (11,188) (24.7) = 276,344 \text{ lbs}
 \end{aligned}$$

$$\begin{aligned}
 W_2 &= (\text{weight of cask outer shell, thermal shield, etc.}) (24.7 \text{ g's}) \\
 &= (40,412) (24.7) = 998,176 \text{ lbs}
 \end{aligned}$$

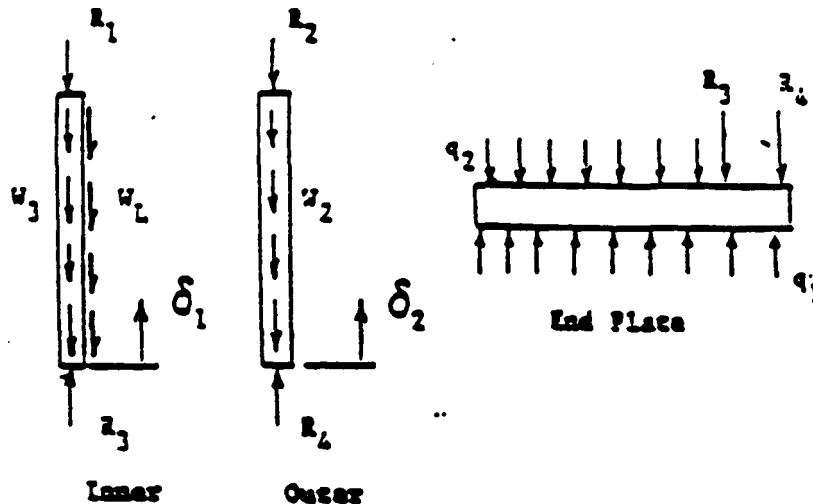
$$\begin{aligned}
 W_3 &= (\text{weight of cask inner shell}) (24.7 \text{ g's}) \\
 &= (10,541) (24.7) = 260,363 \text{ lbs}
 \end{aligned}$$

$$\begin{aligned}
 W_4 &= (\text{weight of lead}) (24.7 \text{ g's}) \\
 &= (50,037) (24.7) = 1,235,914 \text{ lbs}
 \end{aligned}$$

$$\begin{aligned}
 q_2 &= (\text{weight of basket and payload}) (24.7 \text{ g's}) / \pi (28.5)^2 \\
 &= (48,696) (24.7) / 2,552 = 471.3 \text{ lbs/in}^2
 \end{aligned}$$

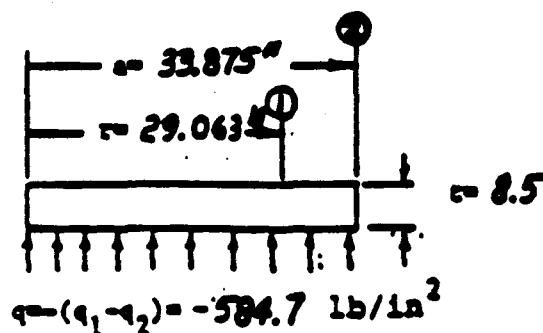
$$\begin{aligned}
 q_1 &= [W_1 + W_2 + W_3 + W_L + q_2 \pi (28.5)^2] / \pi (34.625)^2 \\
 &= 3,973,588 / 3,766 = 1,056 \text{ lbs/in}^2
 \end{aligned}$$

Free-body diagrams of the outer cask inner and outer shells, and end plate are illustrated below:



Assuming reaction R_1 does not affect the deflection of the lower end plate (conservative for maximum R_3), the differential deflection, $\delta_1 - \delta_2$, may be calculated. Utilizing Table 24, Case 10a, of Reference (15), assuming a simply supported circular plate with a uniform load, the differential deflection may be determined as follows:

$$\delta_1 - \delta_2 = y_o + M_o r^2 / 2D(1 + \mu) - q r^4 G_{11} / D$$



Where:

$$\begin{aligned} y_o &= -qa^4(5 + \mu)/64D(1 + \mu) \\ &= 872,113/E \end{aligned}$$

$$\begin{aligned} M_o &= qa^2(3 + \mu)/16 \\ &= -138,384 \end{aligned}$$

$$\begin{aligned} D &= Et^3/12(1 - \mu^2) \\ &= (56.238)E \end{aligned}$$

$$\begin{aligned} G_{11} &= \left[(1 + 4(r_o/r)^2 - 5(r_o/r)^4 - 4(r_o/r)^2[2 \right. \\ &\quad \left. + (r_o/r)^2] \ln(r_o/r)) / 64 \right] < r - r_o >^0 \\ &= 1/64 \end{aligned}$$

$$\begin{aligned} q &= -(q_1 - q_2) = \\ &= -(1,056.0 - 471.3) = -584.7 \text{ lb/in}^2 \end{aligned}$$

$$a = 33.875 \text{ in}$$

$$r = 29.063 \text{ in}$$

$$r_o = 0$$

$$t = 8.5 \text{ in}$$

$$\mu = 0.3$$

Then, the differential deflection, $\delta_1 - \delta_2$, may be rewritten as:

$$\begin{aligned} \delta_1 - \delta_2 &= 872,113/E - (138,384)(29.063)^2/2E(56.238)(1 + 0.3) \\ &\quad + (584.7)(29.063)^4/64E(56.238) \\ &= 872,113/E - 799,399/E + 115,900/E = 188,614/E \end{aligned}$$

Also, from the free-body diagrams for the outer cask inner and outer shells and axial stiffness relations for the shells ($\delta = PL/AE$ for an end load, and $\delta = PL/2AE$ for self-weight, i.e., distributed load), the deflections, δ_1 and δ_2 , and reactions, R_3 and R_4 , may be found:

$$\delta_1 = R_3 L / A_1 E - (W_3 + W_L) L / 2 A_1 E$$

$$\delta_2 = R_4 L / A_2 E - W_2 L / 2 A_2 E$$

Where:

$$L = 177.0 \text{ in}$$

$$\begin{aligned} A_1 &= \pi[(29.625)^2 - (28.500)^2] \\ &= 205.4 \text{ in}^2 \end{aligned}$$

$$\begin{aligned} A_2 &= \pi[(34.625)^2 - (33.125)^2] \\ &= 319.3 \text{ in}^2 \end{aligned}$$

Then,

$$\delta_1 = (0.8617/E) R_3 - 644,696/E$$

$$\delta_2 = (0.5543/E) R_4 - 276,663/E$$

and,

$$\begin{aligned} R_3 + R_4 &= W_1 + W_2 + W_3 + W_L \\ &= 2,770,797 \text{ lbs} \end{aligned}$$

Solving simultaneously,

$$\begin{aligned} R_3 &= [188,614/E + 644,696/E - 276,663/E \\ &\quad + 2,770,797(0.5543/E)] / (0.8617/E + 0.5543/E) \\ &= 1,477,754 \text{ lbs} \end{aligned}$$

$$R_4 = 2,770,797 - 1,477,754 = 1,293,043 \text{ lbs}$$

Therefore, the axial compressive stress in the cask inner shell, $\sigma_{i,}$ and outer shell, $\sigma_{o,}$ is:

$$\sigma_{i,} = R_3/A_1 = 7,195 \text{ psi}$$

$$\sigma_{o,} = R_4/A_2 = 4,050 \text{ psi}$$

It has been shown above, based upon the stated assumptions, that Young's Modulus, E , is independent of the axial stresses calculated. Therefore, the shell stresses are summarized, with reference to Section 2.6.2 for hoop stresses, as follows:

Temperature (°F)	Inner Shell Stresses (psi)		Outer Shell Stresses (psi)	
	<u>Axial ($\sigma_{i,}$)</u>	<u>Hoop ($\sigma_{h,}$)</u>	<u>Axial ($\sigma_{o,}$)</u>	<u>Hoop ($\sigma_{h,}$)</u>
300	-7,195	-1,134	-4,050	0
-20	-7,195	-3,099	-4,050	0

Tables 2.6.7.1-4 and 2.6.7.1-5 present the corresponding Margins of Safety for the more critical inner shell. An internal pressure of zero is used in order to maximize compressive stresses in the inner shell. Note: Axial stresses will actually decrease with increasing drop temperature since acceleration loads are lower at higher temperatures as shown in Table 2.10.2-8 of Appendix 2.10.2 (due to decreased foam compressive strength with increased temperature). This effect is conservatively ignored in Tables 2.6.7.1-4 and 2.6.7.1-5. Buckling is addressed in Appendix 2.10.3.

- (7) Stresses in the Cask Shells and Lead
(Zero Fabrication Stress Condition Assumed)

The second initial lead condition assumes that the fabrication stress has fully crept away resulting in a stress free column of lead just in contact with the inner and outer shells. This is a potential worst case since any axial load imposed on the lead will directly load, radially, both the inner and outer shells (i.e., the lead need not flow away from the inner shell, into the outer shell and back into the inner shell to develop a compressive hoop stress in the inner shell).

TABLE 2.6.7.1-4
STRESS SUMMARY: Cask Inner Shell, Maximum
Fabrication Stresses at -20 °F, Zero Internal Pressure

-
- Loading Condition: Normal Condition Flat End Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Containment Boundary
 - Stress Allowable (Section 2.1.2.1): S_a
 - Material Temperature: -20 °F
 - Allowable Stress, S_a : 33,300 psi

	Stress Components (psi)				Reference
	Radial σ_r	Hoop σ_θ	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					<u>Section</u>
Fab. and End Drop	0	-3,099	-7,195	0	2.6.7.1(6)
Direct Stress Sum:	0	-3,099	-7,195	0	
Principal Stresses:	0	-3,099	-7,195		
Stress Differences:	7,195	4,096	3,099		

Maximum Stress Intensity (SI): 7,195 psi
 Allowable Stress Intensity, S_a : 33,300 psi
 Margin of Safety ($(S_a/SI) - 1$): +3.61

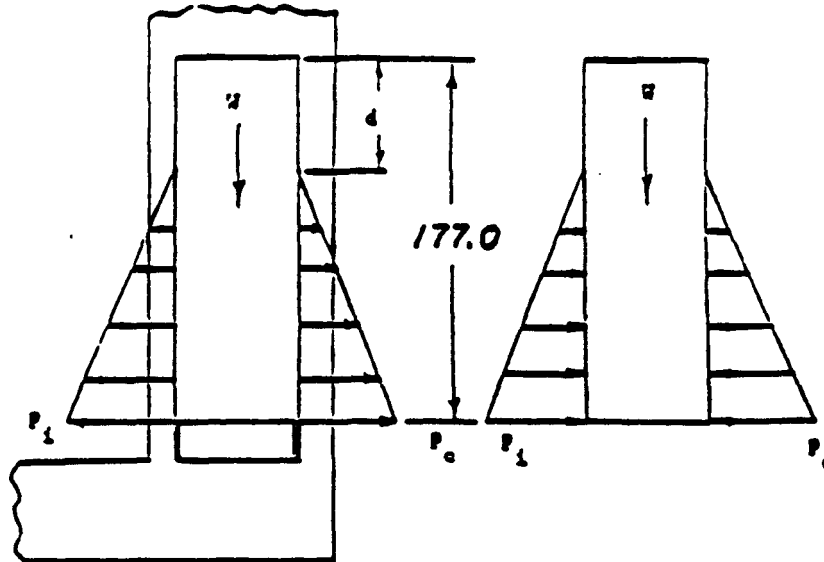
TABLE 2.6.7.1-5
STRESS SUMMARY: Cask Inner Shell, Maximum
Fabrication Stresses at 300 °F, Zero Internal Pressure

- Loading Condition: Normal Condition Flat End Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Containment Boundary
- Stress Allowable (Section 2.1.2.1): S_a
- Material Temperature: 300 °F
- Allowable Stress, S_a : 31,400 psi

	Stress Components (psi)				Reference Section
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					
Fab. and End Drop	0	-1,134	-7,195	0	2.6.7.1(6)
Direct Stress Sum:	0	-1,134	-7,195	0	
Principal Stresses:	0	-1,134	-7,195		
Stress Differences:	7,195	6,061	1,134		

Maximum Stress Intensity (SI): 7,195 psi
 Allowable Stress Intensity, S_a : 31,400 psi
 Margin of Safety ($[S_a/SI] - 1$): +1.36

For this condition, initial stresses in the lead and the steel shells are taken as zero. As axial load is applied to the lead and shells, the lead will attempt to move downward and outward and develop pressures on both the inner and outer shells as shown below:



For purposes of this analysis, let $p_i = p_o = p$. This is considered reasonable once lead starts to flow.

Conservatively assuming that the lead receives no axial support at its base, but rather is supported solely by friction at the inner and outer shells, the maximum value of the pressure, p , is determined as follows:

$$W = W_L = 1,235,914 \text{ lbs (with 24.7 g's inertia loading)}$$

For the maximum pressure, p , use the minimum expected coefficient of friction of $f = 0.5$. Then, the force, W , is:

$$W = \pi(29.625 + 33.125)(177.0 - d)f$$

Therefore,

$$P_{max} = 15,135 / (177.0 - d)$$

Assuming lead flow initiates at the yield strength of the lead, the height of the lead column, d , required to yield the lead cross-section due to 24.7 g's normal condition acceleration is:

$$\sigma_y = \rho d(24.7 \text{ g's})$$

Where:

$$\rho = 0.41 \text{ lbs/in}^3$$

Then,

$$d = (0.0987)\sigma_y$$

Therefore, for 300 °F, and -20 °F, and conservatively estimating an upper bound on the lead properties presented in Figures 2.3-1 through 2.3-4, the maximum pressure, p_{max} , and hoop stress, $\sigma_t = p_{max}R/t$, may be summarized as follows:

Temperature (°F)	σ_y (psi)	d (in)	p_{max} (psi)	σ_{it} (inner shell)	σ_{ot} (outer shell)
300	~350	34.5	106.2	-2,744	2,398
-20	~1,000	98.7	193.3	-4,994	4,365

Maximum axial stress in the shells is determined using the same approach as developed in the preceding calculations in Step 6, except that one-half the lead weight is considered to act on each shell rather than all lead weight being carried on the inner shell. Again, no support is considered at the base of the lead.

Repeating the previous calculational approach,

$$\delta_1 - \delta_2 = 188,614/E$$

$$\begin{aligned}\delta_1 &= R_3 L / A_1 E - (W_3 + W_L / 2) L / 2 A_1 E \\ &= (0.8617/E) R_3 - 378,439/E\end{aligned}$$

$$\begin{aligned}\delta_2 &= R_4 L / A_2 E - (W_2 + W_L / 2) L / 2 A_2 E \\ &= (0.5543/E) R_4 - 447,942/E\end{aligned}$$

$$R_3 + R_4 = W_1 + W_2 + W_3 + W_L = 2,770,797 \text{ lbs}$$

Solving simultaneously,

$$\begin{aligned}R_3 &= [188,614/E + 378,439/E - 447,942/E \\ &\quad + 2,770,797(0.5543/E)] / (0.8617/E + 0.5543/E) \\ &= 1,801,446 \text{ lbs}\end{aligned}$$

$$R_4 = 2,770,797 - 1,801,446 = 969,351 \text{ lbs}$$

Therefore, the axial compressive stress in the outer cask inner shell, $\sigma_{i,}$ and outer shell, $\sigma_{o,}$ is:

$$\sigma_{i,} = R_3 / A_1 = 4,719 \text{ psi}$$

$$\sigma_{o,} = R_4 / A_2 = 3,036 \text{ psi}$$

The calculated stresses at 300 °F, and -20 °F, for a normal condition drop load of 24.7 g's, are presented below:

Temperature (°F)	Inner Shell Stresses (psi)		Outer Shell Stresses (psi)	
	Axial ($\sigma_{i,}$)	Hoop ($\sigma_{i,}$)	Axial ($\sigma_{o,}$)	Hoop ($\sigma_{o,}$)
300	-4,719	-2,744	-3,036	2,398
-20	-4,719	-4,994	-3,036	4,365

Tables 2.6.7.1-6 through 2.6.7.1-7 present the corresponding Margins of Safety for the more critical inner shell. Zero pressure is used in order to maximize compressive stresses in the inner shell. Again, it is noted that the margin presented for the 300 °F case is conservative in that at this temperature, the acceleration load will actually be less than the 24.7 g's conservatively considered herein. Buckling is addressed in Appendix 2.10.3.

TABLE 2.6.7.1-6
 STRESS SUMMARY: Cask Inner Shell, Zero
 Fabrication Stresses at -20 °F, Zero Internal Pressure

- Loading Condition: Normal Condition Flat End Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Containment Boundary
- Stress Allowable (Section 2.1.2.1): S_a
- Material Temperature: -20 °F
- Allowable Stress, S_a : 33,300 psi

	Stress Components (psi)				Reference
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					<u>Section</u>
End Drop	0	-4,994	-4,719	0	2.6.7.1(7)
Direct Stress Sum:	0	-4,994	-4,719	0	
Principal Stresses:	0	-4,994	-4,719		
Stress Differences:	4,994	4,719	275		

Maximum Stress Intensity (SI): 4,994 psi
 Allowable Stress Intensity, S_a : 33,300 psi
 Margin of Safety ($[S_a/SI] - 1$): +5.67

TABLE 2.6.7.1-7
STRESS SUMMARY: Cask Inner Shell, Zero
Fabrication Stresses at 300 °F, Zero Internal Pressure

- Loading Condition: Normal Condition Flat End Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Containment Boundary
- Stress Allowable (Section 2.1.2.1): S_m
- Material Temperature: 300 °F
- Allowable Stress, S_a : 31,400 psi

	Stress Components (psi)				Reference Section
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					
End Drop	0	-2,744	-4,719	0	2.6.7.1(8)
Direct Stress Sum:	0	-2,744	-4,719	0	
Principal Stresses:	0	-2,744	-4,719		
Stress Differences:	4,719	2,744	1,973		

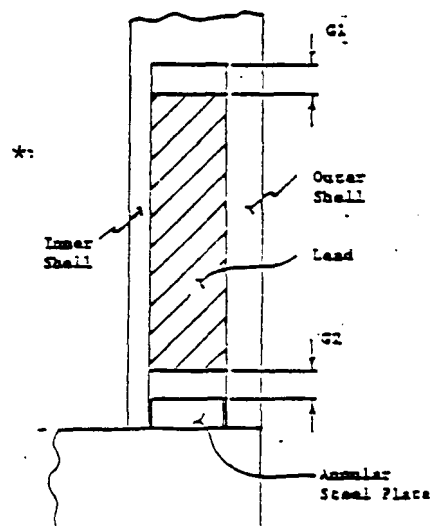
Maximum Stress Intensity (SI): 4,719 psi
 Allowable Stress Intensity, S_a : 31,400 psi
 Margin of Safety ($[S_a/SI] - 1$): +6.63

(8) Lead Slump

The following Table 2.6.7.1-8 summarizes the status of the lead within the shells at fabrication conditions following normal and hypothetical accident end drop conditions. For this table, it is assumed that friction is overcome before reaching the normal drop load of 24.7 g's. The data in the table were extrapolated, for NuPac 140-B Cask application, based on similar results for a similar Nuclear Packaging Cask reported in Reference (27), which were calculated employing finite element structural modeling methods. Since both casks are quite similar in the design of the lead gamma shields and thicknesses of the steel shells, the results in the following table are comparable. This is demonstrated in the following hand calculations.

TABLE 2.6.7.1-8
Summary of Lead Slump Analyses

Condition	300 °F		-20 °F	
	G1 (inches)	G2 (inches)	G1 (inches)	G2 (inches)
Fabrication (1 g)	0.264	0.169	0.410	0.315
Normal End Drop (24.7 g's)	0.732	0.000	0.714	0.000
Accident End Drop (49.1 g's)	1.670	0.000	1.425	0.000



For purposes of comparison, lead slump, ΔH , will also be calculated using the lead slump equation available in Reference (24) (Eq. 2.16, Page 63):

$$\Delta H = RWH/\pi[(R^2 - r^2)(t_o\sigma_o + R\sigma_{pb})]$$

Where:

R = lead outer radius

= 33.125

W = lead weight

= 50,037

H = drop height

= 12.0 in

r = lead inner radius

= 29.625

t_o = outer shell thickness

= 1.50

σ_o = approximate dynamic flow stress of outer shell

= 50,000 psi

σ_{pb} = approximate dynamic flow stress of lead

= 5,000 psi

Then,

$$\Delta H = 0.120$$

This slump is in addition to the initial gaps at the top and bottom ends of the lead column due to fabrication conditions, as depicted in the previous table. Therefore, adding this 0.120 inch slump to the sum of the initial fabrication induced gaps determined from the finite element analyses yields the following for the one (1) foot normal condition drop case:

Temperature	Total Lead Slump
(°F)	(inches)
300	0.553
-20	0.845

It is noted that the preceding lead slump calculation employed typical flow stress values for steel and lead (σ_s , σ_{pb}) and did not account for variations of these parameters with temperature. Flow stress will increase with decreasing temperature, thereby decreasing the amount of lead slump, ΔH . The results presented in the previous tables, which are based upon extrapolation of finite element evaluation, more accurately reflects temperature effects; thus, the finite element determined lead slump values will be used elsewhere in this report as required. It is also noted that the lead slump formula in Reference (24) is based on the drop of a "bare", that is, unprotected cask. As the NuPac 140-B Cask is well protected by energy absorbing impact limiters, the lead slump formula does not directly apply.

(9) Fuel Assembly Payload

Considerations of the payload fuel assemblies for the normal condition end drop loading are discussed in Section 2.9.

2.6.7.2 Corner and Oblique Drops

Analysis of the NuPac 140-B Cask behavior during the corner drop, including oblique drop orientations, will be performed in the following steps:

- (1) Determine the force-deflection relations for a range of initial impact angles, using the CYDROP-PC computer program. Utilize the force-deflection data from CYDROP-PC as input to the computer program OBLIQUE-PC to determine the package response for various initial impact angles.

- (2) Based on the maximum moments, shear and thrust forces predicted in Step (1), determine the equivalent "static" forces on the cask and basket, both considered as rigid bodies with coincident mass centers and zero relative motion.
- (3) Based on the loads from Step (2), analyze the cask shells and payload baskets as simple beams, finding the internal forces: shear, moment, and thrust. Compute stresses for the maximum value of each of these forces, and by inspection of the relative magnitudes of these maximum component stresses, select load cases for further analysis which will have the maximum combined stresses.
- (4) Analyze the cask shells for stress, assuming the two shells act as parallel beams, under the loads determined in Step (3).
- (5) Analyze the fuel assembly payload baskets for stress under the loads determined in Step (3).
- (6) Analyze the impact limiter attachment forces.
- (7) Based on the forces from Step (6), calculate the impact limiter attachment stresses.
- (8) Analyze the stresses in the cask lid bolts.
- (9) Calculate the maximum impact limiter deformations and residual clearances for cask hard spots.
- (10) Discussion of lead effects in oblique drops.

Material properties and allowable stresses are available respectively from Tables 2.3-1, 2.1-1 and 2.1-2. Material properties and allowables corresponding to maximum enveloping temperature (300 °F) for the outer cask and (300 to 600 °F) for baskets are as follows. Properties and allowables at other temperatures can be readily obtained from Tables 2.3-1, 2.1-1 and 2.1-2. Buckling is addressed in Appendix 2.10.3.

<u>Material Property</u>	<u>Cask</u>	<u>Baskets</u>
Elastic Modulus, E, x 10 ⁶ psi	27.8	25.3 to 27.0
Yield Strength, S _y , ksi	43.4	37.3 to 43.4
Ultimate Strength, S _u , ksi	94.3	87.8 to 94.3
Design Stress, S _m , ksi	31.4	29.2 to 31.4
Allowable Stresses, ksi		
Containment Structures		
Normal Primary Membrane	31.4	29.2 to 31.4
Normal Primary Membrane plus Bending	47.1	43.8 to 47.1
Normal Pure Shear	18.8	17.5 to 18.8
Normal Bearing	43.4	37.3 to 43.4
Non-Containment Structures		
Normal Primary Membrane	43.4	37.3 to 43.4
Normal Primary Membrane plus Bending	47.1	43.8 to 47.1
Normal Pure Shear	26.0	22.4 to 26.0
Normal Bearing	43.4	37.3 to 43.4

(1) Corner Drop Computer Analysis (CYDROP-PC and OBLIQUE-PC)

Analytic predictions of package performance for drop orientations impacting upon the corner of the package employ two computer programs: CYDROP-PC and OBLIQUE, documented in Appendix 2.10.1. CYDROP-PC uses an energy-balance technique to determine loads and deformations of the impact limiter. Since CYDROP-PC assumes all drop energy is absorbed in deformations of the impact limiter, it provides valid results when the impact orientation places the center of gravity directly over the impacted corner, approximately 25° from the vertical for the NuPac 140-B Cask. At other impact orientations, the force-deflection values generated by CYDROP-PC are used in OBLIQUE-PC, a dynamic analysis program that considers rotational motion effects.

Two (2) cases were run which accounted for variations in foam strength due to temperature effects and the possible variations in foam strength from nominal values (see Section 2.3 and Appendix 2.10.2). Detailed output for each of the analysis cases is presented in Appendix 2.10.2. Note: The load and stress evaluations presented in this section are typically based on the results from the -20 °F foam stress case. Impact limiter responses (attachment forces, deformations, and residual clearances) presented herein directly consider both analysis cases.

(2) Equivalent "Static" Loads

The cask and payload baskets are treated as rigid bodies under longitudinal, transverse and angular accelerations, along with an angular velocity. The two bodies will be assumed to move together with no relative motion. The centers of gravity of the two bodies are assumed to be coincident.

Based on quarter-scale drop test results from Reference (27) of a similar Nuclear Packaging cask, no significant loads on the cask structure are anticipated to result from oblique drop orientations for normal conditions of transport. This conclusion will be verified during final design and can also be discerned from Table 2.10.2-8 of Appendix 2.10.2. There it is shown that the maximum acceleration during oblique drops is 11.1 g's which is less than half that for the maximum end drop acceleration of 24.7 g's. Thus, analysis Steps (3) through (5) will not be included for preliminary design. The only potentially serious loading problem will affect cask impact limiter attachment fastener and outer cask lid fastener integrity. These load cases are dealt with in Step (6) below.

(3) Internal Forces

To be completed in final design.

(4) Cask Stresses

To be completed in final design.

(5) Payload Basket Stresses

To be completed in final design.

(6) Impact Limiter Attachment Forces

The maximum attachment moment for the impact limiters, for normal conditions of transport, is given in Table 2.10.2-22 of Appendix 2.10.2 as $1.33(10)^7$ in-lbs. Eight (8), ASTM A-320, Grade L43, 1-1/2 - 6 UNC bolts (necked down to a 1.25 inch diameter) secure each impact limiter to the outer cask. Assuming that tension varies linearly from one side of the cask to the other, the moment is resisted by six of the eight bolts as follows:

$$\begin{aligned} M &= 2P_{\max} [(66.288)^2 + (46.873)^2 \\ &\quad + (19.415)^2] / 66.288 \\ &= (210.24) P_{\max} \end{aligned}$$

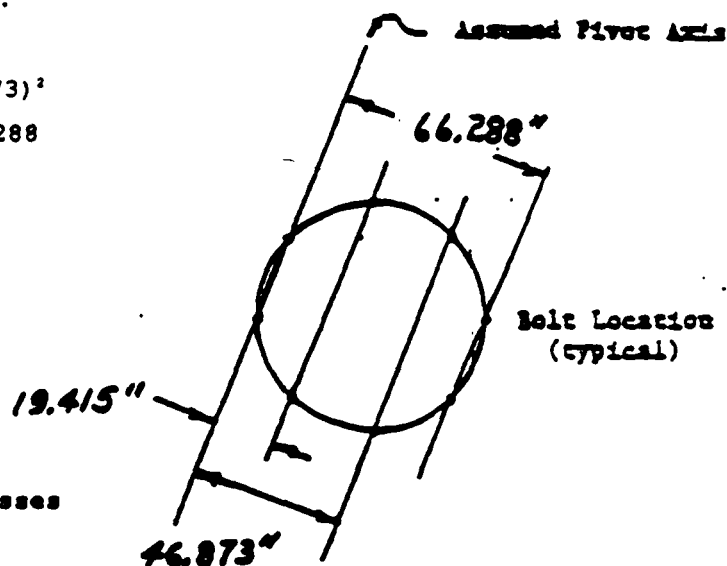
Thus,

$$\begin{aligned} P_{\max} &= M / 210.24 \\ &= 1.33(10)^7 / 210.24 \\ &= 63,261 \text{ lbs} \end{aligned}$$

(7) Impact Limiter Attachment Stresses

The stress area of a 1-1/2 - 6 UNC bolt is 1.3832 in² per Reference (32), thus, the 1.25 inch shaft provides the minimum cross-sectional area. The maximum impact limiter attachment bolt stress, σ_b , is:

$$\begin{aligned} \sigma_b &= P_{\max} / A = 63,261 / (\pi/4) (1.25)^2 \\ &= 51,550 \text{ psi} \end{aligned}$$



From Table 2.1-2, the allowable stress for normal condition, non-containment fasteners is given as the greater of $(2.0)S_u$ or S_y :

$$(2.0)S_u = 2.0(31,900) = 63,800 \text{ psi at } 300^\circ\text{F}$$

$$S_y = 95,700 \text{ psi at } 300^\circ\text{F}$$

Therefore, the allowable stress, σ_a , is 95,700 psi and the impact limiter attachment bolt Margin of Safety is:

$$\begin{aligned} \text{M.S.} &= (\sigma_a/\sigma_u) - 1 \\ &= (95,700/51,550) - 1 = \underline{+0.86} \end{aligned}$$

Bolt engagement is checked as follows. From the preceding calculations, the maximum bolt force is 63,261 pounds. Per the General Arrangement Drawings in Appendix 1.3.2), the minimum engagement length for the impact limiter attachment fasteners is 2.00 inches. Per Reference (32), the shear area per inch of engagement for a 1-1/2 - 6UNC-2B internal thread is 3.566 in²/in. The resultant shear stress is therefore:

$$\begin{aligned} \tau &= P/A \\ &= 63,261 / (2.00) (3.566) = 8,870 \text{ psi} \end{aligned}$$

With an allowable shear stress, τ_a , for the Type XM-19 stainless steel nut of 18,800 psi per Table 2.1.2-1, the Margin of Safety becomes:

$$\begin{aligned} \text{M.S.} &= (\tau_a/\tau) - 1 \\ &= (18,800/8,870) - 1 = \underline{+1.22} \end{aligned}$$

(8) Stresses in the Cask Lid Bolts

There are four basic loading mechanisms for the closure bolts of the cask:

1. Longitudinal impact forces inducing tensile bolt stresses which attempt to separate the lid from the body.
2. Lateral impact forces inducing bolt shear.
3. Pressure forces inducing tensile bolt stresses.
4. Bolt preload effects.

Maximum axial impact forces are experienced during vertical or near vertical impact. These have been conservatively assessed in Section 2.6.7.1, End Drop Analysis. In Case 2 of the end drop analysis, conservatism is assured by neglecting the external impact limiter forces imposed upon the lid which directly react against the impact associated with bolt tensile stresses. Thus, in this earlier end drop analysis, all impact forces are conservatively reacted against by direct and uniform bolt tension.

Similarly, maximum lateral impact forces are experienced during side impacts and have been assessed in Section 2.6.7.3, Side Drop Analysis. In Section 2.6.7.3, lateral impact forces are assumed to be carried by one or more of three bounding mechanisms.

The first and most probable mechanism assumes frictional forces between the lid and body carry all lateral forces. Bolt preloads are shown to be sufficient to assure a positive margin of safety.

The second lateral force reacting mechanism demonstrates that in the absence of bolt preload forces, all loads are carried in bearing between the stepped lid and the body. Under this mechanism, clearances are such that no lateral loads are imposed via bearing stresses upon the bolts themselves. Specifically, radial clearances between lid and body are 0.006 to 0.010 inches whereas radial clearances between bolts and lids are no less than a factor of three larger.

In the third and least likely lateral force reacting mechanism, friction force resistance capability is assumed degraded (to zero) by internal pressure effects and all lateral forces are shown to be resisted by bolt shear.

Although the consequences of both maximum lateral and longitudinal impact forces are addressed in the analysis of end and side drop, corner and oblique drops pose a somewhat different set of circumstances. Specifically, friction forces between lid and body, associated with bolt preloads, may not exist (or be sufficient) to carry lateral loads where longitudinal impact forces are sufficiently large as to relieve the normal forces between lid and body. In these circumstances, associated only with corner and oblique loads, lateral forces may be imposed on the bolts. Such circumstances are addressed in analyses discussed below.

Comprehensive analysis of all oblique orientations is summarized in Table 2.10.5-2 of Appendix 2.10.5. The minimum margin of safety is found at a near vertical orientation of 85°, with respect to horizontal. At this orientation angle, direct tension stress due to impact, pressure effects, thermal effects, and preload effects is estimated at 57,840 psi for all 32 bolts. The associated shear stress is 113 psi. The resultant Margin of Safety, based on an allowable stress of $(2.0)S_u$, is computed at +0.10.

As described with Appendix 2.10.5, bolt impact stresses are calculated assuming a triangular distribution of bolt forces and external impact forces arbitrarily (and conservatively) applied at the corner of the cask. Thus, external energy absorber forces are not employed to reduce bolt tensile forces. For near vertical impacts, this is the most conservative assumption.

Margin of safety for bolt engagement need not be calculated since it is greater than for the end drop case possessing a margin of +1.22.

(9) Impact Limiter Deformations and Residual Clearances

Maximum impact limiter deformations and residual clearances due to normal conditions of transport corner/oblique drops are directly available from Appendix 2.10.2, Table 2.10.2-17. Per that table, maximum crush of an impact limiter corner is 14.14 inches for the hot condition foam stress case and occurs for a drop angle of 55 degrees with respect to horizontal. Minimum residual clearance for the main cask body is 23.89 inches for the hot condition foam stress case and occurs for a drop angle of 5 degrees with respect to horizontal. These deformations and residual clearances are of little consequence for the NuPac 140-B Cask.

(10) Lead Effects in Oblique Drops

Lead effects in oblique drops are discussed and thoroughly analyzed in Reference (27) for a similarly designed cask. These effects were found to be most significant for impact angles from 60° from horizontal to vertical, or the 90° end drop. This is because cask thrust force significantly decreases at impact angles less than 60°. This is readily apparent in Table 2.10.2-13 (see Appendix 2.10.2) where the thrust force is approximately one-half at an impact angle of 85°.

In Table 2.10.2-13, the thrust force for oblique drops is similar to the force found in Table 2.10.2-11 for the end drop case. Lead effects are thoroughly analyzed for the end drop orientation in Section 2.6.7.1, Cases (6) and (7). The analysis technique suggested in Reference (27) is interpolation of the shell stresses based on the findings for end drop using the ratio of the oblique analysis thrust force to the end drop analysis force. This technique will be employed in detail for normal conditions of transport during final design, and it will be employed for the 85° oblique angle for hypothetical accident condition analysis found in Section 2.7.1.2, Case (10).

For normal conditions of transport during preliminary design, it is assumed that the end and side drop analyses are the bounding design conditions. This may be discerned from a study of Table 2.10.2-8 (see Appendix 2.10.2) where it is shown that the maximum acceleration for any of the oblique drop analyses is 11.1 g's which is significantly less than that for that of end drop at 24.7 g's.

2.6.7.3 Flat Side Drop

Analysis of the NuPac 140-B Cask behavior during the side drop impact is performed in the following steps:

- (1) Analyze the impact force using the SYDROP-PC computer program.
- (2) Analyze the cask outer shell as though it alone carries the loads.
- (3) Analyze the cask inner and outer shells as though they act in parallel to carry the loads.
- (4) Analyze the payload baskets. Apply cask deflections to the payload baskets and evaluate the resulting bending stresses.
- (5) Analyze the cask lid for bearing against the top ring forging of the cask. Analyze the cask closure bolts. Analyze the top ring forging for shear.
- (6) Analyze the cask end plate welds for shear.
- (7) Calculate the side impact clearances for the lifting and tiedown trunnions.
- (8) Address fuel assembly payload behavior.

Material properties and allowable stresses are available respectively from Tables 2.3-1, 2.1-1 and 2.1-2. Material properties and allowables corresponding to maximum enveloping temperature (300 °F) for the outer cask and (300 to 600 °F) for baskets are as follows. Properties and allowables at other temperatures can be readily obtained from Tables 2.3-1, 2.1-1 and 2.1-2. Buckling is addressed in Appendix 2.10.3.

<u>Material Property</u>	<u>Cask</u>	<u>Baskets</u>
Elastic Modulus, E , x 10^6 psi	27.8	25.3 to 27.0
Yield Strength, S_y , ksi	43.4	37.3 to 43.4
Ultimate Strength, S_u , ksi	94.3	87.8 to 94.3
Design Stress, S_n , ksi	31.4	29.2 to 31.4
Allowable Stresses, ksi		
Containment Structures		
Normal Primary Membrane	31.4	29.2 to 31.4
Normal Primary Membrane plus Bending	47.1	43.8 to 47.1
Normal Pure Shear	18.8	17.5 to 18.8
Normal Bearing	43.4	37.3 to 43.4
Non-Containment Structures		
Normal Primary Membrane	43.4	37.3 to 43.4
Normal Primary Membrane plus Bending	47.1	43.8 to 47.1
Normal Pure Shear	26.0	22.4 to 26.0
Normal Bearing	43.4	37.3 to 43.4

Where possible, the analyses are done on the basis of a unit (1 g) lateral acceleration, with actual acceleration numbers being substituted at the final point of calculating stresses.

(1) Side Drop Computer Analysis (SYDROP)

The side drop analysis was performed utilizing the energy-balance computer program, SYDROP-PC, documented in Appendix 2.10.1. Two cases were run which considered the variations in impact limiter foam strength due to temperature effects and the possible variations in foam strength from nominal values (see Section 2.3 and Appendix 2.10.2). The drop cases consider 86% of the impact limiter length available for crush which takes allowance for the soft foam (approximately three pounds per cubic foot foam density) corners extending 9.5 inches back from the outermost corners. Based on scale test results from a similar cask published in Reference (27) (i.e., observed impact limiter displacements and cask accelerations), this assumption is valid. Output for each of the side drop cases is presented in Appendix 2.10.2. The worst case deflections and accelerations obtained from the bounding cases are utilized for all analyses. Maximum resultant acceleration for the normal side drop is 16.3 g's (-20 °F foam) and maximum impact limiter deflection is 3.31 inches (hot foam) per Tables 2.10.2-8 and 2.10.2-17 of Appendix 2.10.2.

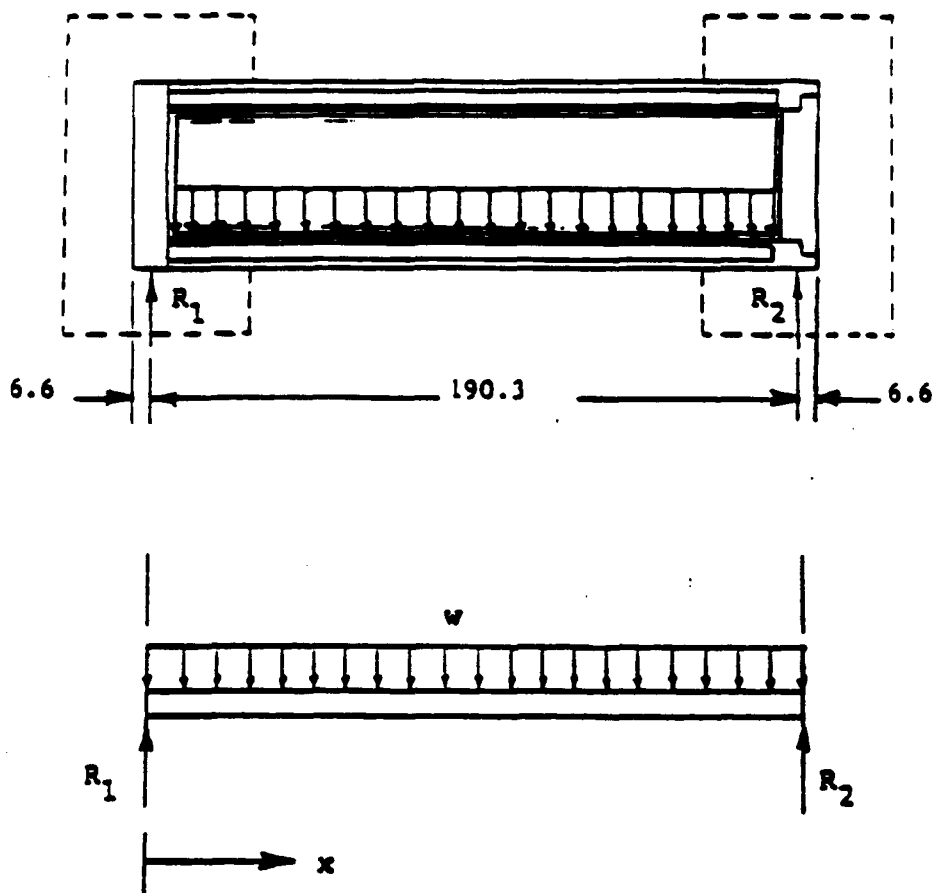
(2) Cask Outer Shell Carries Impact Load

The loads on the cask outer shell can be represented as shown in Figure 2.6.7.3-1. However, the portion of the cask analyzed in detail in this section corresponds only to the 177.0 inch length of the cask shells. As shown, a 13.25 inch length (end plate forging) extends to the left and a 13.25 inch length (upper forging and closure lid) extends to the right. The load distribution on the analyzed portion of the cask (i.e., the shells) is also shown in Figure 2.6.7.3-1. Note that the cask is conservatively modeled as a beam simply supported at the mid-point of the bottom closure plate and top lid. This is considered to be conservative, since the impact limiters will tend to provide bending restraint on side impact.

For additional conservatism, the distributed load, w , is derived as though the lead and neutron shielding, the steel shells, the copper fins, and the fully loaded basket extend the full length between support points at R_1 and R_2 .

$$w = z (W_{pb} + W_{ss} + W_{lf} + W_{co} + W_{cf} + W_{ts} + W_b) / L.$$

Loads on Cask During Side Impact



w = distributed load due to outer cask shells, lead shielding, neutron shielding, copper fins, and fully loaded inner basket

R_1, R_2 = reaction loads

FIGURE 2.6.7.3-1
Cask Load During Side Impact

Where:

W_{pb} = weight of the lead
= 50,037 lbs

W_{ns} = weight of the neutron shield
= 10,835 lbs

W_{is} = weight of the cask inner shell
= 10,541 lbs

W_{os} = weight of the cask outer shell
= 16,338 lbs

W_{cf} = weight of the copper fins
= 9,429 lbs

W_{ts} = weight of the thermal shield and load collars
= 3,810 lbs

W_b = weight of a loaded BWR basket (maximum)
= 48,696 lbs

L_o = cask length between support points
= 190.3 in

Substituting in the proper values yields:

$$w = 530.7 \text{ lbs/in}$$

To determine the cask reaction forces R_1 and R_2 , balance forces on the cask in the lateral direction:

$$\Sigma F = R_1 + R_2 - (530.7)(190.3)$$

$$R_2 = 100,990 - R_1$$

Balancing moments about point x = 190.3:

$$\Sigma M = (190.3)R_1 - (190.3/2)(503.7)(190.3) = 0$$

$$R_1 = 50,496 \text{ lbs}$$

$$R_2 = 50,496 \text{ lbs}$$

A shear-moment diagram for the cask under the defined loading is shown in Figure 2.6.7.3-2. As can be seen, the maximum bending moment of 2,402,352 in-lbs occurs at the cask center for a unit (1 g acceleration) loading. The moment of inertia of the cask outer shell is:

$$\begin{aligned} I_o &= \pi/4(R_{oo}^4 - R_{ii}^4) \\ &= 183,270 \text{ in}^4 \end{aligned}$$

Where:

$$\begin{aligned} R_{oo} &= \text{outer radius of cask outer shell} \\ &= 34.625 \text{ in} \end{aligned}$$

$$\begin{aligned} R_{ii} &= \text{inner radius of cask outer shell} \\ &= 33.125 \text{ in} \end{aligned}$$

The maximum bending stress, σ_b , in the cask shells can now be found as:

$$\sigma_b = Mc/I_o$$

Where:

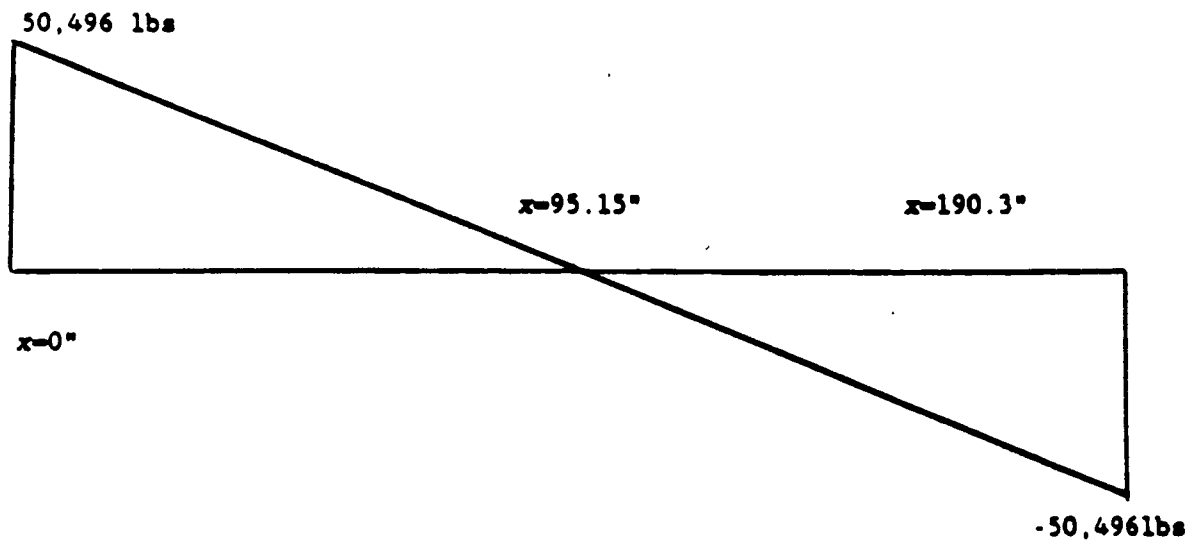
$$c = R_{oo} = 34.625 \text{ in}$$

Then,

$$\begin{aligned}\sigma_b &= (2,402,352) (34.625) / 183,270 \\ &= 453.8 \text{ psi per "g"}\end{aligned}$$

The maximum normal condition side drop acceleration, from Table 2.10.2-8, is 16.3 g's. Thus, the maximum bending stress in the cask outer shell is:

SHEAR DIAGRAM



MOMENT DIAGRAM

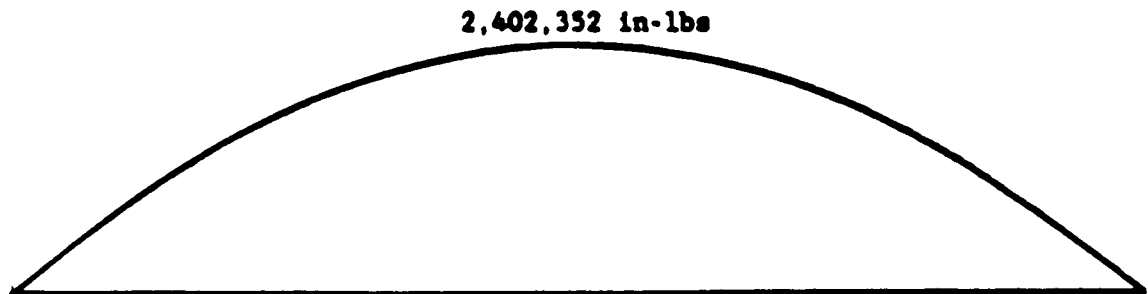


FIGURE 2.6.7.3-2

Cask Shear and Moment Diagrams for Side Drop Loading

Since this stress is nearly uniform through the outer shell thickness, it will be treated as a primary membrane stress. The outer shell is not considered a pressure boundary, so there are no other stresses to be combined. The membrane stress allowable, σ_s , for a non-containment structure under normal operating conditions is $S_y = 31,400$ psi. Thus, the Margin of Safety is:

$$\begin{aligned} \text{M.S.} &= (\sigma_s / \sigma_o) - 1 \\ &= (31,400 / 7,399) - 1 = \underline{+3.24} \end{aligned}$$

(3) Analyze the Cask Inner and Outer Shells

Since the maximum cask outer shell stress is 7,399 psi at the extreme fiber of the outer cask (i.e., the outside surface of the outer cask outer shell), the corresponding inner shell stress may be derived directly by multiplying this stress value by the ratio of the outer and inner shell outside radii:

$$\sigma_i = (R_{io} / R_{oo}) (7,399)$$

Where:

$$R_{io} = 29.625 \text{ in}$$

$$R_{oo} = 34.625 \text{ in}$$

Then,

$$\begin{aligned} \sigma_i &= (29.625 / 34.625) (7,399) \\ &= 6,331 \text{ psi} \end{aligned}$$

The maximum shear stress across the section, τ , may be calculated as:

$$\tau = (2)V / (A_{oo} + A_{io})$$

Where:

$$\begin{aligned} V &= \text{maximum shear force} \\ &= 50,496 \text{ lbs} \end{aligned}$$

$$\begin{aligned} A_o &= \text{outer shell area} \\ &= 319.3 \text{ in}^2 \end{aligned}$$

$$\begin{aligned} A_i &= \text{inner shell area} \\ &= 205.4 \text{ in}^2 \end{aligned}$$

Thus, the maximum shear stress in the inner and outer shells for a unit (1 g) side drop acceleration load is:

$$\begin{aligned} \tau &= (2) (50,496) / (319.3 + 205.4) \\ &= 192.5 \text{ psi (per "g")} \end{aligned}$$

and, for a normal conditions of transport side drop acceleration of 16.3 g's:

$$\tau = (16.3) (192.5) = 3,137 \text{ psi}$$

A detailed buckling analysis for this load combination is presented in Appendix 2.10.3.

These stresses are combined with the design pressure of 284 psi, taken from Section 2.6.1.3.2. As shown in Table 2.6.7.3-1, the resulting Margin of Safety is +1.61.

(4) Analyze Payload Baskets

The payload basket stresses and margins of safety are analyzed and presented in Section 2.6.8. Significant worst case results are summarized in Table 2.6.7.3-2 for the outer shells, and Table 2.6.7.3-3 for the truss core panels. The effects of outer cask deflections on payload basket stresses will be analyzed during the final design phase of the project.

(5) Outer Cask Lid Analysis

The most probable mechanism for transfer of lateral loads between lid and body is by frictional forces induced by bolt preloads. Frictional capability will be briefly assessed. Should this frictional force not develop, load transfer will occur by shearing of the lid bolts.

TABLE 2.6.7.3-1
STRESS SUMMARY: Cask Inner Shell

- Loading Condition: Normal Condition Side Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Containment Boundary
- Stress Allowable (Section 2.1.2.1): S_a
- Material Temperature: 300 °F
- Allowable Stress, S_a : 31,400 psi

	Stress Components (psi)				Reference
	Radial σ_r	Hoop σ_h	Axial σ_a	Shear τ_{rz}	
<u>Load Condition</u>					<u>Section</u>
Internal Pressure	-284	7,304	3,652	0	2.6.1.3.2
Side Drop	0	0	6,331	3,137	2.6.7.3(3)
Direct Stress Sum:	-284	7,304	9,983	3,137	
Principal Stresses:	-1,167	7,304	10,866		
Stress Differences:	12,033	8,471	3,562		
Maximum Stress Intensity (SI):	12,033 psi				
Allowable Stress Intensity, S_a :	31,400 psi				
Margin of Safety ($[S_a/SI] - 1$):	<u>1.61</u>				

TABLE 2.6.7.3-2
STRESS SUMMARY: Basket Outer Shell

- Loading Condition: Normal Condition Side Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Non-Containment Structure
- Stress Allowable (Section 2.1.2.1): S_y
- Material Temperature: 368 °F
- Allowable Stress, S_a : 41,500 psi

	Stress Components (psi)				Reference
	Radial σ_r	Hoop σ_θ	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					<u>Section</u>
Side Drop	0	0	14,800	0	2.6.8.2
Direct Stress Sum:	0	0	14,800	0	
Principal Stresses:	0	0	14,800		
Stress Differences:	14,800	0	0		

Maximum Stress Intensity (SI): 14,800 psi

Allowable Stress Intensity, S_a : 41,500 psi

Margin of Safety ($[S_a/SI] - 1$): +1.80

TABLE 2.6.7.3-3
STRESS SUMMARY: Truss Core Panels

- Loading Condition: Normal Condition Side Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Non-Containment Structure
- Stress Allowable (Section 2.1.2.1): S_y
- Material Temperature: 420 °F
- Allowable Stress, S_a : 40,400 psi

	Stress Components (psi)				Reference
	Radial σ_r	Hoop σ_t	Axial σ_s	Shear τ_{rs}	
<u>Load Condition</u>					<u>Section</u>
Side Drop	0	0	14,400	0	2.6.8.2
Direct Stress Sum:	0	0	14,400	0	
Principal Stresses:	0	0	14,400		
Stress Differences:	14,400	0	0		

Maximum Stress Intensity (SI): 14,400 psi

Allowable Stress Intensity, S_a : 40,400 psi

Margin of Safety ($[S_a/SI] - 1$): 1.81

The capability of the bolted joint to withstand this shearing load is fully assessed in Table 2.10.5-1 and 2.10.5-2 of Appendix 2.10.5. From Table 2.10.5-2, total bolt tensile stress, including pressure and preload effects, is determined to be 57,014 psi. The corresponding shear stress is 1,494 psi. The resultant Margin of Safety with an allowable stress of $(2.0)S_u$ is computed in Table 2.10.5-2 as +0.12.

Returning now to the more likely frictional load transfer mechanism, the normal clamping force can be directly assessed by reference to the relations of Note 16 associated with Tables 2.10.5-1 and 2.10.5-2 of Appendix 2.10.5. The clamping force per bolt, from Table 2.10.5-1, is given there as:

$$\begin{aligned} F_b &= K_b P / (K_b + K_l) - F_{\text{eff}} \\ &= -96,287 \text{ lbs} \quad (\text{minus sign indicates a compressive force}) \end{aligned}$$

Where:

$$\begin{aligned} P &= \text{external load} \quad (\text{see Note 15}) \\ &= 30,807 \text{ lbs} \end{aligned}$$

$$\begin{aligned} F_{\text{eff}} &= \text{bolt preload} \quad (\text{see Note 6}) \\ &= 62,536 \text{ lbs} \end{aligned}$$

$$\begin{aligned} K_b &= \text{bolt stiffness} \quad (\text{see Note 8}) \\ &= 2.05(10)^7 \text{ lbs/in} \end{aligned}$$

$$\begin{aligned} K_l &= \text{lid stiffness} \quad (\text{see Note 9}) \\ &= 2.35(10)^8 \text{ lbs/in} \end{aligned}$$

The frictional shear resistance capacity per bolt is obtained by applying an appropriate coefficient of friction to the predicted clamping force of -96,287 pounds. Reference (36), Page 611, gives the coefficient of friction, μ , between two Type 304 stainless steel surfaces as 0.9. For conservatism, this value is reduced by one-half to 0.45 for capacity evaluation. As shown in Table 2.10.5-1, of Appendix 2.10.5, the applied shearing force per bolt, V_b , is found as 5,081 pounds. Thus, the Margin of Safety against lid slippage is:

$$\begin{aligned} \text{M.S.} &= (\mu F_b / V_b) - 1 \\ &= [(0.45)(96,287) / 5,081] - 1 = \underline{+7.53} \end{aligned}$$

Thus, slippage is clearly not likely to occur and no shear loads are likely to be imposed upon the lid bolts.

In the most unlikely event that the lid displaces laterally to bear upon the body ring forging, lateral loads will be transferred by bearing. The following calculations demonstrate the adequacy of this load transfer mechanism.

The weight of the cask lid (9,974 lbs) may be carried in bearing against the lid end ring forging of the outer cask if sliding occurs at the lid to forging interface. Utilizing a Hertz analysis for bearing between two cylinders, including the normal conditions of transport, 16.3 g's maximum acceleration:

$$\sigma_b = [2P / \pi L (KD) (CE)]^{0.5}$$

Where:

$$\begin{aligned} P &= \text{load between cylinders} \\ &= (9,974)(16.3) = 162,576 \text{ lbs} \end{aligned}$$

$$\begin{aligned} L &= \text{length of contact} \\ &= 1.641 \text{ in (approximately)} \end{aligned}$$

$$\begin{aligned} D_1 &= \text{diameter of outer cylinder (maximum)} \\ &= 59.720 \text{ in} \end{aligned}$$

$$\begin{aligned} D_2 &= \text{diameter of inner cylinder (minimum)} \\ &= 59.700 \text{ in} \end{aligned}$$

$$\begin{aligned} KD &= D_1 D_2 / (D_1 - D_2) \\ &= 178,264 \text{ in} \end{aligned}$$

$$\begin{aligned} CE &= 2(1 - \mu^2)/E \\ &= 6.741(10)^{-8} \text{ in}^2/\text{lb} \end{aligned}$$

$$E = 27.0(10)^6 \text{ psi at } 300^\circ\text{F}$$

$$\mu = 0.30$$

Then,

$$\sigma_b = 2,291 \text{ psi}$$

The normal allowable bearing stress, σ_a , is given as 43,400 psi. Therefore, the outer cask lid bearing stress Margin of Safety is:

$$\begin{aligned} \text{M.S.} &= (\sigma_a/\sigma_b) - 1 \\ &= (43,400/2,291) - 1 = \underline{+17.9} \end{aligned}$$

The lip of the forging which supports the lid is loaded in shear by the lid. Conservatively assuming that only one-half the lip is effective in resisting this unit load (1 g), the shear stress, τ , is:

$$\begin{aligned} \tau &= (9,974)/(\pi[(38.70)^2 - (29.86)^2]/2) \\ &= 10.48 \text{ psi} \end{aligned}$$

For the normal condition side impact acceleration of 16.3 g's, the shear stress is:

$$\tau = (16.3)(10.48) = 171 \text{ psi}$$

The normal allowable pure shear stress, τ_a , is given as 18,800 psi. Therefore, the outer cask lid lip shear stress Margin of Safety is:

$$\begin{aligned} \text{M.S.} &= (\tau_a/\tau) - 1 \\ &= (18,800/171) - 1 = \underline{+108.9} \end{aligned}$$

(5) Welds Joining the Cask Shells to the End Closure Plate

The worst case shear loads at the joints of the cask shells to the end closure plate are found to be as follows:

- (a) For the cask inner shell, the shear load, V_{i1} , is taken as the shear load determined previously in this section from Figure 2.6.7.3-2.

$$V_{i1} = 50,496 \text{ lbs}$$

- (b) For the cask outer shell, the shear load, V_{o1} , is taken as the same shear force shown above.

$$V_{o1} = 50,496 \text{ lbs}$$

Since the welds are full penetration bevel welds, the effective weld shear area, A_w , for each shell may be found by the following equation: (i.e., use half the total weld area per Table 17, Case 18, of Reference (15)).

$$A_w = 2\pi R t / 2$$

Where:

R = mean radius, in

t = shell thickness, in

The stress in the welds is:

$$\tau_w = V / A_w$$

Applying a unit (1 g) acceleration, results from the above equations are summarized in the table below:

<u>Shell Weld</u>	<u>R (in)</u>	<u>t (in)</u>	<u>A_w (in²)</u>	<u>t_w (psi)</u>
Inner	29.063	1.125	102.7	492
Outer	33.875	1.500	159.6	316

Applying 16.3 g's to the outer cask inner and outer shell end closure weld stresses results in:

$$\tau_{wi} = (16.3)(492) = 8,020 \text{ psi}$$

$$\tau_{wo} = (16.3)(316) = 5,151 \text{ psi}$$

Table 2.6.7.3-4 presents stress and Margin of Safety results for the more critical inner shell weld under worst case load combinations. A zero internal pressure is used in order to maximize inner shell compressive stresses.

(7) Calculate Side Impact Clearances for the Lifting and Tiedown Trunnions

From Table 2.10.2-17 of Appendix 2.10.2, the maximum impact limiter deflection is 3.31 inches. Impact limiter diameter is 128.0 inches, while the outer cask outside diameter is 77.4 inches. The tiedown trunnions extend 5.5 inches from the cask outer surface. Therefore, the minimum clearance, χ , between the impact limiter crush plane and the trunnions is:

$$\chi = [(128.0/2) - 3.31] - [(77.4/2) + 5.5] = 16.49 \text{ in}$$

(8) Address Fuel Assembly Payload Behavior

Considerations of the payload fuel assemblies for the normal condition side drop loading will be discussed in Section 2.9.

2.6.8 Payload Baskets

Both the PWR and the BWR baskets have been analyzed for normal conditions of transport in sufficient detail to assure that they will have adequate margins of safety for transportation and handling loads. This section summarizes the normal conditions of transport and handling loads to allow an overall evaluation of the baskets.

The minimum Margins of Safety for the normal conditions of transport, one foot drop, for the two baskets are:

PWR Basket, M.S. = +1.80 (Side Drop, see Table 2.6.8-1)

BWR Basket, M.S. = +1.26 (Side Drop, see Table 2.6.8-3)

The following subsections describe the analyses which determined these margins.

TABLE 2.6.7.3-4
STRESS SUMMARY: Cask Inner Shell End Welds

- Loading Condition: Normal Condition Side Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Containment Boundary
- Stress Allowable (Section 2.1.2.1): S_a
- Material Temperature: 300 °F
- Allowable Stress, S_a : 31,400 psi

	Stress Components (psi)				Reference
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					<u>Section</u>
Fabrication	0	-3,099	-3,174	0	2.6.2
Side Drop	0	0	-6,331	8,020	2.6.7.3(6)
Direct Stress Sum:	0	-3,099	-9,505	8,020	
Principal Stresses:	0	2,334	-14,938		
Stress Differences:	17,272	14,938	2,334		

Maximum Stress Intensity (SI): 17,272 psi

Allowable Stress Intensity, S_a : 31,400 psi

Margin of Safety ($[S_a/SI] - 1$): +0.82

2.6.8.1 PWR Basket Analyses

The capacity of the PWR basket is 21 fuel elements. With the exception of borated copper sheets, all the PWR basket structural elements are Type XM-19 austenitic stainless steel. The borated copper sheets and the 0.030 inch thick tube liner sheets are considered as non-structural elements, but the weight of all components of the basket are considered for purposes of analysis.

The following list describes both the primary structural and non-structural elements of the basket. Figure 2.6.8-1 depicts the structural elements of the PWR basket in cross-section. Later in this section, stresses, temperatures, and margins of safety are presented in tabular form in the same order as in the list:

- (1) The 57.0 inch outer diameter basket outer shell is constructed of 0.25 inch thick, Type XM-19, austenitic stainless steel.
- (2) The jointed, truss core panels are constructed with a 0.59 inch wide water gap separated by stainless/copper/stainless laminated sheets. The laminated sheets consist of 0.165 inch thick borated copper between an inner facing, 0.060 inch thick, and an outer facing (non-structural), 0.030 inch thick, Type XM-19, austenitic stainless steel sheets. A diagonal core web 0.035 inch thick located in the water gap structurally connects the two laminated sheets.

The panels are approximately 180 inches long, with a "dovetail" joint running the full length at each side. There does not appear to be any structural requirement for full length panels, and short abutting panels may be considered in final design.

- (3) Welded truss core panels are located towards the outer radius of the PWR basket. They are identical to the jointed panels with the exception that along one side, the panels have a reinforced fitting which is welded to the axially spaced gussets and basket outer shell. These welded joints provide added stability to the basket, especially for any out-of-cask handling, while the inner jointed connections allow sufficient flexibility for the basket to partially conform to the shape of the cask during loadings due to normal conditions of transport.

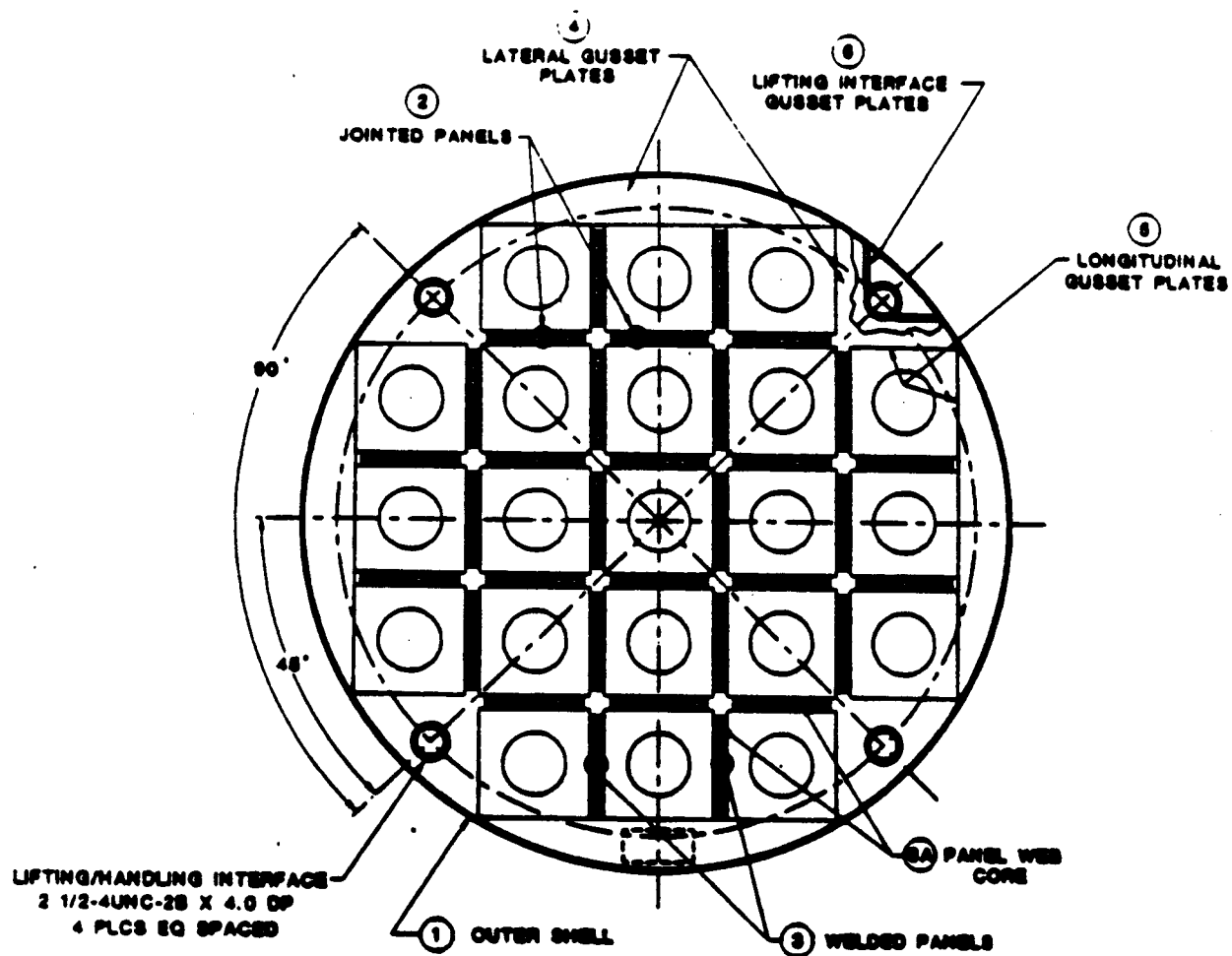
**PWR BASKET STRUCTURAL COMPONENTS**

FIGURE 2.6.8-1

PWR Basket Structural Components

- (4) Lateral gusset plates are welded to the basket outer shell, and lie in planes perpendicular to the longitudinal (axial) axis of the basket. The gussets are spaced 12 inches apart along the length of the basket and are constructed of 0.25 inch thick, Type XM-19, austenitic stainless steel.
- (5) Longitudinal gusset plates are welded to the basket outer shell and lateral gussets, and run parallel to the longitudinal (axial) axis of the basket. These weldments provide stiffness to the basket, and attachment locations for the welded truss core panels.

The weldments also result in fluid-tight compartments which reduce the weight of water which must be lifted when the NuPac 140-B Cask is removed from the fuel pool during loading operations.

The normal conditions of transport drop loads induce the highest stresses in the basket. From the side and end drop analyses, for the impact limiters in their stiffest condition (i.e., -20 °F polyurethane foam), the following design accelerations were obtained:

- Side drop (1 ft): 16.3 g's
- End drop (1 ft): 24.7 G's

All stress calculations use maximum acceleration combined with maximum payload weight and degraded (hot) material properties resulting in worst case conditions.

2.6.8.1.1 PWR Basket Side Drop Analysis

Due to the multiple load paths and the sensitivity of results to the flexibility of the structure, a finite element approach was utilized to analyze the PWR basket in the side drop orientation.

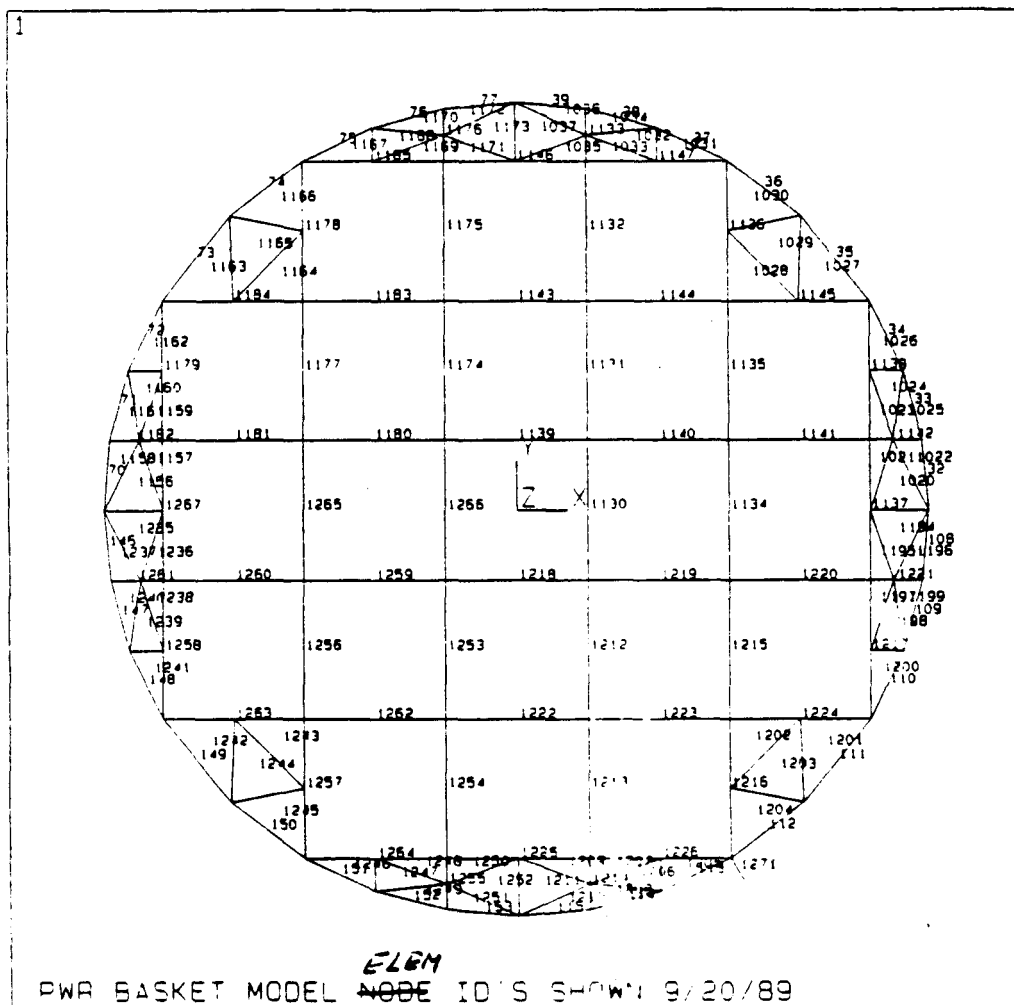
The structural response was especially sensitive to the representation of boundary conditions. The interface between the basket and inner diameter of the cask results in non-linear support for sideways accelerations. Initially, with self-weight (1 "g" unit acceleration), the outside diameter of the basket makes a line contact with the inside diameter of the cask. For higher loads, the basket cross-section becomes oval and contacts the cask over a wider area as it conforms to the inside diameter of the cask. For purposes of these analyses, the cask was assumed to maintain a circular cross-section.

Figures 2.6.8-2 and 2.6.8-3 delineate element and node geometry plots for the ANSYS finite element model used for side load analysis. The large outer triangles represent a finite element mesh of the lateral gussets. The basket outer shell elements are modeled using 0.25 inch thick shell elements. The inner square grid is modeled using beam elements (intersecting the outer gusset elements) with tension/compression and bending capabilities, and pin-ended spar elements with tension/compression capabilities only for all interior grid members not intersecting the outer gusset elements. Element 1271 (in the lower, right-hand quadrant in Figure 2.6.8-2) is a soft spring element introduced to give a small amount of extra stability in some models to speed up the convergence of the iterative solution.

The finite element model uses non-linear, interface (gap) elements between the basket outside diameter and cask inside diameter to model the varying interface contact as a function of the applied force. The location of the elements is depicted in Figure 2.6.8-2 by the pairs of triangles along the outside diameter of the basket. The triangles represent translational fixity at the grounded end of the gap elements. The superimposed triangles at Node 253 signify full fixity for all degrees of freedom (both translational and rotational) as it is the point of initial contact between the basket and cask. Initially, the interface elements are open, from zero at the initial point of contact (Node 253), and increasing to 0.150 inches at about $\pm 90^\circ$ from that point (Nodes 166 and 366). The interface elements are arranged radially and are free to slip in the tangential direction thereby representing a frictionless interface. This is a conservative modeling assumption, as friction tends to reduce the loads and stresses in the basket.

As the load is applied incrementally, the gaps begin to close and additional reaction forces appear. These additional reactions distribute the load, so that as loads increase, the stresses and deflections increase at a lesser rate. However, the total resultant reaction force is directly proportional to load.

As the load is applied incrementally, the gaps begin to close and additional reaction forces appear. These additional reactions distribute the load, so that as loads increase, the stresses and deflections increase at a lesser rate. However, the total resultant reaction force is directly proportional to load.



ANSYS 4
SEP 24
17:26:46
PLOT NO.
ELEMENTS

ZV = 1
DIST = 35

FIGURE 2.6.8-3

PWR Basket Element Geometry Plot

The total of the reaction forces will always equal the product of the weight of the basket and the acceleration and is used as a check for finite element analysis modeling accuracy.

An additional assumption made in the model is that the truss core dovetail joints are frictionless. Friction at the joints will tend to give additional support to the basket, and will in general reduce stresses at the most highly stressed points.

The maximum payload weight of 32,200 pounds (21 fuel elements, weight 1,533 pounds each) is distributed over the full 180 inch length of the basket. This weight is assumed to be distributed over the structural panels and ignores the contributing strength and stiffness of the fuel elements themselves. The weight, distributed over the panel width of 8.75 inches, induces greater bending in the panels than would an actual fuel element which would load the panels mostly near their edges.

The side drop angle defines the angle between the acceleration vector and the horizontal panel direction in the cross-section view. For the PWR basket analysis, three (3) angles were chosen:

- (1) 90°, with contact between the basket and cask initially at 90° from the horizontal; initial contact at Nodes 269 and 369,
- (2) 45°, with contact between the basket and cask initially at 45° from the horizontal; initial contact at Nodes 235 and 253, and
- (3) 59°, with contact between the basket and cask initially at 59° from the horizontal; initial contact at Node 253. This angle was chosen as it puts the initial point of contact, and the resultant load vector, through the weakest part of the lateral gussets.

Appendix 2.10.4 presents the data files of input and output data for the baskets.

An additional side loading analysis of 1.4 g's applied at an angle 59° from horizontal for an "out-of-cask" condition was performed. Figure 2.6.8-4 illustrates the displaced shape of a basket superimposed on the original shape and Figure 2.6.8-5 shows the stress intensity in the lateral gusset plates when only Node 253 is used for support (to react against the load). Stresses for other components cannot be clearly shown on these plots. The results for all loading conditions for the PWR side drop finite element analyses are summarized in the computer print out of Appendix 2.10.4.

The results of the analyses for normal conditions of transport side drop loads are given in Table 2.6.8-1. The normal conditions of transport allowable stresses for non-containment components is provided in Table 2.1-2. The primary membrane allowable stress intensity is the greater of S_x or S_y . By definition, since $S_x = (2/3)S_y$, the allowable stress intensity, $S_x = S_y$. The primary membrane plus bending allowable stress intensity is the greater of $(1.5)S_x$ or S_y . Again, by definition, since $S_x = (2/3)S_y$, the allowable stress intensity, $S_x = S_y$.

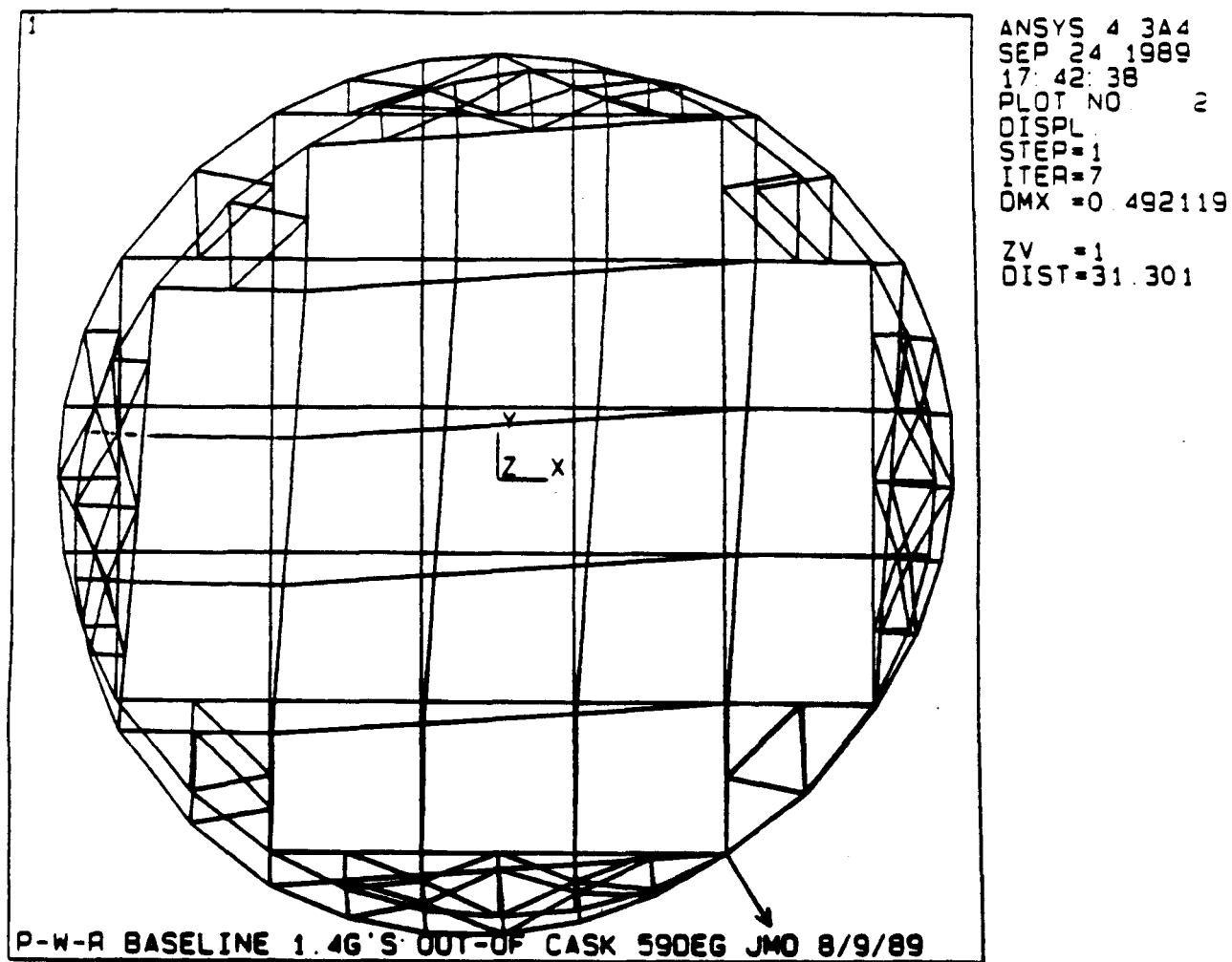


FIGURE 2.6.8-4
PWR Basket Displacement Plot (Over Non-Displaced Shape)

TABLE 2.6.8-1
PWR Basket Side Drop Analysis Stress Summary

(Item)		Maximum	Element	Allowable	
Component	Element	Temperature	Stress	Stress	Margin
<u>Description</u>	<u>Number</u>	<u>(°F)</u>	<u>(psi)</u>	<u>(psi)</u>	<u>of Safety</u>
(1) Outer Shell	114	368	14,800	41,500	1.80
(2) Jointed (Pinned) Panels	1212	580	5,300	37,800	8.09
(3) Welded (Fixed) Panels	1220	420	-14,800	40,400	1.81
(4) Lateral Gussets	251	420	9,000	40,400	3.49
(5) Longitudinal Gussets	1214	420	-13,700	40,400	1.95

Table 2.6.8-1 presents the Margins of Safety for the various components of the PWR basket based on material yield strength, S_y , as the allowable stress intensity. The exception is in the web core where the minimum margins of safety are based on local buckling of the thin sheets which make up the diagonal core.

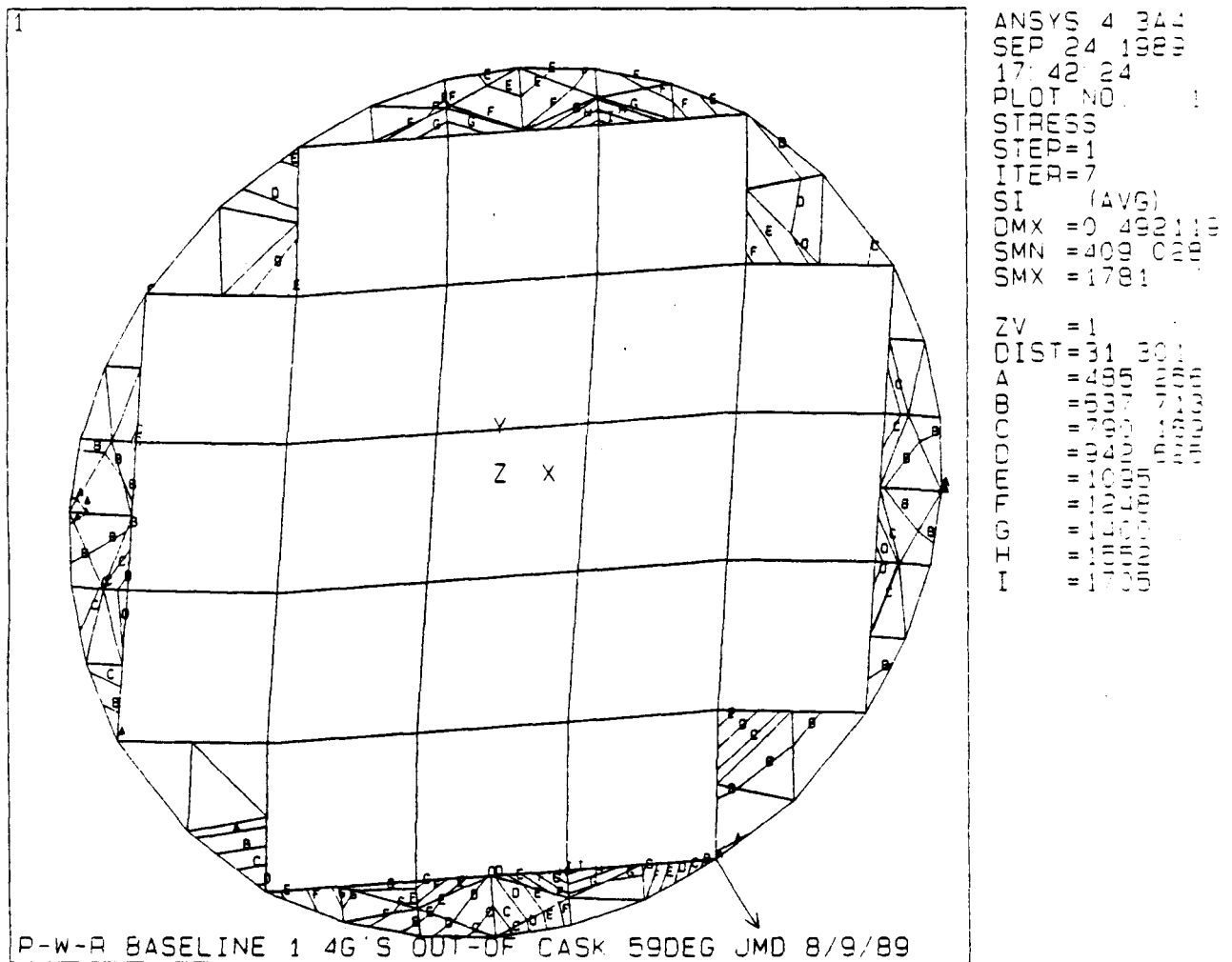


FIGURE 2.6.8-5
 PWR Basket Stress Intensity Contour Plot (Lateral Gussets)

From Table 2.6.8-1, the minimum Margin of Safety for the PWR basket for the normal condition side drop occurs in Item 1, the outer shell.

$$\begin{aligned} \text{M.S.} &= (S_e/S_{\max}) - 1 \\ &= (41,500/14,800) - 1 = \underline{+1.80} \end{aligned}$$

2.6.8.1.2 PWR Basket End Drop Analysis

The maximum acceleration for the normal conditions of transport end drop orientation is 24.7 g's.

For the basket shell alone, neglecting internal stiffening except as it prevents localized buckling, the critical buckling stress in a cylindrical tube is found using Table 35, Case 15, from Reference (15):

$$\begin{aligned} \sigma_{cr} &= (0.3)Et/r \\ &= (0.3)[26.66(10)^6](0.25/28.5) = 70,158 \text{ psi} \end{aligned}$$

Where:

$$\begin{aligned} E &= 26.66(10)^6 \text{ psi at } 368^\circ\text{F} \\ t &= 0.25 \text{ in} \\ R &= 28.5 \text{ in} \end{aligned}$$

This is conservative because the equation treats the load as uniform while the actual acceleration loads linearly decrease from the base to the top of the basket outer shell.

The compressive stress in the basket outer shell may be determined by simply taking the product of the density of steel, $\rho_s = 0.283 \text{ lbs/in}^3$, and the height of the basket, $h = 180 \text{ inches}$. Thus, the compressive stress at the end of the basket outer shell, σ_s , is 50.94 psi/g, or 1,258 psi for a normal condition end drop load of 24.7 g's. The resulting Margin of Safety against basket outer shell buckling is:

$$\begin{aligned} \text{M.S.} &= (\sigma_{cr}/\sigma_s) - 1 \\ &= (70,158/1,258) - 1 = \underline{+54.8} \end{aligned}$$

For a typical truss core panel skin, neglecting support from the web construction and assuming the 180 inch long panel is simply supported on all sides, the critical buckling stress is found using Table 35, Case 1a, from Reference (15):

$$\begin{aligned}\sigma_{cr} &= K[E/(1 - \nu^2)(t/b)^2] \\ &= (3.23) \{ [25.4(10)^6] / [1 - (0.3)^2] \} (0.06/8.75)^2 = 4,239 \text{ psi}\end{aligned}$$

Where:

$$\begin{aligned}K &= 3.23 \quad (\text{conservatively use the minimum "K" value}) \\ b &= 8.75 \text{ in} \\ E &= 25.4(10)^6 \text{ psi at } 580^\circ\text{F} \\ t &= 0.06 \text{ in}\end{aligned}$$

The compressive stress, σ_c , in the truss core panel is the same as the basket outer shell, or 1,258 psi for a normal condition end drop load of 24.7 g's. The resulting Margin of Safety against truss core panel buckling is:

$$\begin{aligned}\text{M.S.} &= (\sigma_{cr}/\sigma_c) - 1 \\ &= (4,239/1,258) - 1 = \underline{+2.37} \quad (\text{minimum end drop Margin of Safety})\end{aligned}$$

As before, this margin is conservative as the load is assumed to be constant over the entire length. Also, the allowable critical stress of the stiffened panel is much greater than for the thin skin of the panel alone. The effect of end support at the lower end is not significant as the effect becomes small for relatively long panels.

2.6.8.1.3 PWR Basket Handling Loads Analysis

In addition to the basket drop analyses, the case of out-of-cask handling was considered. Analysis was performed on a loaded PWR basket oriented horizontally on a flat, rigid surface. With a saddle support conforming to the diameter of the PWR basket, the loads would be similar to the in-cask environment.

Table 2.6.8-2 presents the stresses for the out-of-cask conditions with an acceleration load of 1.4 g's. The load factor of 1.4 g's includes basket self-weight with an assumed 40% factor for handling. By circumferentially orienting the basket 45° from horizontal (all the load reacted at Node 266), the basket

stresses exceed the normal condition side drop results. This is due to the severe local loading induced by the out-of-cask support on a flat and rigid surface whereas the in-cask loads re-distribute over a finite arc.

TABLE 2.6.8-2
PWR Basket Out-of-Cask Handling Analysis Stress Summary

(Item)		Maximum	Element	Allowable	
Component	Element	Temperature	Stress	Stress	Margin
<u>Description</u>	<u>Number</u>	<u>(°F)</u>	<u>(psi)</u>	<u>(psi)</u>	<u>of Safety</u>
(1) Outer Shell	113	368	-21,300	41,500	0.95
(2) Jointed (Pinned) Panels	1262	580	-1,000	37,600	Large
(3) Welded (Fixed) Panels	1213	420	15,300	40,400	1.64
(4) Lateral Gussets	243	420	2,800	40,400	Large
(5) Longitudinal Gussets	1221	420	-22,500	40,400	0.80

From Table 2.6.8-2, the minimum Margin of Safety for the PWR basket for the out-of-cask loading condition occurs in Item 5, the longitudinal gusset.

$$\begin{aligned}
 \text{M.S.} &= (S_e/S_{\max}) - 1 \\
 &= (40,400/22,500) - 1 = \underline{+0.80}
 \end{aligned}$$

2.6.8.2 BWR Basket Analysis

The above discussions for the PWR basket apply to the similar BWR basket. However, the panels of the BWR basket are inherently stronger, as the structural skin of the panel is much thicker, and there are more interconnecting elements for approximately the same payload weight. In addition, the smaller width of the panels also aids in bridging the distance between joints. Buckling is also much less of a factor, due to the narrower and thicker elements and the absence of the diagonal thin webs eliminates an element which is limited by buckling stresses.

One major difference between the PWR and BWR baskets is at the panel to gusset plate connection. Where the PWR basket utilizes welded connections, the BWR basket maintains flexibility by continuing to use the pin-ended panels at the gusset plate connection. Note that this represents the current point of development of the BWR basket during preliminary design, and in final design and analysis the structural design may vary.

The BWR basket cross-section is illustrated in Figure 2.6.8-6, and the corresponding nodal plot for the ANSYS finite element model is presented Figure 2.6.8-7.

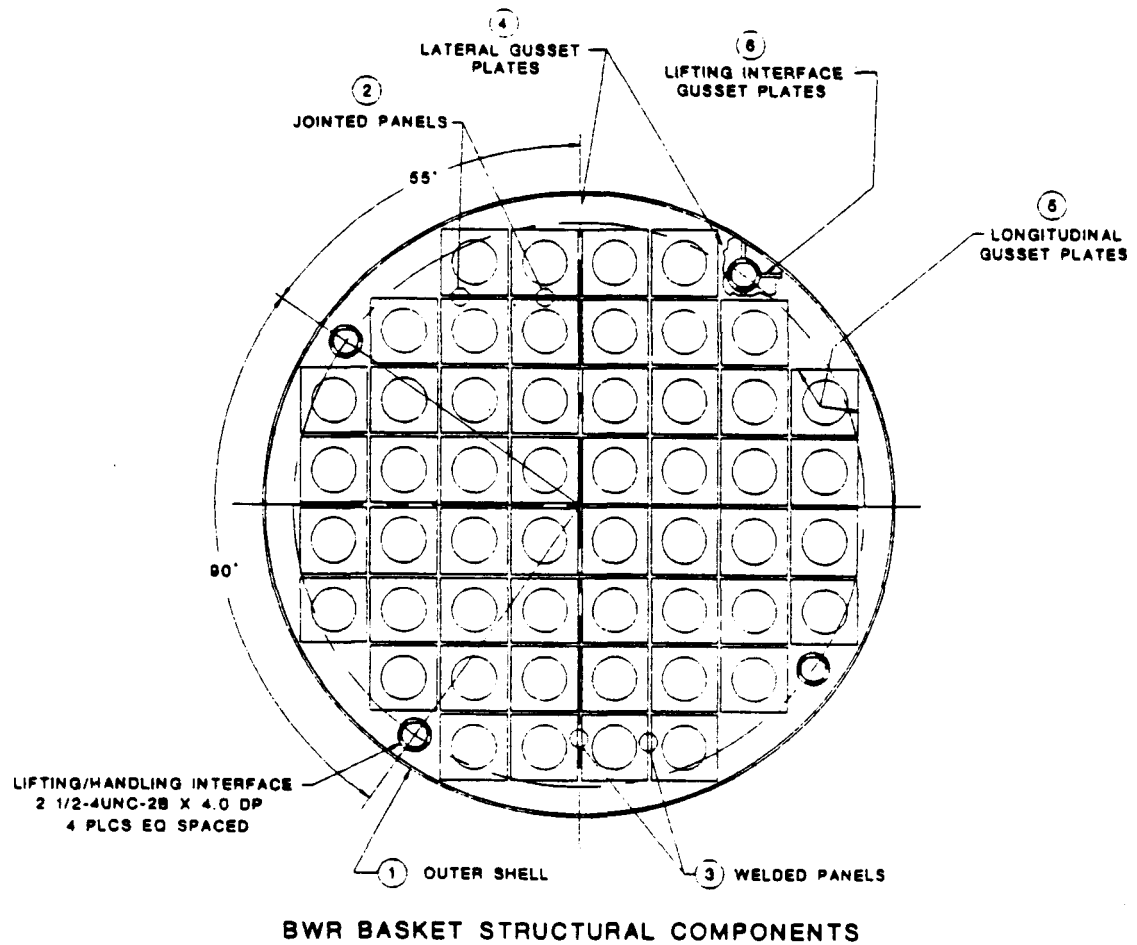


FIGURE 2.6.8-6
BWR Basket Element Geometry Plot

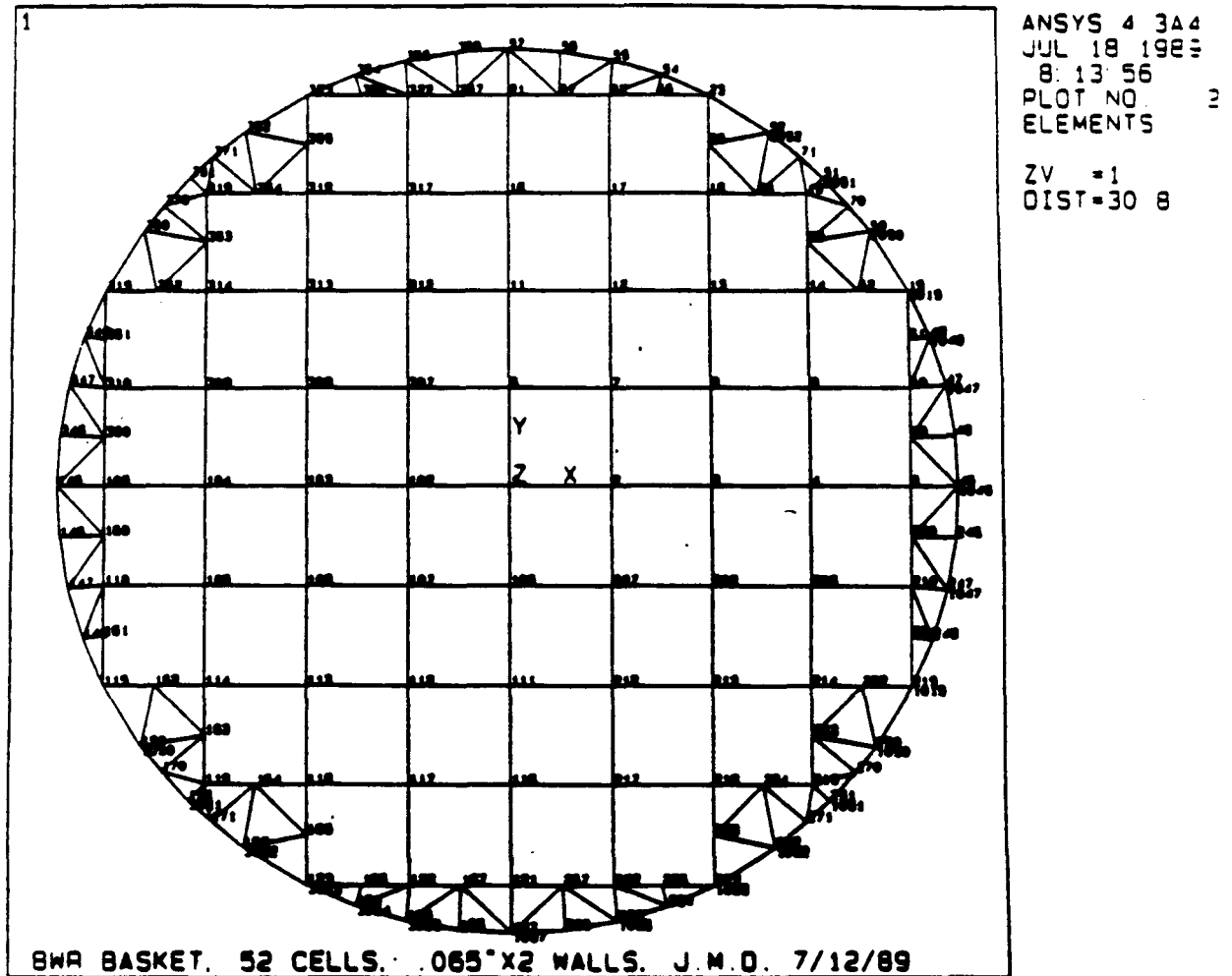


FIGURE 2.6.8-7
BWR Basket Node Geometry Plot

2.6.8.2.1 BWR Basket Side Drop Analysis

All finite element modeling and methodologies utilized for the PWR basket side drop analyses apply to the BWR basket side drop analyses.

Figure 2.6.8-8 illustrates the displaced shape of a basket superimposed on the original shape for a normal conditions of transport drop (16.3 g's) at an angle 45° from horizontal. Figure 2.6.8-8 also illustrates the relative magnitude and direction of the reaction forces using arrows. Figure 2.6.8-9 presents the stress intensity in the lateral gusset plates, also for the normal condition drop of 16.3 g's. As before, stresses for other components cannot be clearly shown on these plots. The results for all loading conditions for the BWR side drop finite element analyses are summarized in the computer print out of Appendix 2.10.4.

The results of the analyses for normal conditions of transport side drop loads are given in Table 2.6.8-3. The normal conditions of transport allowable stresses for the BWR basket are the same as for the PWR basket, $S_x = S_y$.

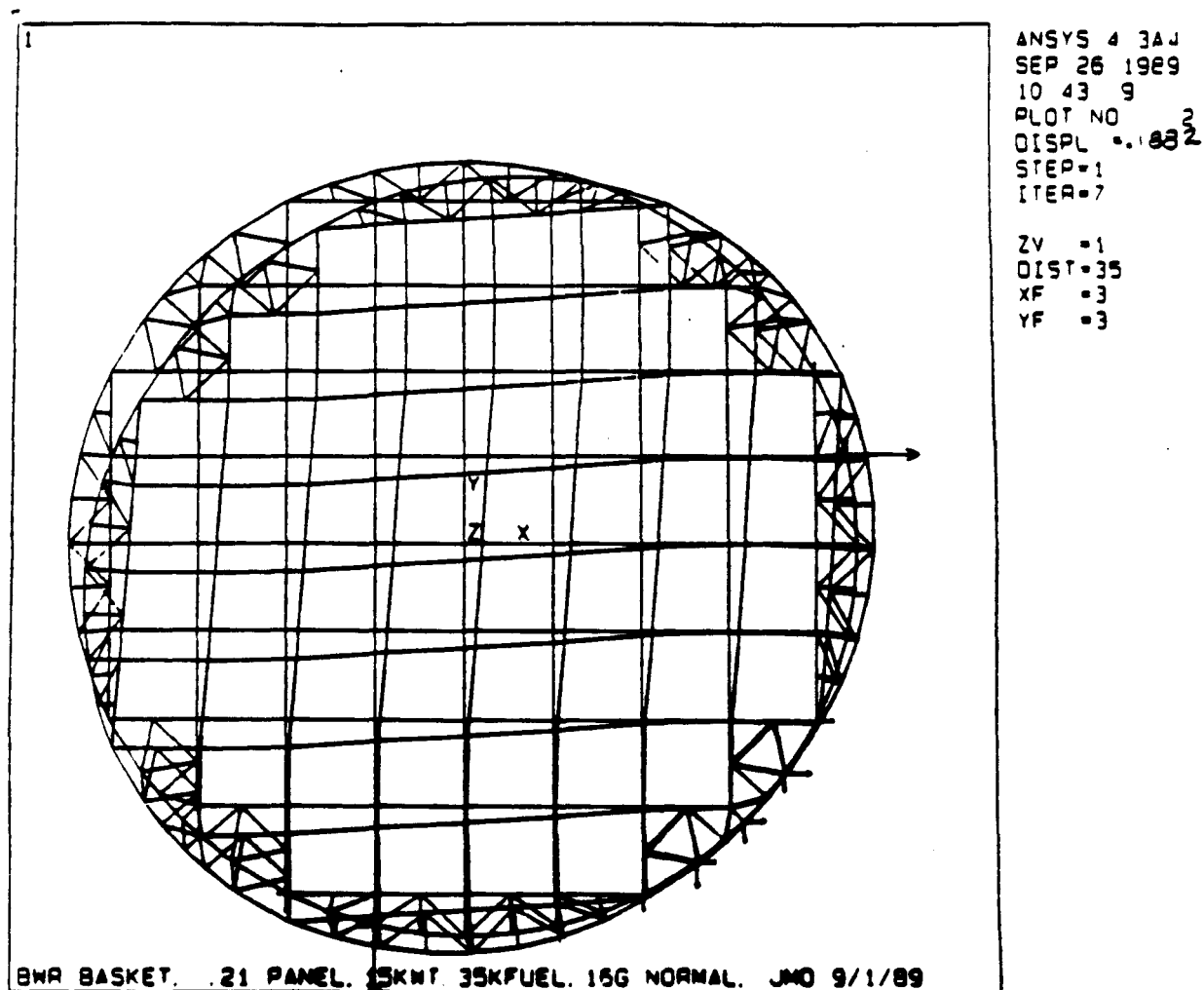


FIGURE 2.6.8-8

BWR Basket Displacement Plot (Over Non-Displaced Shape)

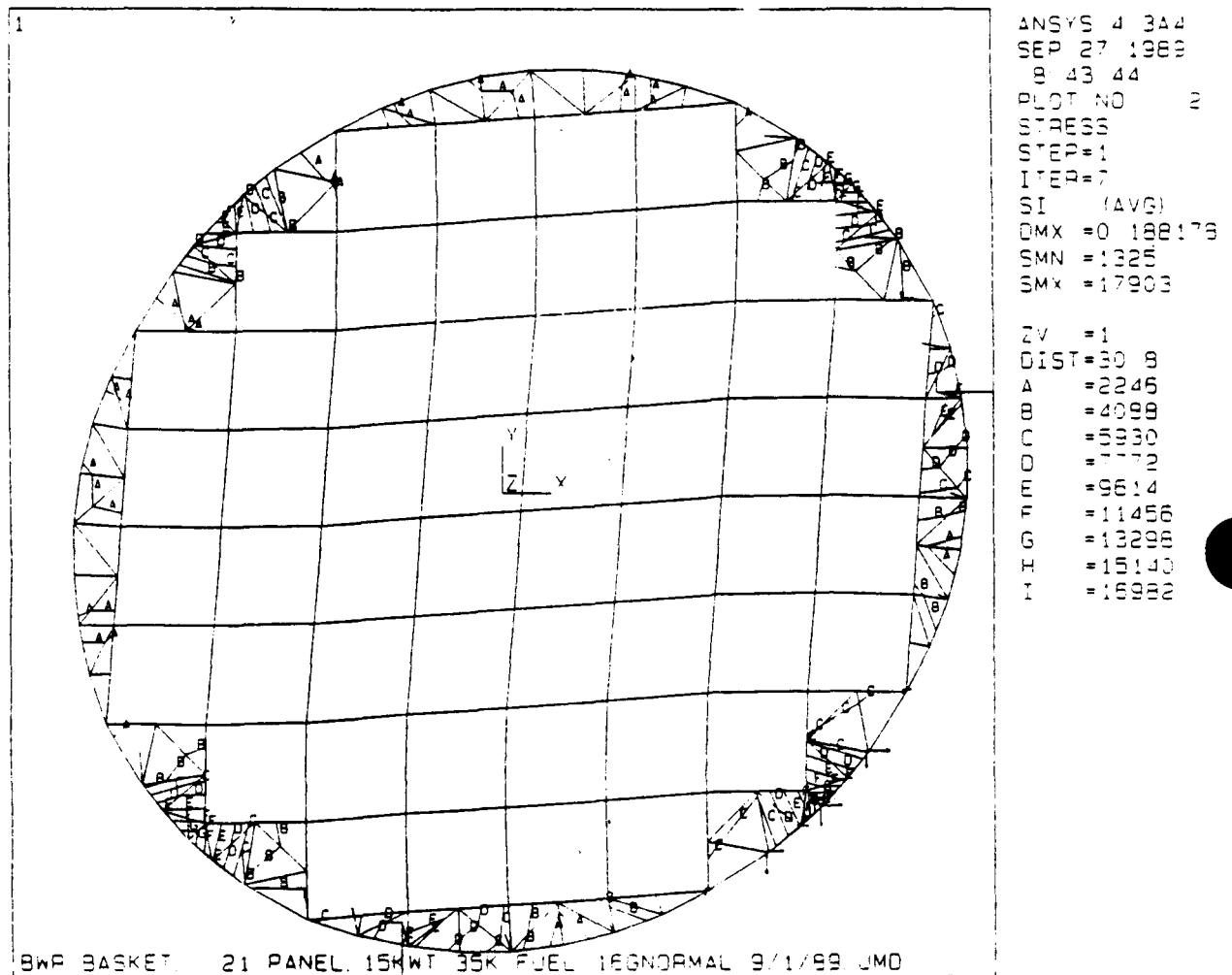


FIGURE 2.6.8-9

BWR Basket Stress Intensity Contour Plot (Lateral Gussets)

TABLE 2.6.8-3
BWR Basket Side Drop Analysis Stress Summary

(Item)		Maximum	Element	Allowable	
Component	Element	Temperature	Stress	Stress	Margin
<u>Description</u>	<u>Number</u>	<u>(°F)</u>	<u>(psi)</u>	<u>(psi)</u>	<u>of Safety</u>
(1) Outer Shell	38	368	-11,180	41,500	2.70
(2) Jointed (Pinned) Panels	8	580	-11,900	37,600	2.16
(3) Welded (Fixed) Panels	111	420	-7,870	40,400	4.13
(4) Lateral Gussets	51	420	17,900	40,400	1.26
(5) Longitudinal Gussets	48	420	-6,130	40,400	5.59

Table 2.6.8-3 presents the Margins of Safety for the various components of the BWR basket based on material yield strength, S_y , as the allowable stress intensity. From Table 2.6.8-3, the minimum Margin of Safety for the BWR basket for the normal condition drop occurs in Item 4, the lateral gusset.

$$\begin{aligned}
 \text{M.S.} &= (S_a/S_{max}) - 1 \\
 &= (40,400/17,900) - 1 = \underline{+1.26}
 \end{aligned}$$

2.6.8.2.2 BWR Basket End Drop Analysis

The end drop analysis for the BWR basket is the same as for the PWR basket, except that in place of the thin 0.060 structural panel skins, the BWR panel structure is a single 0.21 inch thick member. The BWR basket outer shell

thickness is the same as the PWR basket outer shell thickness. Therefore, the Margin of Safety is identical.

Using the same formula as for buckling of the PWR panel skin as in Section 2.6.8.1.2, the critical stress is found to be:

$$\begin{aligned}\sigma_{cr} &= K[E/(1 - \nu^2)(t/b)^2] \\ &= (3.23) \{ [25.4(10)^6] / [1 - (0.3)^2] \} (0.21/5.83)^2 = 116,976 \text{ psi}\end{aligned}$$

Where:

$$\begin{aligned}K &= 3.23 \quad (\text{conservatively use the minimum "K" value}) \\ b &= 5.83 \text{ in} \\ E &= 25.4(10)^6 \text{ psi at } 580^\circ\text{F} \\ t &= 0.21 \text{ in}\end{aligned}$$

The compressive stress, σ_c , in the panel is the same as the PWR basket, or 1,258 psi for a normal condition end drop load of 24.7 g's. The resulting Margin of Safety against panel buckling is:

$$\begin{aligned}\text{M.S.} &= (\sigma_{cr}/\sigma_c) - 1 \\ &= (116,976/1,258) - 1 = \underline{+92.0}\end{aligned}$$

2.6.8.2.3 BWR Basket Handling Loads Analysis

The considerations for out-of-cask handling presented above for the PWR basket apply for the BWR also, and are not repeated here.

(Blank Page)

2.7 Hypothetical Accident Conditions

The NuPac 140-B Cask, when subjected to hypothetical accident conditions as specified in 10 CFR 71.73, meets the performance requirements specified in Subpart E of 10 CFR 71. This is demonstrated in the following subsections where each accident condition is addressed and shown to meet the applicable design criteria (except buckling) previously discussed in Section 2.1.2. Buckling evaluations for all buckling sensitive cask components are presented in Section 2.10.3. The detailed structural evaluation of the payload baskets is included as Section 2.7.7.

2.7.1 Free Drop

Subpart F of 10 CFR 71 requires that a 30 foot free drop be considered for the NuPac 140-B Cask. The drop is to be onto a flat, essentially unyielding, horizontal surface, and the cask is to strike the surface in a position for which maximum damage is expected. Per 10 CFR 71.73(b) the initial temperature for the drop is to be the worst case constant ambient air temperature between -20°F and 100°F. Internal heat generation from the payload and insolation are also considered when it is conservative to do so. (Note: 10 CFR 71 does not require consideration of insolation as an initial condition for accident conditions). Regarding initial internal pressure, the maximum normal operating pressure of 351 psig must be considered unless a lower internal pressure consistent with the ambient temperature assumed to precede and follow the drop is more unfavorable.

The analyses in this section extract accelerations from Appendix 2.10.2 and statically apply them to the package. Static application is justified since the natural frequencies of the cask are relatively high and the duration of the impact loadings relatively long. The cask frequencies are high as the result of using relatively thick, stiff shells for containment.

The durations of impact loadings are relatively long as the result of using soft (relative to the steel structures) energy absorbing impact limiters to protect the cask during free drop events. In addition, inspection of accelerometer data available in Reference (27) from 1/4 scale drop tests of a similar cask indicates that the cask responds essentially as a rigid body (e.g., for end and oblique drops, all accelerometer traces are very nearly the same, and for side drop, the accelerations for all accelerometers oriented in the direction of the drop are nearly identical). This observation further justifies static application of g loads.

2.7.1.1 Flat End Drop

Analysis of the NuPac 140-B Cask behavior during the end drop impact is performed in the following steps:

- (1) Analyze the impact force using the EYDROP-PC computer program.
- (2) Analyze the cask lid for bending assuming the inner basket with payload act as a uniform load upon a simply supported plate (no impact limiter support).
- (3) Analyze the cask lid for bending, as performed in Step 2, but with the impact limiter foam pressure acting in the opposite direction (impact limiter support).
- (4) Analyze the stresses in the cask lid bolts.
- (5) Analyze the payload basket stresses.
- (6) Analyze the axial and hoop stresses in the cask shells and lead (maximum fabrication stress condition assumed).
- (7) Analyze the axial and hoop stresses in the cask shells and lead (zero fabrication stress condition assumed).

- (8) Determine the amount of lead slump.
- (9) Address fuel assembly payload behavior.

Material properties and allowable stresses are available respectively from Tables 2.3-1, 2.1-1 and 2.1-2. Material properties and allowables corresponding to a maximum enveloping temperature (250°F) for the cask and (300 to 600°F) for baskets are as follows. Properties and allowables at other temperatures can be readily obtained from Tables 2.3-1, 2.1-1 and 2.1-2. Buckling is addressed in Appendix 2.10.3.

<u>Material Property</u>	<u>Cask</u>	<u>Baskets</u>
Elastic Modulus, E, 10 ⁶ psi	27.3	25.3 to 27.3
Yield Strength, S _y , ksi	45.2	37.3 to 45.2
Ultimate Strength, S _u , ksi	96.9	87.8 to 96.9
Design Stress, S _m , ksi	32.3	29.2 to 32.3
Allowable Stresses, ksi		
Containment Structures (other than forgings)		
Accident Primary Membrane S.I.	67.8	61.5 to 67.8
Accident Primary Membrane + Bending S.I.	96.9	87.8 to 96.9
Accident Pure Shear	40.7	36.9 to 40.7
Accident Bearing-Seal Surfaces	45.2	37.3 to 45.2
-elsewhere	96.9	87.8 to 96.9
Containment Structures (forgings)		
Accident Primary Membrane S.I.	66.9	60.7 to 66.9
Accident Primary Membrane + Bending S.I.	95.7	86.7 to 95.7
Accident Pure Shear	38.0	34.4 to 38.0
Accident Bearing-Seal Surfaces	45.2	37.3 to 45.2
-elsewhere	95.7	86.7 to 95.7

<u>Material Property</u>	<u>Cask</u>	<u>Baskets</u>
Non-Containment Structures		
Accident Primary Membrane		
S.I.	67.8	61.5 to 67.8
Accident Primary Membrane		
+ Bending S.I.	96.9	87.8 to 96.9
Accident Pure Shear	40.7	36.9 to 40.7
Accident Bearing	96.9	87.8 to 96.9

The 9 steps of this accident condition end drop analysis are carried out in the same fashion as was done for the normal end drop analysis, Section 2.6.7.1. In most cases, accident condition results can be obtained directly from normal condition results by applying the ratio of maximum accident condition 'g' load to maximum normal condition 'g' load. This ratio is $49.1/24.7 = 1.99$.

(1) End Drop Computer Analysis (EYDROP-PC)

The end drop analysis was performed utilizing the energy-balance computer program, EYDROP-PC, documented in Appendix 2.10.1. Two cases were considered assuming fully effective impact limiters and considering the variations in foam strength due to temperature effects and the possible variations in foam strength from nominal values (see Section 2.3 and Appendix 2.10.2). Fully effective impact limiter cases consider the entire impact limiter impact area available for crush. This area does not include that of the soft corners extending 12.5 inches inward from the 128 inch outer diameter. Based on 1/4 scale test results from Reference (27) for a similar cask (i.e., observed impact limiter displacements and cask acceleration), the fully effective impact limiter assumption is realistic. Output for each of the end drop cases is presented in Section 2.10.2. Worst case deflections and accelerations are utilized for all analyses. From Table 2.10.2-10 of Section 2.10.2, the maximum accident condition acceleration and impact limiter deflection for a 30 foot end drop are 49.1 g's and 8.69 inches, respectively.

(2) Outer Cask Lid Bending Analysis Assuming No Impact Limiter Support

From Section 2.6.7.1, Item (2), maximum lid bending stress for a unit (1 'g') acceleration is 320 psi. Applying 49.1 g's acceleration, the bending stress becomes:

$$\sigma = (320)(49.1) = 15,712 \text{ psi}$$

Combining this stress with stresses due to a 351 psig internal pressure (9,004 psi bending stress, -351 psi radial stress) results in a stress intensity of S.I. = 15,712 + 9,004 + 351 = 25,067 psi.

Note:

That pressure stress results of Section 2.6.1.3 have been increased by the ratio of 351 psig to 284 psig.

The accident allowable primary membrane plus bending stress intensity, S_a , is given as 96,900 psi. Therefore, the outer cask lid bending stress Margin of Safety is:

$$M.S. = (S_a/\sigma) - 1 = (96,900/25,067) - 1 = \underline{+2.87}$$

(3) Outer Cask Lid Bending Analysis Assuming Full Impact Limiter Support

From Section 2.6.7.1, Item (3), maximum lid bending stress for a unit acceleration is 788 psi. Applying 49.1 g's acceleration, the bending stress becomes:

$$\sigma = (49.1)(788) = 38,691 \text{ psi}$$

Internal pressure would reduce this bending stress, therefore, the outer cask lid bending stress Margin of Safety is:

$$M.S. = (S_a/\sigma) - 1 = (96,900/38,691) - 1 = \underline{+1.50}$$

(4) Outer Cask Lid Bolt Stresses

Outer cask bolt stresses under hypothetical accident end impact conditions are computed and summarized within Tables 2.10.5-3 and 2.10.5-4 (for 1.625 nominal thread diameter) of Appendix 2.10.5. Assuming an impact acceleration of 49.1 g's acting on the full weight of the lid (9,974 lbs), BWR basket (16,508 lbs) and payload (32,188 lbs), the net bolt tensile stress is found as 68,962 psi, assuming equal loading of all 32 lid bolts. These values include pressure and thermal stress effects. These stresses neglect impact limiter reaction forces, which would reduce estimated bolt impact stresses to near zero (if considered). With a total tensile stress of 68,962 psi, the resultant margin of safety (versus S_y = maximum 105,000 psi) is computed, in Table 2.10.5-3 (for 1.625 nominal thread diameter), as +0.52. At 250°F S_y = 97,400 psi and the margin of safety would still be positive at +0.41.

Bolt engagement is checked as follows:

From Section 2.6.7.1, Item (4), a 58,553 psi bolt stress corresponds to a 8,462 psi shear stress in internal threads of the forging. With a 68,962 psi bolt stress, shear stress becomes 9,966 psi. With an allowable shear stress for accident conditions of 38,000 psi ($.42 S_u$) at 250°F, the margin of safety becomes:

$$M.S. = (38,000/9,966) - 1 = \underline{+2.81}$$

Note:

If the bolts were stressed to their accident condition allowable limit of S_y = 97,400 psi at 250°F, the shear stress in the forging would become 14,076 psi and a positive margin would still exist. The specified engagement length for these bolts is therefore sufficient to develop the full bolt strength.

(5) Payload Basket Stresses

Payload Basket stresses and margins of safety are analyzed and presented in Section 2.7.7. Significant worst case results are summarized in Tables 2.7.1.1-1 for the outer shells, and 2.7.1.1-2 for the truss core panels.

TABLE 2.7.1.1-1
STRESS SUMMARY: Basket Outer Shell

-
- Loading Condition: Accident Condition Flat End Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Non-Containment Structure
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: 368 °F
 - Allowable Stress, S_a : 64,300 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Referen</u> <u>Section</u>
	<u>Radial</u> σ_r	<u>Hoop</u> σ_t	<u>Axial</u> σ_z	<u>Shear</u> τ_{rz}	
End Drop	0	0	-2,501	0	2.7.1.1(5)
Direct Stress Sum:	0	0	-2,501	0	
Principal Stresses:	0	0	-2,501		
Stress Differences:	2,501	0	0		

Maximum Stress Intensity (SI): 2,501 psi

Allowable Stress Intensity, S_a : 64,300 psi

Margin of Safety ($[S_a/SI] - 1$): +25.7

TABLE 2.7.1.1-2
STRESS SUMMARY: Truss Core Panels

-
- Loading Condition: Accident Condition Flat End Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Non-Containment Structure
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: 420 °F
 - Allowable Stress, S_a : 63,300 psi

	<u>Stress Components (psi)</u>				<u>Reference</u>
	<u>Radial</u> σ_r	<u>Hoop</u> σ_t	<u>Axial</u> σ_z	<u>Shear</u> τ_{rz}	
<u>Load Condition</u>					<u>Section</u>
End Drop	0	0	-2,501	0	2.7.1.1(5)
Direct Stress Sum:	0	0	-2,501	0	
Principal Stresses:	0	0	-2,501		
Stress Differences:	2,501	0	0		

Maximum Stress Intensity (SI): 2,501 psi
 Allowable Stress Intensity, S_a : 63,300 psi
 Margin of Safety ($[S_a/SI] - 1$): +25.3

(6) Stresses in the Cask Shells and Lead (Maximum Fabrication Stress Condition Assumed)

The principal concern for the cask under normal end drop conditions is with buckling of the inner shell. As shown by the following stress calculations and the detailed buckling evaluation presented in Section 2.10.3, buckling will not occur as the result of the hoop and axial compressive stresses which develop in the outer cask inner shell under accident end drop conditions.

Various initial conditions can be assumed for the accident drop event. In particular, a temperature must be assumed in order to establish an initial fabrication stress for the inner shell. A lower assumed temperature will result in a higher initial hoop stress on the inner shell (see Section 2.6.2) but higher allowable stresses. For purposes of this analysis, drops at 250°F (conservatively bounding maximum lead and outer cask shells normal temperatures per Section 3.5.3, Table 3.5.3-2) and -20° F are considered. Regardless of temperature, the maximum 'g' load of 49.1 g's (for -20°F drop case) is conservatively employed.

To adequately bound the consequences of the drop event at a given temperature, two initial lead conditions are also considered. The first assumes that the lead has shrunk onto the inner shell and away from the outer shell. In addition, due to the combined effects of friction between the lead and inner shell, and axial shrinkage of the lead relative to the stainless shells, axial gaps will develop between the lead and the steel structures at the top and bottom end of the lead column. These axial gaps are important in that, until friction is overcome, under increased axial loading, the lead will impose a direct axial load on the inner shell. Once friction is overcome, the lead will become supported at its base (the bottom of the lead column) and will grow radially outward due to the 'Poisson Effect' under increased axial loading. This radial growth will tend to relieve the initial fabrication hoop stress as the lead separates from the inner shell. If sufficient axial load develops, the lead would grow out to the outer shell creating tensile hoop stresses therein, and under further loading would eventually flow back inward into the inner shell, thereby developing compressive hoop stresses in the inner shell.

From Section 2.6.2, hoop stress in the inner shell due to fabrication is as follows:

Inner Shell	
Temperature	Hoop Stress
<u>(°F)</u>	<u>(psi)</u>
250	-1,462 (extrapolated)
-20	-3,099

Note:

The outer shell hoop stress is considered negligible since the lead separates from the outer shell upon cooling.

The equivalent pressure at the lead/inner shell interface is as follows:

$$p = \sigma t / r$$

Where:

$$t = 1.125 \text{ in}$$

$$r = 29.625 \text{ in}$$

Thus, the interface pressures at the different temperatures are:

Interface	
Temperature	Pressure, p
<u>(° F)</u>	<u>(psi)</u>
250	55.5
-20	117.7

With a coefficient of friction, f , for lead on stainless steel assumed to fall in the 0.5 to 1.0 range (Reference (33) p. 3-26), the load, P , which can be supported by friction at the lead/inner shell interface, may be determined as follows:

$$P = \pi D L p f$$

Where:

$D = 59.25$ in (inner shell outside diameter)

$L = 177.0$ in (lead column height)

p = interface pressure, psi

$f = 0.5$ to 1.0 (coefficient of friction)

Applying the interface pressures determined earlier, the total load which may be supported is:

Temperature (°F)	Coefficient of Friction	Load Supported (lbs)
250	0.5	914,583
	1.0	1,829,166
-20	0.5	1,938,911
	1.0	3,877,823

With the total lead weight equal to 50,037 pounds, per Table 2.2-1, the minimum 'g' load which can be supported by friction is $(914,583/50,037) = 18.3$ g's. The maximum 'g' load which can be supported by friction is $3,877,823/50,037 = 77.5$ g's (-20°F with friction coefficient of 1.0). Since end drop g-loads range from 29.1 g's at the maximum foam temperature to 49.1 g's at -20°F, lead slippage may or may not occur. The following analysis assumes no slippage for the 49.1 accident condition 'g' load. Axial stress in the inner and outer shells of the outer cask are therefore determined by ratioing the normal end drop results available in Section 2.6.7.1, Item (6) by a factor of $49.1/24.7 = 1.99$ g's. The following table summarizes the axial and hoop stress results.

Temperature (°F)	Inner Shell Stresses (psi)		Outer Shell Stresses (psi)	
	Axial (σ_1)	Hoop (σ_2)	Axial (σ_1)	Hoop (σ_2)
250	-14,318	-1,462	-8,060	0
-20	-14,318	-3,099	-8,060	0

Tables 2.7.1.1-3 and 2.7.1.1-4 present the corresponding Margins of Safety for the more critical inner shell. Zero pressure is used in order to maximize compressive stresses in the inner shell. Note: Axial stresses will actually decrease with increasing drop temperature since g loads are lower at higher temperatures as shown in Table 2.10.2-8 of Appendix 2.10.2 (due to decreased foam strength with increased temperature). This effect is conservatively ignored in Tables 2.7.1.1-3 and 2.7.1.1-4. Buckling is addressed in Section 2.10.3.

(7) Stresses in the Cask Shells and Lead (Zero Fabrication Stress Condition Assumed)

The second initial lead condition assumes that the fabrication stress has fully crept away resulting in a stress free column of lead just in contact with the inner and outer shells. This is a potential worst case since any axial load imposed on the lead will directly load, radially, both the inner and outer shells (i.e., the lead need not flow away from the inner shell, into the outer shell and back into the inner shell to develop a compressive hoop stress in the inner shell).

For this condition, initial stresses in the lead and the steel shells are taken as zero. As axial load is applied to the lead and shells, the lead will attempt to move downward and outward and develop pressures on both the inner and outer shells as shown below:

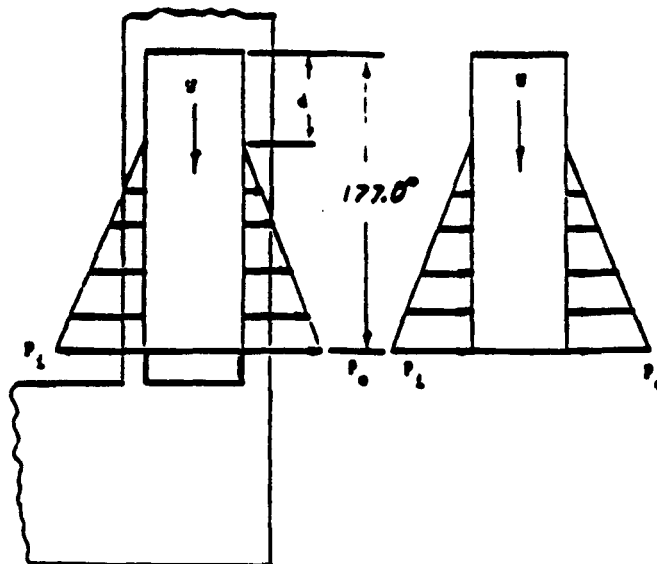


TABLE 2.7.1.1-3
STRESS SUMMARY: Cask Inner Shell, Maximum
Fabrication Stresses at -20°F, Zero Internal Pressure

-
- Loading Condition: Accident Condition Flat End Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Containment Boundary
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: -20 °F
 - Allowable Stress, S_u : 70,000 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Referen</u> <u>Section</u>
	<u>Radial</u>	<u>Hoop</u>	<u>Axial</u>	<u>Shear</u>	
	σ_r	σ_t	σ_z	τ_{rz}	
Fab. and End Drop	0	-3,099	-14,318	0	2.7.1.1(6)
Direct Stress Sum:	0	-3,099	-14,318	0	
Principal Stresses:	0	-3,099	-14,318		
Stress Differences:	14,318	11,219	3,099		

Maximum Stress Intensity (SI): 14,318 psi

Allowable Stress Intensity, S_u : 70,000 psi

Margin of Safety ($[S_u/SI] - 1$): +3.82

TABLE 2.7.1.1-4
 STRESS SUMMARY: Cask Inner Shell, Maximum
 Fabrication Stresses at 250 °F, Zero Internal Pressure

-
- Loading Condition: Accident Condition Flat End Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Containment Boundary
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: 250 °F
 - Allowable Stress, S_u : 67,830 psi

	<u>Stress Components (psi)</u>				<u>Referenc</u>
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					<u>Section</u>
Fab. and End Drop	0	-1,462	-14,318	0	2.7.1.1(6)
Direct Stress Sum:	0	-1,462	-14,318	0	
Principal Stresses:	0	-1,462	-14,318		
Stress Differences:	14,318	12,856	1,462		

Maximum Stress Intensity (SI): 14,318 psi

Allowable Stress Intensity, S_u : 67,830 psi

Margin of Safety ($[S_u/SI] - 1$): +3.74

For purposes of this analysis, let $p_i = p_o = p$. This is considered reasonable once lead starts to flow.

Conservatively, assuming that the lead receives no axial support at its base, but rather is supported solely by friction at the inner and outer shells, the maximum value of the pressure, p , is determined as follows:

$$W = W_L = (49.1)(50,037) = 2,456,817 \text{ lbs}$$

For the maximum pressure, p , use the minimum expected coefficient of friction of $f = 0.5$. Then, the force, W , is:

$$W = p\pi(29.625 + 33.125)(177.0 - d)f$$

Therefore,

$$p_{\max} = 24,925/(177.0 - d)$$

Assuming lead flow initiates at the yield strength of the lead, the height of the lead column, d , required to yield the lead cross-section, due to 49.1 g's accident condition acceleration, is:

$$\sigma_y = \rho d(49.1 \text{ g's})$$

Where:

$$\sigma = 0.41 \text{ lb/in}^3$$

Then,

$$d = (0.04967)\sigma_y$$

Therefore, for 250°F, and -20° F, and conservatively estimating an upper bound on the lead properties delineated in Figures 2.3-1 through 2.3-4, the pressures and hoop stresses may be summarized as follows:

Temperature (°F)	σ_y (psi)	d (in)	P_{max} (in)	σ_2 ($P_{max}R/t$ psi)	
				Inner Shell	Outer Shell
250° F	-390	19.4	158.2	-4,087	3,573
-20° F	-1,000	49.7	195.8	-5,058	4,422

Maximum axial stress in the shells is determined by multiplying the results available from Section 2.6.7.1, Item (7) by the ratio of accident to normal end drop 'g' loads.

The resultant stresses in the inner and outer shells of the outer cask for the accident condition end drop load of 85.0 g's are summarized in the following table.

Temperature (°F)	Inner Shell Stresses (psi)		Outer Shell Stresses (psi)	
	Axial (σ_1)	Hoop (σ_2)	Axial (σ_1)	Hoop (σ_2)
250	-9,391	-4,087	-6,042	3,573
-20	-9,391	-5,058	-6,042	4,422

Tables 2.7.1.1-5 and 2.7.1.1-6 present the corresponding Margins of Safety for the more critical inner shell. Zero pressure is used in order to maximize compressive stresses in the inner shell. Again, it is noted that the margin presented for the 250°F case is conservative in that at this temperature, 'g' loads will actually be less than the 49.1 g's conservatively considered. Buckling is addressed in Appendix 2.10.3.

TABLE 2.7.1.1-5
STRESS SUMMARY: Cask Inner Shell, Zero
Fabrication Stresses at -20°F, Zero Internal Pressure

- Loading Condition: Accident Condition Flat End Drop, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Containment Boundary
- Stress Allowable (Section 2.1.2.1): $0.7 S_u$
- Material Temperature: -20 °F
- Allowable Stress, S_u : 70,000 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Reference Section</u>
	<u>Radial</u> σ_r	<u>Hoop</u> σ_t	<u>Axial</u> σ_z	<u>Shear</u> τ_{rz}	
End Drop	0	-5,058	-9,391	0	2.7.1.1(7)
Direct Stress Sum:	0	-5,058	-9,391	0	
Principal Stresses:	0	-5,058	-9,391		
Stress Differences:	9,391	4,333	5,058		

Maximum Stress Intensity (SI): 9,391 psi
 Allowable Stress Intensity, S_u : 70,000 psi
 Margin of Safety ($[S_u/SI] - 1$): +6.45

TABLE 2.7.1.1-6
STRESS SUMMARY: Cask Inner Shell, Zero
Fabrication Stresses at 250 °F, Zero Internal Pressure

-
- Loading Condition: Accident Condition Flat End Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Containment Boundary
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: 250 °F
 - Allowable Stress, S_a : 67,830 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Referenc Section</u>
	Radial	Hoop	Axial	Shear	
	σ_r	σ_t	σ_z	τ_{rz}	
End Drop	0	-4,087	-9,391	0	2.7.1.1(8)
Direct Stress Sum:	0	-4,087	-9,391	0	
Principal Stresses:	0	-4,087	-9,391		
Stress Differences:	9,391	5,304	4,087		
Maximum Stress Intensity (SI): 9,391 psi					
Allowable Stress Intensity, S_a : 67,830 psi					
Margin of Safety ($[S_a/SI] - 1$): <u>+6.22</u>					

- (8) Determine the amount of lead slump.

The analytical determination of the amount of lead slump has already been made as presented in Table 2.6.7.1-8. It shows that the maximum gap that may be expected at the end of the lead gamma shield is 1.670 inches.

- (9) Fuel assembly payload behavior

Considerations of the payload fuel assemblies for the accident condition end drop loading are discussed in Section 2.9.

2.7.1.2 Corner And Oblique Drops

Analysis of the NuPac 140-B Cask behavior during the accident corner drop including oblique drop orientations is analyzed for an angle of 85° from the horizontal where Appendix 2.10.2, Table 2.10.2-8 shows a maximum inertia load of 73.5 g's. This oblique drop angle was found to represent the worst case for a similar cask as reported in Reference (27). Other drop angles will be studied in detail during the final design.

Accident corner and oblique drops lead to the possibility of a secondary impact, or slapdown, of the cask on the end opposite the initially impacted end. This possibility is addressed in Reference (27) for a similar cask. In that Reference, the consequences of slapdown are shown to be less severe than those associated with a flat side drop from a height of 30 feet. For this reason, and the similarity of the NuPac 140-B Cask, slapdown is not felt to be a limiting case for the package. Slapdown will be studied in detail during the final design.

Analysis of the NuPac 140-B Cask behavior during the corner drop, including oblique drop orientations, will be performed in the following steps:

- (1) Determine the force-deflection relations for a range of initial impact angles, using the CYDROP-PC computer program. Utilize the force-deflection data from CYDROP-PC as input to the computer program OBLIQUE-PC to determine the package response for various initial impact angles.

- (2) Based on the accelerations and rotations predicted in Step (1), determine the equivalent 'static' forces on the cask and payload baskets, considered as rigid bodies with coincident mass centers and zero relative motion.
- (3) Based on the loads from Step (2), analyze the cask shells and payload baskets as simple beams, finding the internal forces: shear, moment, and thrust. Compute stresses for the maximum value of each of these forces, and by inspection of the relative magnitudes of these maximum component stresses, select load cases for further analysis which will have the maximum combined stresses.
- (4) Analyze the cask shells for stress, assuming the two shells act as parallel beams, under the loads determined in Step (3).
- (5) Analyze the payload baskets for stress under the loads determined in Step (3).
- (6) Analyze the impact limiter attachment forces.
- (7) Based on the forces from Step (6), calculate the impact limiter attachment stresses.
- (8) Analyze the stresses in the outer cask lid bolts.
- (9) Analyze the maximum impact limiter deformations and residual clearances for cask hard spots.
- (10) Address lead effects in oblique drops.

Material properties and allowable stresses are available respectively from Tables 2.3-1, 2.1-1 and 2.1-2. Material properties and allowables corresponding to a maximum enveloping temperature (250°F) for the outer cask and (250 to 600°F) for baskets are as follows. Properties and allowables at other temperatures can be readily obtained from Tables 2.3-1, 2.1-1 and 2.1-2. Buckling is addressed in Appendix 2.10.3.

<u>Material Property</u>	<u>Cask</u>	<u>Baskets</u>
Elastic Modulus, E, 10^6 psi	27.3	25.3 to 27.3
Yield Strength, S_y , ksi	45.2	37.3 to 45.2
Ultimate Strength, S_u , ksi	96.9	87.8 to 96.9
Design Stress, S_m , ksi	32.3	29.2 to 32.3
Allowable Stresses, ksi		
Containment Structures (other than forgings)		
Accident Primary Membrane S.I.	67.8	61.5 to 67.8
Accident Primary Membrane		
+ Bending S.I.	96.9	87.8 to 96.9
Accident Pure Shear	40.7	36.9 to 40.7
Accident Bearing-Seal Surfaces	45.2	37.3 to 45.2
-elsewhere	96.9	87.8 to 96.9
Containment Structures (forgings)		
Accident Primary Membrane S.I.	66.9	60.7 to 66.9
Accident Primary Membrane		
+ Bending S.I.	95.7	86.7 to 95.7
Accident Pure Shear	38.0	34.4 to 38.0
Accident Bearing-Seal Surfaces	45.2	37.3 to 45.2
-elsewhere	95.7	86.7 to 95.7
Non-Containment Structures (Note: No non-containment forgings exist)		
Accident Primary Membrane S.I.	67.8	61.5 to 67.8
Accident Primary Membrane		
+ Bending S.I.	96.9	87.8 to 96.9
Accident Pure Shear	40.7	36.9 to 40.7
Accident Bearing	96.9	87.8 to 96.9

(1) Corner Drop Computer Analysis (CYDROP and OBLIQUE):

Analytic predictions of package performance for drop orientations impacting upon the corner of the package employ two computer programs: CYDROP-PC and OBLIQUE-PC, documented in Section 2.10.1. CYDROP-PC uses an energy-balance technique to determine loads and deformations of the impact limiter. Since CYDROP-PC assumes all drop energy is absorbed in deformations of the impact limiter, it provides valid results when the impact orientation places the center of gravity directly over the impacted corner, approximately 25° from the vertical for the NuPac 140-B Cask. At other impact orientations, the force-deflection values generated by CYDROP-PC are used in OBLIQUE-PC, a dynamic analysis program that considers rotational motion effects.

Two (2) cases were run which accounted for variations in foam strength due to temperature effects and the possible variations in foam strength from nominal values (see Section 2.3 and Appendix 2.10.2). Detailed output for each of the analysis cases is presented in Section 2.10.2. Note: The load and stress evaluations presented in this section are typically based on the results from the -20°F foam stress case (i.e., stiffest foam). Impact limiter responses (attachment forces, deformations, and residual clearances) presented herein directly consider both analysis cases.

(2) Equivalent 'Static' Loads:

Equivalent static loads acting on the NuPac 140-B Cask as the result of an accident corner/oblique drop were determined in exactly the same manner as for the normal corner/oblique drop (see Section 2.6.7.2, Item (2)).

Based on a similar cask from Reference (27) the worst case cask shell stress for buckling occurred at the 85° oblique angle. Thus, the worst case static loads will be assumed to occur for this angle and the cask shells will be analyzed for bending, compression and shear in item (4) below. Detailed analysis results and calculations will be presented for other oblique angles in the final Safety Analysis Report of Packaging. The other potentially serious loading problems will affect cask impact

limiter attachment fastener and cask lid fastener integrity. These load cases are dealt with in Item (6) below.

(3) Worst case outer cask shell loads determination.

As mentioned in item (2) above, the worst case oblique drop angle is assumed to be 85° relative to the horizontal. From Table 2.10.2-14 of Appendix 2.10.2, the following cask loads occur for the 85° drop angle:

- ° Inertia load = 73.5 g's
- ° Thrust load (axial force), $P=12,353,210$ lbs
- ° Maximum Moment, $M=30,845,080$ in lbs
- ° Shear load (transverse force), $V=1,023,117$ lbs

Compared with the end drop inertia load of 49.1 g's the 73.5 g value above seems high since the difference in drop angles is only 5°. The soft corners of the impact limiter were neglected for conservatism for oblique drops. Note also that the corresponding end drop axial force is 8,284,549 lbs. (from Table 2.10.2-10 of Appendix 2.10.2). It is significant that the end drop axial force is less than the 85° oblique drop axial force given above.

In the final design, studies will be made comparing shear, thrust, and moment for all oblique drop angles to verify that the 85° angle presents the worst case combination. The method of stress recovery to be employed is taken from Reference (38) and is outlined below.

- ° Assume that the cask shells represent a laminated circular simple beam, and that plane sections remain plane with no slippage between the lead and surrounding steel shells.
- ° The axial contribution for material (i) is given by PE_i/AE , the bending contribution is given by Mc_i/EI , and the shear by $2VE_i/AE$, where:

$$\begin{aligned} AE &= \text{relative axial stiffness of laminated cask} \\ &= A_1 E_1 + A_2 E_2 + A_3 E_3 \end{aligned}$$

$$EI = \text{relative bending stiffness of laminated cask} \\ = E_{i_s} I_{i_s} + E_l I_l + E_{o_s} I_{o_s}$$

$$A_{i_s} = \text{area of inner shell} = 205.4 \text{ in}^2$$

$$A_l = \text{area of lead shell} = 690.0 \text{ in}^2$$

$$A_{o_s} = \text{area of outer shell} = 319.3 \text{ in}^2$$

$$E_{i_s} = \text{modulus of inner shell} = 28.7 (10)^6 \text{ psi at } -20^\circ\text{F}$$

$$E_l = \text{modulus of lead} = 2.42(10)^6 \text{ psi at } -20^\circ\text{F}$$

$$E_{o_s} = \text{modulus of outer shell} = 28.7(10)^6 \text{ psi at } -20^\circ\text{F}$$

$$I_{i_s} = \text{moment of inertia of inner shell} = (\pi) (29.063)^3 (1.125) = 86,761 \text{ in}^4$$

$$I_l = \text{moment of inertia of lead shell} = (\pi) (31.375)^3 (3.5) = 339,601 \text{ in}^4$$

$$I_{o_s} = \text{moment of inertia of outer shell} = (\pi) (33.875)^3 (1.5) = 183,180 \text{ in}^4$$

Thus from the equations and the shell loads above:

$$AE = 5.89(10)^9 + 1.67(10)^9 + 9.16(10)^9 = 1.67 (10)^{10} \text{ lb-in}^2$$

$$EI = 2.49(10)^{12} + 0.82(10)^{12} + 5.26(10)^{12} = 8.57(10)^{12} \text{ lb-in}^2$$

$$\sigma_{\text{THRUST}} = (12,353,210) (28.7(10)^6) / (1.67(10)^{10}) = 2.12(10)^4 \text{ psi}$$

$$\sigma_{\text{BENDING}} = (30,845,080) (34.625) (28.7(10)^6) / (8.57(10)^{12}) = 3.58(10)^3 \text{ psi}$$

$$\sigma_{\text{SHEAR}} = (2) (1,023,117) (28.7(10)^6) / (1.67(10)^{10}) = 3.52(10)^3 \text{ psi}$$

- ° Typical calculations will be made as above for all oblique angles based upon the shear, moment and thrust data from Table 2.10.2-14 of Appendix 2.10.2. All comparisons, as above, will be arbitrarily based on calculations for the outer cask 1.5 inch thick shell.

(4) Calculate shell stresses assuming parallel beams.

Stress recovery of the cask shells will be conservatively calculated as follows utilizing some of the information from step (3):

° Outer Shell bending stress

$$\sigma_b = Mc / (I_{o,s} + I_{i,s}) = (30,845,080) (34.625) / (269,941) \\ = 3,956 \text{ psi}$$

° Inner Shell bending stress

$$\sigma_b = (R_{i,s} / R_{o,s}) 5,830 = (29.625 / 34.625) 3956 = 3,385 \text{ psi}$$

° Outer and Inner Shell axial stress

$$\sigma_a = P / (A_{o,s} + A_{i,s}) = (12,353,210) / (524.7) = 23,543 \text{ psi}$$

° Outer and Inner Shell shear stress

$$\sigma_s = (2)V / (A_{o,s} + A_{i,s}) = (2) (1,023,117) / (524.7) = \\ 3,900 \text{ psi}$$

In the above, it will be noticed that the effects of the lead shell have been conservatively neglected. Thus, the terms for AE and EI calculated in item (3) above were modified by leaving the lead terms out.

Corresponding margins of safety are computed for the cask inner and outer shells in Tables 2.7.1.2.-1 and 2.7.1.2.-2. In these two tables, loads are taken at -20°F and allowables at 250°F. Zero pressure is used in order to maximize compressive stresses in the inner shell. Again, it is noted that the margins presented for the 250°F case are conservative in that at this temperature, 'g' loads will actually be less than the 73.5 g's conservatively considered. Note that maximum shear stresses calculated above are conservatively combined for maximum principal stresses and calculation of margin of safety. Furthermore, since the major stress contribution at the 85° oblique angle comes from axial loads, stress allowable is based upon primary membrane stress intensity.

TABLE 2.7.1.2-1
STRESS SUMMARY: Cask Inner Shell, Zero
Fabrication Stresses at -20°F, Zero Internal Pressure

-
- Loading Condition: Accident Condition Oblique Drop, 85°, -20 °F Foam, Fabrication Stresses at -20°F
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Containment Boundary
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: 250 °F
 - Allowable Stress, S_u : 67,830 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Reference Section</u>
	<u>Radial</u>	<u>Hoop</u>	<u>Axial</u>	<u>Shear</u>	
	σ_r	σ_t	σ_z	τ_{rz}	
Fabrication	0	-3,099	-3,174	0	2.6.2
Oblique Drop, 85°	0	0	-26,928	3,900	2.7.1.2(4)
Direct Stress Sum:	0	-3,099	-30,102	3,900	
Principal Stresses:	0	-2,548	-30,654		
Stress Differences:	30,654	28,106	2,548		

Maximum Stress Intensity (SI): 30,654 psi
 Allowable Stress Intensity, S_u : 67,830 psi
 Margin of Safety ($[S_u/SI] - 1$): 1.21

TABLE 2.7.1.2-2
STRESS SUMMARY: Cask Outer Shell, Zero
Fabrication Stresses at 20 °F, Zero Internal Pressure

-
- Loading Condition: Accident Condition Oblique Drop, 85°, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Non-Containment Boundary
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: 250 °F
 - Allowable Stress, S_u : 67,830 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Reference Section</u>
	Radial	Hoop	Axial	Shear	
	σ_r	σ_t	σ_z	τ_{rz}	
Oblique Drop, 85°	0	0	-27,499	3,900	2.7.1.2(4)
Direct Stress Sum:	0	0	-27,499	3,900	
Principal Stresses:	542	0	-28,584		
Stress Differences:	28,584	28,042	542		

Maximum Stress Intensity (SI): 28,584 psi
 Allowable Stress Intensity, S_u : 67,830 psi
 Margin of Safety ($[S_u/SI] - 1$): +1.17

Thus, axial and bending stress values are combined above for both shells.

Fabrication stresses are also included in Table 2.7.1.2.-1 for the inner shell which were obtained from Section 2.6.2 for -20°F. The lead was assumed to be separated from the outer shell.

(5) Payload Basket Stresses

Payload basket stresses and margins of safety are analyzed and presented in Section 2.7.7; however, only for end drop and side drop inertia loads. Oblique drop effects will be analyzed in detail during the final design. The only difference of concern is the jump from 49.1 g's for end drop to 73.5 g's for the 85° oblique drop. It is believed that enough margin exists for the end drop basket analysis to accommodate the 73.5 g effects.

(6) Impact Limiter Attachment Forces

The maximum impact limiter attachment moment under accident conditions is given in Appendix 2.10.2 as $2.443(10)^6$ in-lbs (Table 2.10.2-22). The normal condition analysis, Section 2.6.7.2, item (6), addressed a $13.33(10)^6$ in-lb moment. This resulted in a 63,261 lb maximum attachment force. By ratio, the maximum accident condition attachment force becomes 69,525 pounds.

(7) Impact Limiter Attachment Stresses

The tensile stress area of a 1.50 - 6 UNC bolt is 1.3832 in^2 per Reference (32), thus, the 1.25 inch shaft provides the minimum cross sectional area. The maximum impact limiter attachment bolt stress is:

$$\begin{aligned} S_b &= P_{\max}/A = 69,525/(\pi/4)(1.25)^2 \\ &= 56,654 \text{ psi} \end{aligned}$$

From Table 2.1-2, the allowable accident condition stress for non-containment fasteners is given as the greater of $0.7S_u$ or S_y . A 250°F

maximum bolt temperature is used for this evaluation. Material data are per Table 2.3-1.

$$0.7S_u = 0.7(125,000) = 87,500 \text{ psi}$$

$$S_y = 97,400 \text{ psi}$$

Therefore, the allowable stress is 97,400 psi. The impact limiter attachment bolt Margin of Safety is:

$$M.S. = (S_y/S_u) - 1 = (97,400/56,654) - 1 = \underline{+0.72}$$

Bolt engagement is checked as follows:

From Section 2.6.7.2 item (7), a 51,550 psi bolt stress corresponds to a shear stress in the internal threads equal to 8,870 psi. For the 56,654 psi accident condition bolt stress, the corresponding shear stress becomes 9,748 psi. With an allowable shear stress in the XM-19 stainless steel flange of 40,700 psi at 250°F, the Margin of Safety becomes:

$$M.S. = (40,700/9,748) - 1 = \underline{+3.18}$$

(8) Stresses in the Outer Cask Lid Bolts

This evaluation directly parallels the analysis and rationale of that used for normal conditions see Section 2.6.7.2, Item (8). Comprehensive analyses of all oblique orientations are summarized in Tables 2.10.5-3 and 2.10.5-4. Minimum margin of safety is found at a near vertical orientation of 80°, with respect to horizontal. At this orientation angle, direct tension stress due to impact, pressure effects, thermal effects, and preload effects is estimated at 95,389 psi for all 32 bolts. The associated shear stress is 1,736 psi. The resultant minimum margin of safety (versus S_y) is computed at +0.10.

As described within Appendix 2.10.5, bolt impact stresses are calculated assuming a triangular distribution of bolt forces and external impact forces arbitrarily (and conservatively) applied at the corner of the cask. Thus, external energy absorber forces are not employed to reduce

bolt tensile forces. For near vertical impacts, this is a most conservative assumption.

It should be noted that a yield criterion for closure bolts is very conservative for bore-sealed packages, such as the NuPac 140-B Cask and Fuel Package, since sealing is dependent on radial, not axial, geometry. This fact was graphically demonstrated by a successful leak test of a similar cask from Reference (27) with 'loose' (untightened) bolts. For face-sealed packages, a yield criterion may be quite appropriate since sealing in this instance depends upon unchanging geometry in the axial direction.

Bolt engagement margin may be demonstrated by comparison with end drop analyses. From Section 2.7.1.1, item (4), margin of safety is estimated at + 2.81 with a bolt tension of 68,962 psi. In these corner/oblique analyses, maximum bolt tensile stress is estimated at 95,389 psi. Thus the margin for bolt engagement is:

$$M.S. = [(1+2.81)(68,962)/(95,389)] - 1 = +1.75$$

(9) Impact Limiter Deformations and Residual Clearances

Maximum impact limiter deformations and residual clearances due to accident corner/oblique drops are directly available from Appendix 2.10.2, Table 2.10.2-17. Per that appendix, maximum crush of an impact limiter corner is 29.20 inches for the hot condition foam stress case and occurs for a drop angle of 55.0° with respect to horizontal.

Minimum residual clearance for the main cask body is 6.77 inches for the hot condition foam stress case and occurs for a drop angle of 5.0° with respect to horizontal. These deformations and residual clearances are of little consequence for the NuPac 140-B Cask.

(10) Lead Effects In Oblique Drops

The purpose of this section is to address the structural consequences of possible plastic deformation of the cask lead shielding during oblique drop events. The approach used is different from that employed in the

oblique drop analyses presented in Section 2.6.7.2 and 2.7.1.2 of this report, where the lead was simply treated as a distributed load. For the stress evaluation herein, the analysis results for flat end drop loadings (outlined in Section 2.7.1.1) were utilized, with modifications made to account for the bending moments which develop in oblique drop orientations. The focus of this analysis is on the more critically loaded outer cask inner shell. The stresses in this shell are specifically determined for drops at -20°F and 250°F.

The basic approach taken to account for lead effects was to derive, for the 85° oblique drop angle which was found to be worst case from Reference (27), at each temperature, the axial (thrust) load acting on the cask. Once this load was established, an interpolation of the end drop stress data of Section 2.7.1.1 was performed to determine stress levels in the outer cask inner shell corresponding to the applied thrust loading. This evaluation accounted for fabrication induced stresses as well as the stresses due to impact induced thrust loadings. Once these stresses were available, stresses due to cask bending during oblique impacts was directly superimposed to give the overall stress state in the inner shell.

Oblique impact loads are available from the data presented in Appendix 2.10.2. Using this data, the appropriate axial (thrust) load components were derived for the 85° impact angle and each drop temperature. For purposes of this analysis, the worst case foam data (that is, the highest strength foam) was utilized at each temperature. Thus, per Appendix 2.10.2, for -20°F drops, Table 2.10.2-4 was used. Likewise, for 250°F drops, the same table was used. Using the OBLIQUE-PC impact data, internal loads analyses were carried out in the same manner as in Sections 2.7.1.2 (item 3). These analyses determined cask bending moments for the 85° drop orientation and temperature of interest. Stresses in the outer cask inner shell resulting from these moments were then calculated and combined directly with the stresses resulting from fabrication and axial thrust.

It was found in Reference (27) for a similar Cask, that the most critical drop angles were near vertical (flat end drop). This is due to the fact that, though cask bending moments might increase somewhat at lower drop

angles, thrust loads decrease rapidly with reduced impact angles. The thrust loading on the cask has a more dominant effect on the resultant axial stress in the outer cask inner shell than does the bending moment, thus the orientations producing significant thrust loadings (i.e., those near vertical) become the worst case.

Stress summaries for the worst case Hypothetical Accident Condition at both temperatures of interest are presented in Tables 2.7.1.2-3 and 2.7.1.2-4. These summaries are presented for the worst case drop orientation of 85° as determined in Reference (27). Buckling checks for these oblique drop load cases are included in Appendix 2.10.3.

By way of example, details of the development of Table 2.7.1.2-3 are presented in the remainder of this Section. Table 2.7.1.2-4 was developed in a similar fashion.

1. Determination of thrust load:

Per Table 2.10.2-14 of Appendix 2.10.2, the maximum impact force for the case of an 85° drop orientation (from vertical) is 12,395,510 pounds. The corresponding thrust load is 12,353,210 pounds and the corresponding axial 'g' load is 73.5 g's.

2. Determination of stresses due to fabrication and thrust:

Per Table 2.7.1.1-3 of Section 2.7.1.1 (item 6), the hoop and axial stresses corresponding to a 49.1 'g' thrust loading are directly available. These stresses are directly increased by the ratio of the actual thrust 'g' load to the 49.1 'g' load. This results in the following hoop and axial stresses:

$$\text{Hoop stress} = (73.5/49.1)(-3,099) = -4,639 \text{ psi}$$

$$\text{Axial stress} = (73.5/49.1)(-14,318) = -21,433 \text{ psi}$$

3. Determination of stress due to cask bending:

The analysis used in Section 2.7.1.2 (item 4) gives the corresponding axial membrane stress in the outer cask inner shell as follows:

$$\begin{aligned}\text{Axial stress} &= Mc/I = 30.845(10)^6(29.625)/269,941 \\ &= -3,385 \text{ psi}\end{aligned}$$

where,

- c = outer radius of outer cask inner shell
- I = moment of inertia of outer cask inner plus outer shells

4. Combined stresses due to thrust, bending and fabrication:

Combining the stresses from steps 2 and 3 above results in the following hoop and axial stresses in the inner shell of the outer cask:

$$\begin{aligned}\text{Hoop stress} &= -4,639 \text{ psi} \\ \text{Axial stress} &= -21,433 + (-3,385) = -24,818 \text{ psi}\end{aligned}$$

5. Comparison of stresses with allowables:

Table 2.7.1.2-3 is a direct update of Table 2.7.1.1-3. As indicated, the allowable primary membrane stress limit is satisfied. The Margin of Safety is equal to +1.82.

TABLE 2.7.1.2-3
STRESS SUMMARY: Cask Inner Shell, Maximum
Fabrication Stresses at -20°F, Zero Internal Pressure

-
- Loading Condition: Accident Condition Oblique Drop, 85°, -20 °F Foam,
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Containment Boundary
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: 20 °F
 - Allowable Stress, S_u : 70,000 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Reference Section</u>
	Radial	Hoop	Axial	Shear	
	σ_r	σ_t	σ_z	τ_{rz}	
Fab & Oblique Drop, 85°	0	-4,639	-24,818	0	2.7.1.2(10)
Direct Stress Sum:	0	-4,639	-24,818		
Principal Stresses:	0	-4,639	-24,818		
Stress Differences:	24,818	20,179	4,639		

Maximum Stress Intensity (SI): 24,818 psi
 Allowable Stress Intensity, S_u : 70,000 psi
 Margin of Safety ($[S_u/SI] - 1$): +1.82

TABLE 2.7.1.2-4
STRESS SUMMARY: Cask Inner Shell
Fabrication Stresses at 250 °F, Zero Internal Pressure

- Loading Condition: Accident Condition Oblique Drop, 85°, -20 °F Foam
- Structural Material: SA-240, Type XM-19
- Stress Condition: Primary Membrane
- Structural Criterion: Containment Boundary
- Stress Allowable (Section 2.1.2.1): $0.7 S_u$
- Material Temperature: 250 °F
- Allowable Stress, S_s : 67,830 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Reference</u> <u>Section</u>
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
Fab & Oblique Drop, 85°	0	-1,462	-24,818	0	2.7.1.2(10)
Direct Stress Sum:	0	-1,462	-24,818	0	
Principal Stresses:	0	-1,462	-24,818		
Stress Differences:	24,818	23,356	1,462		

Maximum Stress Intensity (SI): 24,818 psi
 Allowable Stress Intensity, S_s : 67,830 psi
 Margin of Safety ($[S_s/SI] - 1$): +1.71

2.7.1.3 Flat Side Drop

Analysis of the NuPac 140-B Cask for the accident condition side drop follows the same methodology as that for the normal side drop, Section 2.6.7.3. Since all calculations for the normal side drop were performed on a linear basis, the results for the accident side drop can be found by multiplying the normal condition results by the ratio of accident side drop 'g' load to normal side drop 'g' load. This section presents only the results of the accident side drop analysis. For details of the methods used, refer to Section 2.6.7.3.

Analysis of the NuPac 140-B Cask behavior during the accident side drop impact is performed in the following steps.

- (1) Analyze the impact force using the SYDROP-PC computer program.
- (2) Analyze the cask outer shell as though it, alone, carries the loads.
- (3) Analyze the cask inner and outer shells as though they act in parallel to carry the loads.
- (4) Analyze the payload basket. Apply cask deflections to the payload baskets and evaluate stresses.
- (5) Analyze the cask lid for bearing against the top ring forging of the cask. Analyze the cask closure bolts. Analyze the top ring forging for shear.
- (6) Analyze the cask end plate welds for shear.
- (7) Calculate the side impact clearances for the lifting and tie-down trunnions.
- (8) Address fuel assembly payload behavior.

Material properties and allowable stresses are available respectively from Tables 2.3-1, 2.1-1 and 2.1-2. Material properties and allowables corresponding to a maximum enveloping temperature (250°F) for the outer cask and (300 to 600°F) for baskets are as follows. Properties and allowables at other temperatures can be readily obtained from Tables 2.3-1, 2.1-1 and 2.1-2. Buckling is addressed in Appendix 2.10.3.

<u>Material Property</u>	<u>Cask</u>	<u>Baskets</u>
Elastic Modulus, E, 10 ⁶ psi	27.3	25.3 to 27.3
Yield Strength, S _y , ksi	45.2	37.3 to 45.2
Ultimate Strength, S _u , ksi	96.9	87.8 to 96.9
Design Stress, S _d , ksi	32.3	29.2 to 32.3
Allowable Stresses, ksi		
Containment Structures (other than forgings)		
Accident Primary Membrane S.I.	67.8	61.5 to 67.8
Accident Primary Membrane		
+ Bending S.I.	96.9	87.8 to 96.9
Accident Pure Shear	40.7	36.9 to 40.7
Accident Bearing-Seal Surfaces	45.2	37.3 to 45.2
-elsewhere	96.9	87.8 to 96.9
Containment Structures (forgings)		
Accident Primary Membrane S.I.	66.9	60.7 to 66.9
Accident Primary Membrane		
+ Bending S.I.	95.7	86.7 to 95.7
Accident Pure Shear	38.0	34.4 to 38.0
Accident Bearing-Seal Surfaces	45.2	37.3 to 45.2
-elsewhere	95.7	86.7 to 95.7
Non-Containment Structures (Note: No non-containment forgings exist)		
Accident Primary Membrane S.I.	67.8	61.5 to 67.8
Accident Primary Membrane		
+ Bending S.I.	96.9	87.8 to 96.9
Accident Pure Shear	40.7	36.9 to 40.7
Accident Bearing	96.9	87.8 to 96.9

(1) Side Drop Computer Analysis (SYDROP)

The side drop analysis was performed utilizing the energy-balance computer program, SYDROP-PC documented in Section 2.10.1. Two cases were run which considered the variations in foam strength due to temperature effects and the possible variations in foam strength from nominal values (see Section 2.3). Output for each of the three side drop cases is presented in Section 2.10.2. The drop cases consider 86% of the impact limiter length available for crush, which takes allowance for the soft corners extending 9.5 inches back from the outermost corners. Based on the 1/4 scale test results for a similar cask from Reference (27), it is concluded that this assumption closely reflects reality.

The worst case deflections and the accelerations are typically utilized for all analyses. Maximum resultant acceleration for the side drop accident is 55.6 g's and maximum impact limiter deflection is 19.39 inches per Tables 2.10.2-8 and 2.10.2-17 of Appendix 2.10.2.

(2) Cask Outer Shell Carries Impact Load

The cask shell outer membrane stress (S_m) is as follows for the 55.6 g side drop loading.

$$S_m = (55.6)(453.8 \text{ psi per 'g'}) = 25,231 \text{ psi}$$

The allowable associated with membrane stress in the outer cask outer shell at 250°F is 67,800 psi. The resultant Margin of Safety is:

$$M.S. = (67,800/25,231) - 1 = \underline{+1.69}$$

(3) Analyze the Cask Inner and Outer Shells

The corresponding cask inner shell maximum stress is:

$$S_i = (29.625/34.625)(25,231) = 21,588 \text{ psi}$$

This stress will combine with that due to the 351 psig normal operating pressure (-351 psi radial, 9,027 psi hoop and 4,514 axial stress). The margin of safety is summarized in Table 2.7.1.3-1. Table 2.7.1.3-2 combines the side drop shell membrane stress with fabrication and neglects pressure for maximum shell compressive stress margins. Shear stress is not included in this calculation because its value is zero on the cask at the location of maximum bending stress.

Shear stress in the cask inner and outer steel shells are calculated similar to that of Section 2.6.7.3, item (3); namely

$$\tau = (55.6 \text{ g's})(192.5 \text{ psi per g}) = 10,703 \text{ psi}$$

A detailed buckling analysis for this load combination is presented in Section 2.10.3.

(4) Analyze the Payload Baskets

The payload basket stresses and margins are analyzed and presented in Section 2.7.7. Significant worst case results are summarized in Table 2.7.1.3-3 for the outer shells, and 2.7.1.3-4 for the truss core panels. The effects of cask deflections on payload basket stresses will be analyzed during the final design.

TABLE 2.7.1.3-1
STRESS SUMMARY: Cask Inner Shell

-
- Loading Condition: Accident Condition Side Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Containment Boundary
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$ (forging)
 - Material Temperature: 250 °F
 - Allowable Stress, S_u : 66,900 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Referenc Section</u>
	<u>Radial σ_r</u>	<u>Hoop σ_t</u>	<u>Axial σ_z</u>	<u>Shear τ_{rz}</u>	
Internal Pressure	-351	9,027	4,514	0	2.6.1.3.2
Side Drop	0	0	21,588	0	2.6.7.3(3)
Direct Stress Sum:	-351	9,027	26,102	0	
Principal Stresses:	-351	9,027	26,102		
Stress Differences:	26,453	17,075	9,378		

Maximum Stress Intensity (SI): 26,453 psi
 Allowable Stress Intensity, S_u : 66,900 psi
 Margin of Safety ($[S_u/SI] - 1$): +1.53

TABLE 2.7.1.3-2
STRESS SUMMARY: Cask Inner Shell

-
- Loading Condition: Accident Condition Side Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Containment Boundary
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$ (forging)
 - Material Temperature: 250 °F
 - Allowable Stress, S_u : 66,900 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Reference Section</u>
	<u>Radial σ_r</u>	<u>Hoop σ_t</u>	<u>Axial σ_z</u>	<u>Shear τ_{rz}</u>	
Fabrication	0	-3,099	-3,174	0	2.6.2
Side Drop	0	0	-21,588	0	2.6.7.3(3)
Direct Stress Sum:	0	-3,099	-25,762	0	
Principal Stresses:	0	-3,099	-25,762		
Stress Differences:	25,762	22,663	3,099		

Maximum Stress Intensity (SI): 25,762 psi

Allowable Stress Intensity, S_u : 66,900 psi

Margin of Safety ($[S_u/SI] - 1$): +1.60

TABLE 2.7.1.3-3
STRESS SUMMARY: Basket Outer Shell

-
- Loading Condition: Accident Condition Side Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Non-Containment Structure
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: 368 °F
 - Allowable Stress, S_a : 64,300 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Referen</u> <u>Section</u>
	Radial	Hoop	Axial	Shear	
	σ_r	σ_t	σ_z	τ_{rz}	
Side Drop	0	0	21,500	0	2.7.7
Direct Stress Sum:	0	0	21,500	0	
Principal Stresses:	0	0	21,500		
Stress Differences:	21,500	0	0		

Maximum Stress Intensity (SI): 21,500 psi
 Allowable Stress Intensity, S_a : 64,300 psi
 Margin of Safety ($[S_a/SI] - 1$): +1.99

TABLE 2.7.1.3-4
STRESS SUMMARY: Truss Core Panels

Loading Condition: Accident Condition Side Drop, -20 °F Foam
 Structural Material: SA-240, Type XM-19
 Stress Condition: Primary Membrane
 Structural Criterion: Non-Containment Structure
 Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 Material Temperature: 420 °F
 Allowable Stress, S_a : 63,300 psi

	<u>Stress Components (psi)</u>				Reference Section
	Radial σ_r	Hoop σ_t	Axial σ_z	Shear τ_{rz}	
<u>Load Condition</u>					
Side Drop	0	0	26,900	0	2.7.7
Direct Stress Sum:	0	0	26,900	0	
Principal Stresses:	0	0	26,900		
Stress Differences:	26,900	0	0		
Maximum Stress Intensity (SI): 26,900 psi					
Allowable Stress Intensity, S_a : 63,300 psi					
Margin of Safety ($[S_a/SI] - 1$): <u>+1.35</u>					

(5) Outer Cask Lid Analysis

The most probable mechanism for transfer of lateral loads between lid and body is by frictional forces induced by bolt preloads. Frictional capability will be briefly assessed. Should this frictional force not develop, load transfer will occur by shearing of the lid bolts.

The capability of the bolted joint to withstand this shearing load is fully assessed in Table 2.10.5-3 and 2.10.5-4 of Appendix 2.10.5. From Table 2.10.5-4, total bolt tension, including pressure and preload effects, is found as 57,014 psi. Corresponding shear stress is 9,624 psi. The resultant margin of safety (versus $2S_m$) is computed, in Table 2.10.5-4, as + 0.59.

Returning now to the more likely frictional load transfer mechanism, normal clamping force can be directly assessed by reference to the relations of note (16) associated with Tables 2.10.5-3 and 2.10.5-4 of Appendix 2.10.5. The clamping force per bolt, from Table 2.10.5-3, is given there as:

$$F_b = K_b P / (K_b + K_l) - F_{pre} = -96,287 \text{ lbs}$$

(Minus sign indicates a compressive force)

Where: P = External load, see note (15) = 30,807 lbs

F_{pre} = Preload, see note (6) = 62,536 lbs

K_b = Bolt Stiffness, see note (8) = 2.05×10^7 lbs/in

K_l = Lid Stiffness, see note (9) = 2.35×10^8 lbs/in

The frictional shear resistance capacity, per bolt, is obtained by applying an appropriate coefficient of friction to the predicted clamping force of -96,287 lbs. Reference (36), page 611 gives the coefficient of friction between two 304 stainless steel surfaces as 0.9. For conservatism, this value is reduced to 0.45 for capacity evaluation.

As shown in Table 2.10.5-3, of Appendix 2.10.5, the applied shearing force, per bolt is 17,300 lbs. Thus, the Margin of Safety against slippage is:

$$M. S. = (96,287)(.45)/17,330 - 1 = + 1.50$$

Thus, slippage is clearly not likely to occur and no shear loads are likely to be imposed upon the lid bolts.

In the most unlikely event that the lid displaces laterally to bear upon the body ring forging, lateral loads will be transferred by bearing. The following calculations demonstrate the adequacy of this load transfer mechanism.

The weight of the outer cask lid may be carried in bearing against the lid end ring forging of the outer cask if sliding occurs at the lid to forging interface. Utilizing a Hertz analysis for bearing between two cylinders, including 55.6 g's acceleration:

$$S_b = [2P/\pi L(KD)(CE)]^{0.5}$$

Where:

$$\begin{aligned} P &= \text{load between cylinders} \\ &= \pi(29.90)^2(4.9)(9974)(55.6) = 554,554 \text{ lb} \end{aligned}$$

$$\begin{aligned} L &= \text{length of contact} \\ &= 1.641 \text{ in (approximately)} \end{aligned}$$

$$\begin{aligned} D_1 &= \text{diameter of outer cylinder (maximum)} \\ &= 59.720 \text{ in} \end{aligned}$$

$$\begin{aligned} D_2 &= \text{diameter of inner cylinder (minimum)} \\ &= 59.700 \text{ in} \end{aligned}$$

$$\begin{aligned} KD &= D_1 D_2 / (D_1 - D_2) \\ &= 178,264 \text{ in} \end{aligned}$$

$$CE = 2(1 - \mu^2)/E$$

$$= 6.741(10)^{-6} \text{ in}^2/\text{lb}$$

$$E = 27.0(10)^6 \text{ psi at } 300^\circ \text{ F}$$

$$\mu = 0.30$$

Then,

$$S_b = 4,231 \text{ psi}$$

The accident allowable bearing stress, S_a , is given as 45,200 psi. Therefore, the outer cask lid bearing stress Margin of Safety is:

$$M.S. = (S_a/S_b) - 1 = (45,200/4,231) - 1 = \underline{+ 9.68}$$

The lip of the forging which supports the lid is loaded in shear by the lid. Conservatively assuming that only one-half the lip is effective in resisting this unit load (at 1 'g'), the shear stress is:

$$S_v = (9974)/\{\pi[(38.70)^2 - (29.86)^2]/2\}$$

$$= 10.48 \text{ psi}$$

For the accident condition side impact acceleration of 55.6 g's, the shear stress is:

$$S_v = (55.6)(10.48) = 583 \text{ psi}$$

The accident allowable pure shear stress, S_a , is given as 38,000 psi. Therefore, the outer cask lid lip shear stress Margin of Safety is:

$$M.S. = (S_a/S_v) - 1 = (38,000/583) - 1 = \underline{+ 64.18}$$

(6) Welds Joining the Outer Cask Shells to the End Closure Plate

The worst case shear loads at the joints of the outer cask shells to the end closure plate are found to be as follows:

1. For the outer cask inner shell, the shear load, V_{si} , is taken as the shear load determined previously for normal conditions from Figure 2.6.7.3-2).

$$V_{si} = 50,496 \text{ lb}$$

2. For the outer cask outer shell, the shear load, V_{so} , is taken as the same shear force shown above.

$$V_{so} = 50,496 \text{ lb}$$

Since the welds are full penetration bevel welds, the effective shear area for each shell may be found by the following equation: (i.e. use half the total weld area per Table 17, case 18 of Reference (15).

$$A_w = 2\pi R t / 2$$

Where:

R = mean radius, in

t = shell thickness, in

The stress in the welds is:

$$S_w = V / A_w$$

Applying a unit (1 'g') acceleration, results from the above equations are summarized in the table below:

Shell Weld	R (in)	t (in)	A _w (in ²)	S _w (psi)
Inner	29.063	1.125	102.7	492
Outer	33.875	1.500	159.6	316

Applying 55.6 g's to the outer cask inner and outer shell end closure weld stresses results in:

$$S_{wi} = (55.6)(492) = 27,355 \text{ psi}$$

$$S_{wo} = (55.6)(316) = 17,570 \text{ psi}$$

Table 2.7.1.3-5 - presents stress and Margin of Safety results for the more critical inner shell weld under worst case load combinations. A zero internal pressure is used in order to maximize inner shell compressive stresses.

TABLE 2.7.1.3-5
STRESS SUMMARY: Cask Inner Shell End Welds

-
- Loading Condition: Accident Condition Side Drop, -20 °F Foam
 - Structural Material: SA-240, Type XM-19
 - Stress Condition: Primary Membrane
 - Structural Criterion: Containment Boundary
 - Stress Allowable (Section 2.1.2.1): $0.7 S_u$
 - Material Temperature: 250 °F
 - Allowable Stress, S_u : 66,900 psi

<u>Load Condition</u>	<u>Stress Components (psi)</u>				<u>Reference Section</u>
	<u>Radial σ_r</u>	<u>Hoop σ_t</u>	<u>Axial σ_z</u>	<u>Shear τ_{rz}</u>	
Fabrication	0	-3,099	3,174	0	2.6.2
Side Drop	0	0	21,588	27,355	2.7.1.3(6)
Direct Stress Sum:	0	-3,099	-24,762	27,355	
Principal Stresses:	0	15,490	-43,352		
Stress Differences:	58,842	43,352	15,490		
Maximum Stress Intensity (SI): 58,842 psi					
Allowable Stress Intensity, S_u : 66,900 psi					
Margin of Safety ($[S_u/SI] - 1$): <u>+0.14</u>					

(7) Calculate Side Impact Clearances for the Lifting and Tiedown Trunnions

From Table 2.10.2-17 of Appendix 2.10.2, the maximum impact limiter deflection is 19.39 inches. The maximum protrusion from the 77.40 inch diameter outer cask outer shell is the 5.5 inch long lifting trunnion. Residual clearance is thus:

$$[(128/2) - 19.39] - [(77.40/2) + 5.5] = 0.41 \text{ in}$$

(8) Address Fuel Assembly Payload

Considerations of the payload fuel assemblies for the accident condition side drop loading will be discussed in Section 2.9.

2.7.2 Puncture

Subpart F of 10 CFR 71 requires that a 40 inch free drop of the NuPac 140-B Cask onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface be considered. The bar must be 6 inches in diameter, with the top horizontal and its edge rounded to a radius of not more than 0.25 inches. The cask is to be oriented in a position for which maximum damage is expected and the length of the pin is to be such that maximum damage will occur.

The puncture analyses uses some (or all of) the following data:

h = drop height
= 40 in = 1,016 mm

r = punch edge radius
= 0.25 in = 6.35 mm

d = punch diameter
= 6.0 inches = 152.4 mm

W = package weight
 = 203,159 lbs = 92,135 kg (Section 2.2)

D = package outer shell diameter
 = 69.25 in = 1,758.95 mm

L = package length
 = 203.5 in = 5,168.9 mm

S_u = package outer shell ultimate tensile strength (Section 2.3)
 = 97,400 psi = 68.62 kg/mm² at 240° F

t_a = actual package outer shell thickness
 = 1.50 in = 38.1 mm

t = required package outer shell thickness

Side Puncture:

For impact occurring on the side of the NuPac 140-B Cask, the required cask outer shell thickness, t, for puncture integrity is determined using three independent equations as follows:

Nelms' Equation, Equation 2.1 of Reference (24) is:

$$t = (W/SU)^{0.71}$$

$$= 1.69 \text{ inches}$$

Sakamotos' Equation, Equation 6 of Reference (25) is:

$$(E/S_u) = [0.0030 + (0.047)t/D + (0.002)d/D + (0.006)r/d] \{t^{[1.585 - (0.11)r]} \{d^{[1.465 + (0.077)r]}\}$$

Where energy, $E = (Wh)$ kg-m, lengths are in mm, and S_u is in kg/mm².
Substituting for r , d , and D :

$$[(92,135)(1.016)/97.4] = [0.0034233 + (0.00002672)t](t^{0.8865})(18,426.5)$$

The expression for the outer shell thickness, t , then becomes:

$$0.052157 = [0.0034233 + (0.00002672)t](t^{0.8865})$$

Solving for outer shell thickness, t , yields:

$$t = 18.541 \text{ mm} = 0.73 \text{ in}$$

Shiehs' Equation, Reference (26)

$$(W/S_u) = \{0.37[1 + 6/D + (41/D)^2][e^{(t/3)}](t^{1.68})\} / \{1 + [(0.1)t^{0.87}]\}$$

Where $0.5 < t < D/30$, $45 < D < 90$, $2D < L < 3D$. These conditions are all valid for this case.

Substituting, the expression for the outer shell thickness, t , then becomes:

$$3.9064 = [e^{(t/3)}][t^{1.68}] / \{1 + [(0.1)t^{0.87}]\}$$

Solving for the outer shell thickness, t , yields:

$$t = 1.745 \text{ in.}$$

The three punctures equations resulted in varying results summarized below:

Nelms's Equation	t_{reqd}	=	1.69 inches
Sakamotos' Equation	t_{reqd}	=	0.73 inches
Shiehs' Equation	t_{reqd}	=	1.745 inches

The puncture thickness required for the NuPac 140-B Cask outer shell is calculated using the 1.5 inch structural shell and the neutron shield materials and shells based on their relative ultimate strength. The required puncture thickness is based on XM-19 stainless steel and the different materials of the neutron shield will be calculated as equivalent thicknesses of XM-19 stainless steel. The neutron shield consists of the following materials:

	Su (psi)	Equivalent XM-19 Thickness Thickness (inches)	
Boro Silicone	450	5.875	0.026
Copper	30,000	0.5	0.150
Stainless Steel (XM-19)	100,000	0.125	0.125

The total equivalent thickness of XM-19 provided is $1.5 + 0.026 + 0.15 + 0.125 = 1.80$ inches. XM-19 has a 2.6% reduction in ultimate strength at 240°F. The equivalent thickness at 240°F is therefore $1.8 (.974) = 1.75$ inches. The Margin of Safety resisting puncture for the three equations is summarized below:

	<u>M.S.</u>
Nelms's Equation	+0.04
Sakamotos' Equation	+1.40
Shiehs' Equation	+0.00

A large variation is shown in the Margin of Safety. To more accurately determine the puncture strength of the cask, drop puncture tests have been proposed and will be incorporated in the final design analysis. To determine the overall effect on the cask, consider the puncture impact occurring at the center of length of the package. Applying a unit acceleration and conservatively utilizing only the outer shell for strength, the bending moment induced membrane stresses in the inner and outer shells become:

$$S_m = Mc/I$$

Where:

S_m = membrane stress due to bending

$$M = (W_{oc}L)/8 + (W_{iv}L)/4$$

W_{oc} = Cask weight ready weight
= 180,783 lbs

W_{iv} = Impact Limiter
= 22,376 lbs

L = cask length
= 203.5 inches

c_o = average radius of cask outer shell
= 33.875 inches

c_i = average radius of cask inner shell
= 29.065 inches

$I = 183,270 \text{ in}^4$ (I for cask outer shell)

Then, the unit membrane stress in the inner shell is:

$$S_{mi} = 928 \text{ psi}$$

and, the unit membrane stress in the outer shell is:

$$S_{mo} = 1,084 \text{ psi}$$

The impact acceleration can be estimated from the area, A , and the dynamic flow stress, V_f , of the 6.0 inch diameter, mild steel puncture pin. Assuming the dynamic flow stress, V_f , of the mild steel pin is the mean of yield and ultimate for ASTM A-36 steel, or $(36,000 + 58,000)/2 = 47,000$ psi, acceleration, L , becomes:

$$L = P_f/W$$

Where:

$$P_t = AV_t$$

$$A = \pi(3.0)^2 = 28.274 \text{ in}^2$$

$$W = 180,783 \text{ lbs}$$

Then,

$$L = (28.274)(47,000)/180,783 = 7.35 \text{ g's}$$

Thus, the cask inner and outer shell membrane stresses are:

$$S_{mi} = (7.35)(928) = 6,821 \text{ psi}$$

$$S_{mo} = (7.35)(1,084) = 7,967 \text{ psi}$$

Shear stress for the outer cask shells is determined as follows:

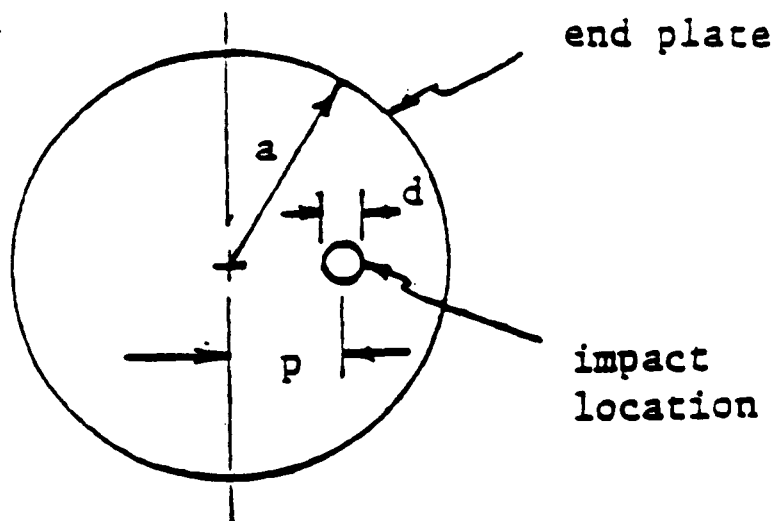
$$\text{Maximum shear force} = V = (7.35)(203,159/2) = 746,609 \text{ lb}$$

$$C = V/A = 746,609/263 = 2,839 \text{ psi}$$

where $A=526/2 = 263 \text{ in}^2$ = half area of cask inner
and outer shells = effective shear area

End Puncture

Using Table 24, Cases 18 and 19, of Reference (15), maximum stress in the end plate as a function of puncture pin impact location is determined. The geometry is as follows:



For a simply supported plate, the maximum moment occurs at the impact point, and is:

$$M = W\{1 + (1 + \mu)\ln[(a - p)/r_o] - [(1 - \mu)(r_o)^2/4(a - p)^2]\}/4\pi$$

For a fixed edge plate, the maximum moment occurs at the impact point or at the near edge, and is:

At impact point:

$$M = W(1 + \mu)\{(4)\ln[(a - p)/r_o] + [r_o/(a - p)]^2\}/16\pi$$

At near edge:

$$M = -W\{2 - [r_o/(a - p)]^2\}/8\pi$$

a = plate radius

= 31.69 in

t = plate thickness

= 8.5 in

p = radial location of puncture pin (inches)

d = diameter of puncture pin

= 6.0 in

$$r_o = d/2 = 3.0 \text{ in}$$

$$r_o' = (1.6r_o^2 + t^2)^{0.5} - 0.675t = 3.571 \text{ in}$$

$$\mu = 0.3$$

$$W = \text{impact force} = \text{force to flow the 6 inch diameter puncture pin} = (3.0)^2(47,000) = 1.329(10)^6 \text{ lbs}$$

With plate bending stress $= V = 6M/t^2$, the calculations can be carried out which result in the following tabular summary.

Maximum Plate Bending Stress (psi)

Impact Location, p (inches)	Plate Boundary Condition	
	Simply Supported	Fixed
0.0	33,689	24,962
10.0	29,338	20,675
15.0	26,318	17,736
18.0	24,021	15,537
20.0	22,179	13,806
23.0	18,677	10,636*
27.0	11,004	4,767*

*Note: maximum occurs at outer edge of plate.

With a yield stress of $V_y = 45,560 \text{ psi}$ at 240°F , yielding of the closure plate will not occur. It is therefore concluded that 'leaktightness' will not be significantly affected by puncture pin impacts occurring anywhere on the end closure. It is further noted that the beneficial effects of having the impact limiters in place were conservatively ignored herein.

The end puncture is of little consequence for any cask component other than the outer cask lid. As determined for the side puncture case, the 6 inch diameter pin will flow when a 7.32 g loading develops on the cask. A much more critical loading case is, therefore, the 49.1 g's associated with a 30- foot flat end drop of the NuPac 140-B Cask and Fuel Package.

2.7.3 Thermal

The hypothetical thermal test condition is analyzed in Section 3.5. From that section, maximum temperatures and pressures are directly available for use.

2.7.3.1 Summary of Pressures and Temperatures

The maximum hypothetical thermal test condition temperatures for the various cask components are presented in Table 2.7.3-1. Maximum internal pressure resulting from the hypothetical accident thermal test condition is 351 psig for any package component as calculated in Section 3.5.4. Finally, to protect against unanticipated pressure buildups (beyond those associated with Regulatory requirements), the cask lid is fitted with a 700 psig rupture disk. The use of 700 psig was chosen because it is sufficiently above the maximum normal or accident condition pressure of 351 psig to preclude venting under all defined regulatory events while low enough to prevent stresses in any other part of the cask from exceeding Regulatory Guide 7.6 accident condition allowables.

Stresses from a unit internal pressure are calculated at 4 points on the containment vessel shells and 2 points on corresponding end closures as shown in Section 2.6.1.3.1. The tables below give the stresses at these points under a 700 psig internal pressure:

Shell Location	Meridional Membrane σ_1 , psi	Hoop Membrane σ_2 , psi	Meridional Bending* σ_1 , psi	Hoop Bending* σ_2 , psi
	7,903	15,806	0	0
P2	9,002	18,004	0	0
P3	9,002	2,702	-28,063	-8,421
P4	7,903	0	-32,312	-9,695

* σ'_1 and σ'_2 are positive when tensile on the shell O.D.

TABLE 2.7.3-1
Maximum Hypothetical Accident Thermal Test Condition Temperatures
(Not Including Damage Conditions)

<u>LOCATION</u>	<u>Temperature (°F)</u>
Basket (center)	602
Basket (shell)	411
Center PWR fuel assembly peak fuel rod cladding	682
Cask Inner Shell	326
Cask Lead Shell	326
Cask Outer Shell	326
Thermal Shield	1,163
Cask End Closure Plate	266
Copper Fins	475
Fuel assembly spacers	443
Cask Lid O-ring Seals	272
Borated-silicone neutron shield	553
Impact Limiter Shell	1,423
Trunnions	260

End Plate	Radial Bending
Location	σ , psi

P5	17,955
P6	17,955

The margins of safety, using Regulatory Guide 7.6 accident condition allowables, are presented below:

Stress Location	Stress Category	Normal Temperature (°F)	Allowable Stress Intensity (psi)	Actual Stress Intensity (psi)	Margin of Safety
P1	Membrane	250	67,800	15,806	+ 3.29
P2	Membrane	250	67,800	18,004	+ 2.77
P3	Membrane	250	66,900	9,002	+ 6.43
	Membrane + Bending	250	95,700	37,065	+ 1.58
P4	Membrane	250	67,800	7,903	+ 7.58
	Membrane + Bending	250	96,900	40,215	+ 1.41
P5	Membrane + Bending	250	95,700	17,955	+ 4.33
P6	Membrane + Bending	250	95,700	17,955	+ 4.33

The stress in the Outer Cask Lid bolts due to a 700 psig internal pressure is:

$$S_b = (\pi/4) (57.0)^2 (700/32 (1.752)) = 31,861 \text{ psi}$$

The accident condition allowable stress for ASTM A-320, Grade L43, bolts at 250°F is 97,400 psi (S_y) so the Margin of Safety in the Cask Lid Bolts is:

$$(97,400/31,861)-1 = \underline{+2.06}$$

In summary, the use of a 700 psig rupture disk will assure that no venting of the cask will occur as the result of specified normal or accident condition regulatory events. In fact, a pressure buildup of 2.0 times that associated with regulatory events would be required for the disks to rupture. Additionally, as evidenced by the preceding evaluation, the disk will rupture before any other cask component fails, should gross, unexpected overpressurization of the cask occur.

2.7.3.2 Differential Thermal Expansion

Differential thermal expansions due to accident conditions are of little consequence for the NuPac 140-B Cask. All stresses resulting from differential expansions can be classified as secondary, displacement limited stresses. As limits on secondary stresses do not apply for accident conditions (per Section 2.1.2, Table 2.1-1), differential expansions in themselves do not compromise the integrity of the cask. Through wall (and through thickness) thermal gradients also result in secondary stresses and again are of little consequence for the cask. To address the requirement specified in paragraph C.7 of Regulatory Guide 7.6 and discussed in Section 2.1.2.2.2.3 regarding the extreme total stress intensity range, analyses were performed in Section 3.5.5. to determine maximum cask stresses and deformations of consequence associated with the hypothetical accident thermal event.

Significant cask thermal stresses and deformations investigated in Section 3.5.5 included the cask shell to end closure interface, cask lid bolts, copper fins, and differential thermal expansion of borated-silicone neutron shield segments. The copper fins are studied primarily to verify that their heat dissipation function will not be upset.

Transient thermal analyses discussed in Section 3.5.5.2 showed that maximum outer shell and copper fin temperatures at the bolted joints were 1°F apart and occurred at identical times during the fire transient. Ignoring the 1°F difference, the thermal stress imposed on the half inch copper fins at the "stud welded bolt" attachment point was found to be 2629 psi. Maximum thermal stresses found along the copper fins at other locations were 2968 psi compression and 832 psi bending. Based on an allowable stress of 5000 psi for 300°F from Reference (6), the above copper fin thermal stresses are not significant. The allowable stresses from Reference (6), are always below the actual yield stresses. Thus, even at the slightly higher peak soakback temperature of 326°F at the "stud welded bolt" attachment point, it is concluded that the heat dissipation function of the copper fins will not be degraded due to loosening from one hypothetical accident thermal event.

The cask lid bolt thermal stress of 9929 psi given in Section 3.5.5.3 has already been incorporated into the analyses of Appendix 2.10.5. Positive margins are shown for the 32, 1.625 - 8 UN bolts; thus, any further discussion shall reference Appendix 2.10.5.

Section 3.5.5.4 mentions that a 61% increase in thermal growth can be expected over the normal design conditions for borated-silicone neutron shield segments adjacent to the cask thermal shield because of the accident condition thermal event. This phenomenon will be studied thoroughly during final design; however, additional forces placed on the copper fins probably will not be found to be significant. The modulus of the borated-silicone is similar to silicone rubber; thus quite low, and at the high temperatures involved, it will be lower. Therefore, corresponding forces on the copper fins due to additional compression of the borated-silicone are not expected to degrade heat dissipation capability. The borated-silicone tolerances, based upon a normal condition temperature of 300°F, given in Section 3.4.5.3 will be recommended for preliminary design. Transient analyses show that peak thermal stresses occur in the load collar 12 minutes after the 30 minute, 1,475 F hypothetical accident thermal test condition; and in the two cask inner and outer shells 9.5 hours afterwards. Shell and nearby forging temperatures are listed below, and are compared with normal condition values.

Cask Forging and Shell Temperatures for Thermal Stress Analyses (F)

	Normal (Shell)	Conditions (Forging)	Accident (Shell)	Conditions (Forging)
° Load Collar	276	288	378	243
° Outer Shell	296	291	269	262
° Inner Shell	297	294	271	265

Corresponding thermal stresses are calculated for accident conditions utilizing the methods of Section 3.4.5.1 and the thermal gradients shown in the Table above. The results of the analysis are presented below.

Thermal Stress Summary for Cask Shells and Load Collar

Cask Component	Wall Thickness (in.)	MEMBRANE STRESS (psi)		BENDING STRESS (psi)	
		Meridional	Hoop	Meridional	Hoop
° Inner Shell	1.13	-468	-1408	-2890	-866
° Outer Shell	1.50	468	-1629	-2953	-866
° Load Collar	0.25	-25166	-31716	-57456	-17237

From the above component stresses, maximum stress intensities for the inner shell, outer shell, and load collar are 4,298 psi, 4,582 psi, and 89,172 psi respectively. To conservatively evaluate the stress state in relation to the requirements of Regulatory Guide 7.6, Paragraph C.7, the worst-case thermal/secondary stress intensity will be used. Note that this stress intensity, 89,172 psi, occurs in the load collar shell, not the containment structure.

Since cask loads are minimal during accident conditions, primary stress in the load collar will be negligible. Thus, primary plus secondary stress intensity will be 89,172 psi. The maximum stress range will be twice this value, or $(2)(89,172) = 178,344$ psi.

Applying an additional factor of 4 to account for possible stress concentrations at structural discontinuities yields a value of $4(178,344) = 713,376$ psi. The appropriate alternating stress will be one-half this value, of $S_{alt} = (0.5)(713,376) = 356,688$ psi. To account for temperature effects, the variation in modulus of elasticity of the material is factored in as follows:

$$\begin{aligned} S_{alt} (378^{\circ}\text{F}) &= (E@70\text{ F}/E@378\text{ F}) S_{alt} \\ &= [28.3(10)^6/26.1(10)^6] (356,688) \\ &= 386,754\text{ psi} \end{aligned}$$

From Reference (6) Appendices, Table I-9.1 (for Figure I-9.2.1), the allowable value of S_{alt} at 10 cycles is 708,000 psi. The resulting Margin of Safety is:

$$\text{M.S.} = (708,000/386,754) - 1 = \underline{+ 0.83}$$

Recognizing the conservative assumption of assuming fully reversing primary and secondary stress states to determine ranges, it is apparent that the actual margin of safety will be larger than +0.83. In any event, the requirement of paragraph C.7 of Regulatory Guide 7.6 is met.

2.7.3.3 Stress Calculations

Except as discussed in Section 2.7.3.2, the concern for accident conditions is with primary, load controlled stresses. The only load controlled stresses during and after the hypothetical accident thermal event are those resulting from pressure and dead weight, and are therefore rather modest. Although temperatures associated with the 30 minute, 1,475 F hypothetical accident thermal event transient are typically higher than those associated with other accident conditions, the NuPac 140-B Cask design is considered, by inspection, to be governed by the significantly more severe drop accident events.

2.7.3.4 Comparison with Allowable Stresses

Based on the discussions presented in Section 2.7.3.1 through 2.7.3.3, it is apparent that large margins of safety exist for the accident condition.

2.7.4 Immersion - Fissile Material

The criticality evaluation presented in Section 6.0 considers the effect of water in leakage. Thus, the requirement of 10 CFR 71.73(c) (4) is met.

2.7.5 Immersion - All Packages

The effect of a 21 psig external pressure due to immersion in 50 feet of water as required by 10 CFR 71.73 (c) (5) is of negligible consequence for the NuPac 140-B Cask. However, contractual requirements in Reference (1) essentially state that the cask must be designed to withstand a 200 meter (656 ft) submergence.

Taking the density of water as 62.4 lbs/ft³, the pressure on the cask submerged at 200 meters is calculated as follows:

$$P = p_h = (62.4)(656)/144 = 284 \text{ psi}$$

Utilizing the cask unit pressure data of Section 2.6.1.3.1 and assuming zero internal pressure, the calculations of Section 2.6.1.3.2 are repeated and summarized below:

Shell Location	Meridional Membrane σ_1 , psi	Hoop Membrane σ_2 , psi	Meridional Bending* σ_1 , psi	Hoop Bending* σ_2 , psi
P1	-3,206	-6,413	0	0
P2	-3,652	-7,304	0	0
P3	-3,652	-1,096	11,386	3,416
P4	-3,206	0	13,109	3,933

* σ'_1 and σ'_2 are positive when tensile on the shell O.D.

End Plate Location	Radial Bending σ , psi
P5	7,285
P6	7,285

With a yield stress of 45,200 psi at 250°F and 55,000 psi at 100°F, structural integrity of the cask regarding its containment function is assured.

2.7.6 Summary of Damage

From the analyses presented in Sections 2.7.1 through 2.7.5, it can be shown that the accident test sequence will not result in any significant structural damage to the NuPac 140-B Cask. Nearly all permanent damage occurs in the external impact limiters as desired. Minor amounts of damage can occur to cask components as follows:

In a 30 foot drop event, the lead shielding can slump to a maximum of 1.670 inches. For a 40 inch drop on a 6.0 inch diameter puncture pin, with impact occurring on the side of the cask at midlength, localized cask damage can occur at the impact point, but the cask outer shell will not be perforated and the overall bending response of the cask remains elastic.

These permanent deformations are of little consequence for the NuPac 140-B Cask as they represent only minor changes in cask geometry. In particular, damage is not sufficient to compromise 'leaktightness' of the cask seals. Lead slump is only of concern relative to shielding. The 1.670 inch maximum slump is, therefore, addressed from a shielding point of view in Section 5.0. For these reasons, the integrity of the cask is not considered to be compromised by the accident test sequence set forth in 10 CFR 71.

2.7.7 Payload Baskets

Both basket designs are analyzed for hypothetical accident conditions using the approach previously described in Section 2.6.8 for normal conditions. The primary differences are threefold, namely:

1. Inertia loads, 49.1 g's for end drop versus 24.7, and 55.6 g's for side drop versus 16.3.
2. Criteria for allowable stress intensity per Table 2.1-2.
3. Lower local structural temperatures; however the higher normal condition temperatures were assumed.

Oblique drop inertia load conditions will be studied thoroughly during the final design phase of the program; however, scoping calculations show that enough margin exists for end drop and side drop orientations to negate possible negative margins for corresponding oblique drop cask orientations.

2.7.7.1 PWR Basket - End Drop

For end drop, the maximum accident load is 49.1 g's. The main potential failure mode is the buckling of elements due to axial compressive stress. Nominal stresses are low, well below yield, and margins for significant components such as outer shells and truss core panels are presented in Tables 2.7.1.1-1 and 2.7.1.1-2. These data also apply to the BWR Basket since both have nearly identical cross-sectional geometry for compressive stress resistance. For the fuel elements carried within the baskets, the axial quasi-static buckling load capability is greater than 82 g's as discussed in Section 2.9 and Reference (11).

For the structural elements of the basket, using simple linear quasi-static analysis, the buckling allowable stresses and the end drop stresses can be obtained by using data from Section 2.6.8.1.2, normal end drop loads. Very similar results can be obtained for the BWR basket from

Section 2.6.8.2.2. The end drop stresses can be obtained from the ratio of normal to accident inertia, g, loads.

From the normal condition analyses the critical buckling stresses for the basket outer shells and truss core panels are 70,158 psi and 4,239 psi respectively. Corresponding respective compressive stresses ratioed by 49.1/24.7 g's are both 2,501 psi.

Thus, the buckling Margin of Safety for the outer shells is:

$$M.S. = (70,158/2,501) - 1 = \underline{27.05}$$

and the buckling Margin of Safety for the truss core panels is:

$$M.S. = (4,239/2,501) - 1 = \underline{+ 0.69}$$

2.7.7.2 PWR Basket-Side Drop

The same ANSYS structural computer models as used for normal conditions in Section 2.6.8 are used for hypothetical accident conditions for side drop inertia loads. As the model is non-linear, the stresses and deflections cannot be scaled up in the same proportion as the inertia, g, loads.

The side drop analysis of the PWR basket, following the assumptions of Section 2.6.8.1, results in maximum stresses for each structural component as presented in Table 2.7.7-1. The maximum deflection is 0.245 inches, which represents a diametrical shortening along the diameter parallel to the direction of the deceleration of the cask and basket. At this deflection, the basket is making contact with the cask along almost 180 degrees of arc. As the deflection is elastic, the basket will return to its original shape.

TABLE 2.7.7-1

Stress Summary For PWR Basket For Hypothetical
Accident Conditions

Load Condition: Side drop, 55.6 g's for maximum local temperatures

<u>Basket Part</u>	<u>Maximum Stress Intensity (KSI)</u>	<u>Part Temperature (F°)</u>	<u>Allowable Stress (KSI)</u>	<u>Margin of Safety</u>
1. Outer Shell	-21.50	368	64.30 (1)	+ 1.99
2. Pinned Truss Core Panels	-12.20	580	61.70 (1)	+ 4.06
3. Welded Truss Core Panels	-26.90	420	63.30 (1)	+ 1.35
4. Web Core	-5.59	580	76.00 (2)	+12.60
5. Lateral Gussets	25.20	420	63.30 (1)	+ 1.51
6. Longitudinal	-21.53	420	63.30 (1)	+ 1.94

Notes: (1) Allowable based on primary membrane, $0.7 S_u$

(2) Allowable based on buckling similar to that of Section 2.6.8.1.2.

Similar stress calculations for the BWR basket show results very similar to those presented in Table 2.7.7-1 for the PWR basket. Positive margins of safety are computed for all BWR basket parts. Detailed BWR stress and margin of safety results will be presented in the final design report.

2.8 Special Form

This section does not apply for the NuPac 140-B Cask.

2.9 Fuel Rods

Regulatory Guide 7.9 requires that analysis or test data be provided showing that fuel rod cladding has structural integrity, justifying the amount of integrity claimed. There is data, both analytical and experimental, showing that the types of fuel elements to be transported in the NuPac 140-B Cask can sustain the worst case design loading conditions.

According to "Dynamic Impact Effects on Spent Fuel Assemblies "by Chun, Reference (11), stresses will be below yield in the Westinghouse 17 x 17 fuel assemblies for a 63 G side drop or an 82 G end drop load. (Of the fuel assemblies examined, the Westinghouse 17 x 17 had the "weakest" structural parameters.) Additional data was presented in the IF-300 SAR, Reference (37) on the strength of fuel rod cladding. Theoretical analyses, including nonlinear, dynamic computer analysis combined with rod bend tests showed that the rods could take significantly higher loads than predicted by Chun, Reference (11), without rupture or other failure to contain the rod solid or gaseous contents.

The NuPac 140-B Cask impact limiter analyses (Appendix 2.10.2) demonstrate that the maximum worst case design end, oblique and side drop loads are 49.1 g's, 73.5 g's and 55.6 g's, respectively. The higher oblique drop loads occur only 5 to 10 degrees off vertical. Thus, the 82 g's limit from Reference (11) is assumed to apply.

Adequate margins are found for the NuPac 140-B Cask loads, even for the more conservative approach based on linear analysis of Reference (11), so the lower capability values are used for this report. Since the cask g loads are less severe than the minimum capabilities of the weakest fuel rods, positive margins are found, and damage to the fuel rods is not likely to occur.

DO NOT REPRODUCE
THIS PAGE

(Blank Page)

2.10 Appendix

- 2.10.1 Nuclear Packaging Proprietary Drop Programs
- 2.10.2 Drop Program Evaluations
- 2.10.3 Buckling Evaluation
- 2.10.4 Finite Element Analyses
- 2.10.5 Closure Bolt Stress Evaluations

APPENDIX 2.10.1

NUCLEAR PACKAGING PROPRIETARY DROP ANALYSIS

COMPUTER PROGRAMS

2.10.1 Description of Nuclear Packaging Proprietary Drop Programs

This section briefly documents the methodology employed by the Nuclear Packaging proprietary computer programs which are used to demonstrate compliance of the package with applicable provisions of 10 CFR 71 for normal and hypothetical accident conditions. The first two subsections deal with the calculation of external and internal forces imposed upon the package, when subjected to drop events. These subsections describe techniques and computer programs developed by Nuclear Packaging, Inc. of Federal Way, Washington, as follows:

- o 2.10.1.1: Describes derivation of energy absorbing impact limiter deflection relations.
- o 2.10.1.2: Describes the methods to evaluate the dynamic behavior of oblique impacts and the associated internal forces generated within the cask body.

2.10.1.1 Impact Limiter Deformation Behavior

The package is protected by foam-filled, energy-absorbing end buffers, called impact limiters. For purposes of analysis, the impact limiters are assumed to absorb, in plastic deformation of foam, the potential energy of the drop event. That is, the analyses assume that none of the drop potential energy is transferred to kinetic or strain energy of the target (the unyielding surface assumption of 10 CFR 71), nor strain energy in the package body itself.

There are three orientations of the package where the potential energy of a drop is assumed totally absorbed by plastic deformations of the impact limiters. At other orientations, where rotational effects are important, the methods outlined in Section 2.10.1.2 are employed. The three orientations where rotational (or pitch) motions play no role in the evaluation of the impact event are:

- o End Drop - on the circular end surface of the impact limiter.
- o Side Drop - on the cylindrical side surfaces of the impact limiters.
- o Corner Drop - with package center of gravity directly above the struck corner of the impact limiter.

For these three orientations, the prediction of impact limiter behavior can be approached from straightforward energy balance principles:

$$E = W(h + \delta) = \int_0^{\delta} F_x dx \quad (1)$$

Where:

W - package weight

h - drop height

δ - maximum impact limiter deformation

F_x - force imposed upon target and package by the impact limiter at a deflection equal to x.

The left-hand term represents the potential energy of the drop. The right hand term represents the strain energy of the deformed impact limiter.

Each of these three orientations is treated by an individual computer program reflecting the differing geometry characteristics of each event. All three computer programs employ common energy balance techniques to assess maximum impact limiter deformations, including utilizing a common description of the crushable energy absorbing foam. The foam typically exhibits a stress-strain plateau of nearly constant stress up to a total strain of 40-60%. Above this strain value, pronounced strain hardening effects commence which reflect the collapse or consolidation of the entrapped bubbles within the foam. Accordingly, a tabular definition of

foam stress-strain relations is employed in each of the three computer programs. This tabular definition is taken directly from measured properties and accurately reflects the strain hardening behavior of the foam up to strains of about 80%.

The following discussion of these three computer programs proceeds from the geometrically simplest (end drop) to the most complex (corner drop).

2.10.1.1.1 End Drop (EYDROP-PC)

EYDROP-PC performs the calculations outlined in Equations (1), (2), and (3) for a trial range of deformation values, δ . For each trial value of total deformation, the energy balance of Equation (1) is monitored and reported. Solution for total impact limiter deformation is found by an interpolated balance of Equation (1). EYDROP-PC assumes a constant foam strain across the crush area, neglecting the affects of any unbacked areas. Quality assurance verification and validation results, and sample problems are given in References (16) and (17).

The force produced by the impact limiter is simply:

$$F_x = A\sigma_\epsilon \quad (2)$$

Where:

A = $\pi D^2/4$, the end area of the package

D = effective diameter of package

σ_ϵ = $\zeta[\epsilon]$, the foam crush stress at a strain of ϵ (3)

$\zeta[\epsilon]$ = the tabular definition of foam stress strain properties

ϵ = x/x_u

x = deformation

x_u = end thickness of impact limiter

2.10.1.1.2 Side Drop (SYDROP-PC)

SYDROP-PC differs from the end drop solution only in the fact that both deformation and strain vary from point to point and total force, at a given crush depth, must be found by geometric integration over these points. The details on this geometry are found in Figure 2.10.1-1. SYDROP-PC assumes all foam is backed, exhibiting homogeneous properties along the package length. Quality assurance verification and validation results, and sample problems are given in Reference (18) and (19).

For each trial deformation value, the force is found as:

$$F_s = 2L \int_0^{x_{max}} \sigma_{Nx} dx$$

Where:

L = effective length of the impact limiter

$$x_{max} = [r_o^2 - (r_o - \delta)^2]^{0.5}$$

σ_{Nx} = $\zeta[\epsilon_x]$, tabular definition of foam stress-strain properties

ϵ_x = foam strain at location 'x'

Referring to Figure 2.10.1-1, the strain at a point 'x' is found by:

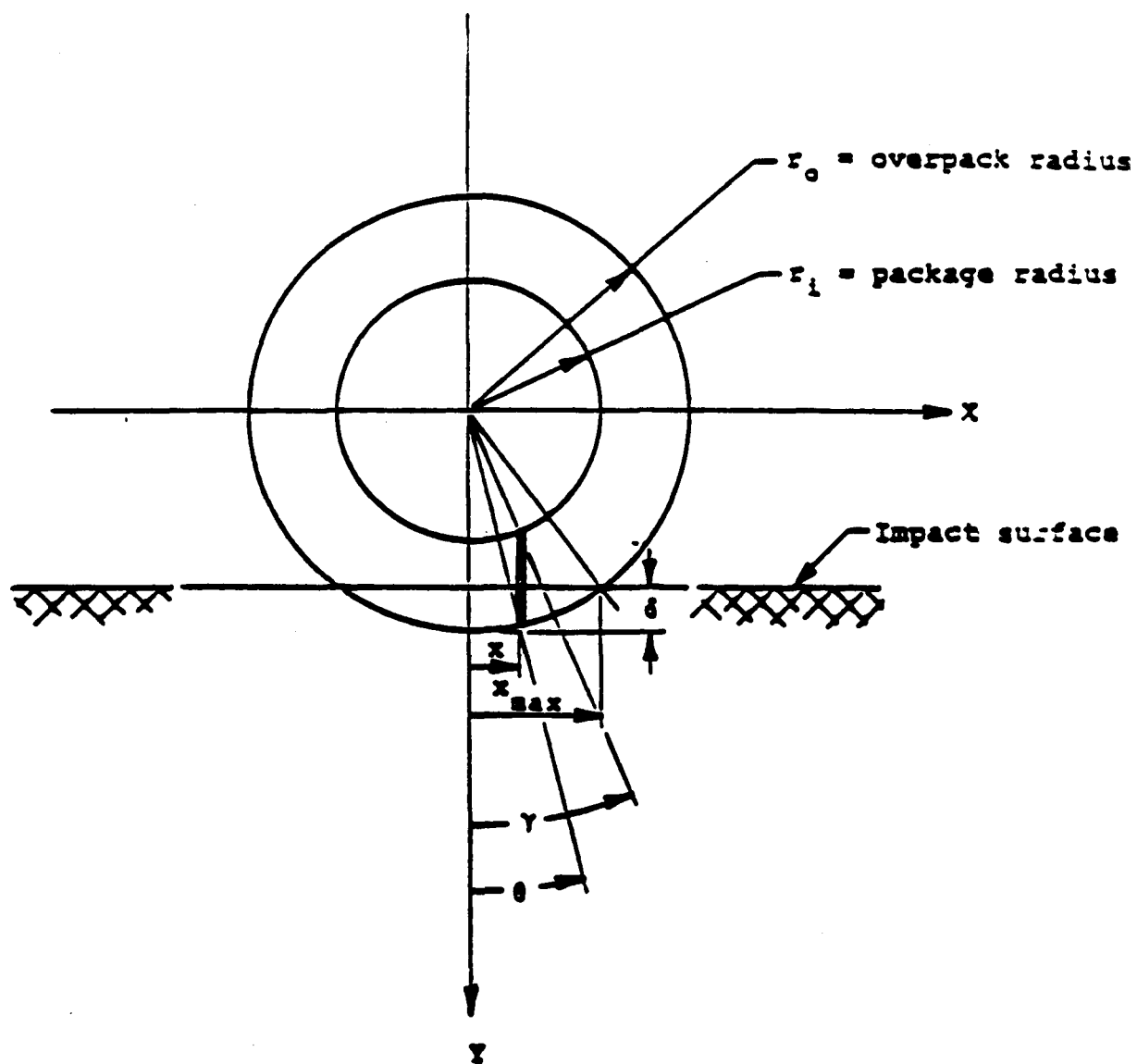
$$\epsilon_x = [\delta - r_o(1 - \cos \theta)] / [r_o(\cos \theta) - r_i(\cos \gamma)]$$

Where:

$$\theta = \sin^{-1}(x/r_o)$$

$$\gamma = \sin^{-1}(x/r_i)$$

FIGURE 2.10.1-1
Side Impact Geometry (SYDROP)



2.10.1.1.3 Corner Drop (CYDROP-PC)

CYDROP-PC is similar to SYDROP-PC, except that a two dimensional geometric integration is required to assess the impact limiter crush force at each deformation. Quality assurance verification and validation results, and sample problems are given in Reference (20) and (21).

CYDROP-PC treats the corner impact of a cylindrical package upon an unyielding surface. The package itself consists of a cylindrical payload portion surrounded by a larger cylindrical column composed of a crushable media. So long as the deformations of the crushable media are modest, the problem may be approximately solved by assuming a uniform crush stress exists over the elliptical surface of the crush plane (contact surface). CYDROP-PC was developed specifically to address problems of large deformations of this crushable media and to treat geometries where the cylindrical impact limiter envelope possesses axisymmetric cylindrical voids (e.g., does not completely cover the cylindrical ends of the payload package).

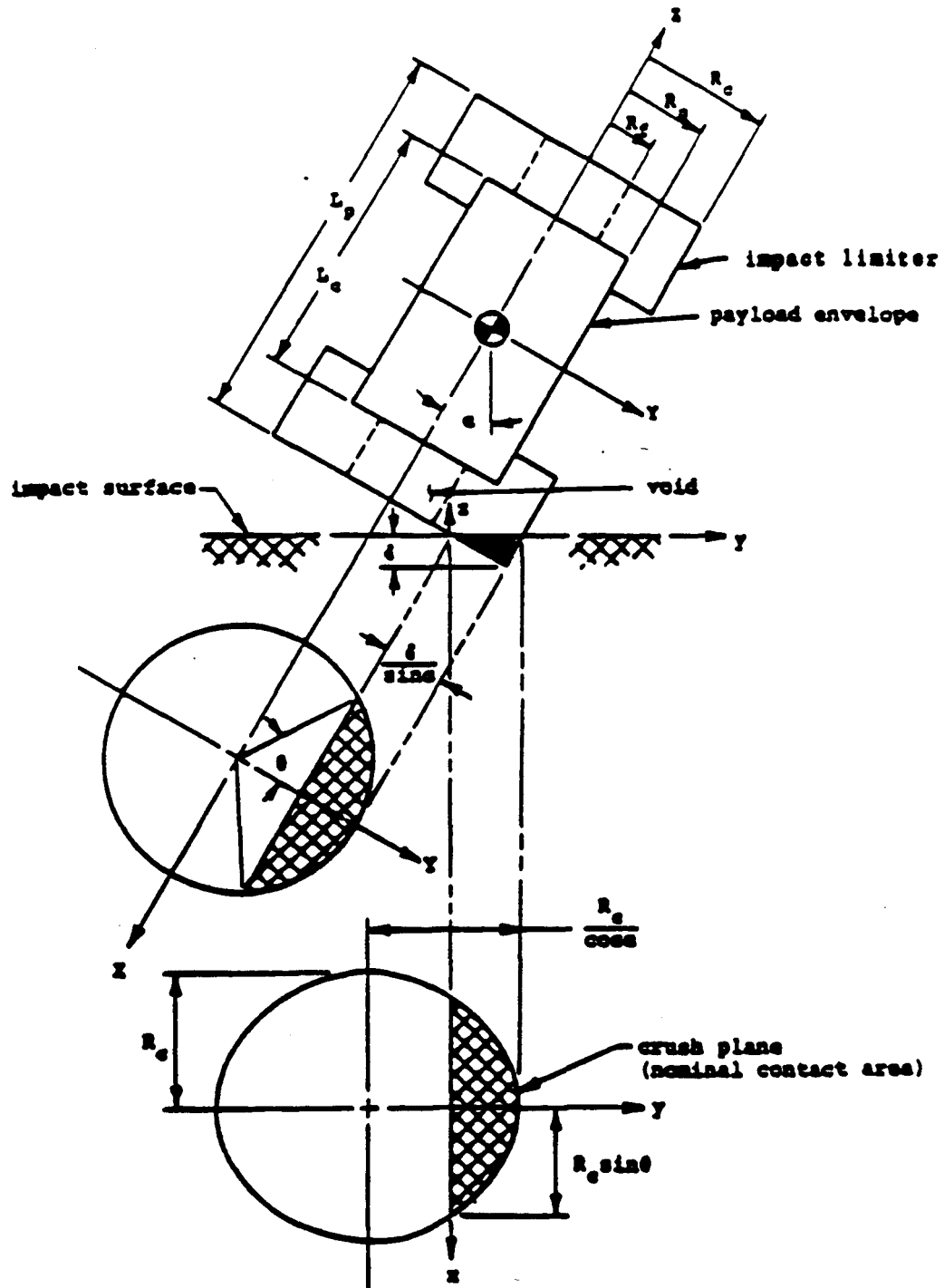
The large deformation behavior of the crushable media is accommodated by determining the actual strain of the crushable media at a point. This strain is used to determine the corresponding stress from an implicit tabular definition of media stress-strain characteristics. The total crush force is found by a double integration over the contact area of the crush plane.

Strain energy absorbed by the crushable media is determined by integrating the crush force and its associated deformation. The package is assumed to be at rest when the computed strain energy value equals the applied drop energy.

The geometric calculations for the contact surface and the associated strains are carried out using a moving (x, y, z) coordinate system in which the x-y plane corresponds to the crush plane, as illustrated in Figure 2.10.1-2. The crush plane itself represents a segment of an ellipse. The contact area is this ellipse segment, provided no cylindrical end void exists. When a cylindrical end void exists, the contact area of the crush plane is reduced by the removal of a second

elliptical region associated with the projection of this void into the contact plane.

FIGURE 2.10.1-2
Corner Impact Geometry (CYDROP-PC)



Calculation of strain is somewhat more complex. In principle, the distance from point (x, y) in the crush plane to the payload is found and denoted, Z_{top} . Similarly, the distance to the undeformed external impact limiter envelope is found and denoted, Z_{bot} . The strain represents deformation divided by original thickness, or:

$$\epsilon = Z_{bot} / (Z_{bot} + Z_{top})$$

At any point (x, y), the calculation of Z_{top} may follow three branches, according to location. The three possible branches relate to the payload surface intercepted. They are the circular bottom of the payload, the cylindrical surface of the payload, and the unbacked regions, each of which are separately addressed below:

The Circular Bottom of the Payload:

The bottom of the payload cylinder describes an ellipse in the crush plane. If point (x, y) is inside this ellipse, the point is considered backed by the bottom of the payload. An exception to this general statement is noted in the discussion of the unbacked region, below.

The Cylindrical Surface of the Payload:

The cylindrical surface of the payload describes a rectangular region tangent to the payload bottom ellipse at its major axes. If point (x, y) is outside the bottom ellipse yet possesses an x-coordinate less than the radius of the payload bottom, the point is considered backed by the payload cylinder.

Unbacked Regions:

Unbacked regions are of two forms - those associated with the cylindrical end void and those near the external surface of the impact limiter. The unbacked region associated with the end void is a point in the crush plane which lies within the ellipse defined by the void circle lying in the plane of the payload bottom. The unbacked region associated with points near the impact limiter extremities is defined by those points (x, y) where the x-coordinate exceeds the radius of the payload volume. Points which are unbacked employ a nominal crush stress for force integration purposes.

The calculation of Z_{bot} , the distance to the undeformed impact limiter envelope, may follow two branches. These branches correspond to intercepts with either the cylindrical surface of the impact limiter or the circular end of the impact limiter.

The analytics describing the geometry discussed above, consists of the sequential application of a series of geometric transformations of surfaces described in the coordinates of the cylindrical package (X, Y, Z) to the coordinates of the contact plane (x, y, z). The surfaces in package coordinates are:

Impact Limiter Cylinder: $X^2 + Y^2 = R_c^2$

Impact Limiter Bottom Circle: $X^2 + Y^2 = R_c^2$
 $Z = -L_c/2$

Payload Cylinder: $X^2 + Y^2 = R_p^2$

Payload Bottom Circle: $X^2 + Y^2 = R_p^2$
 $Z = -L_p/2$

Void Circle at Payload: $X^2 + Y^2 = R_i^2$
 $Z = -L_p/2$

Void Circle at Impact Limiter Exterior: $X^2 + Y^2 = R_i^2$
 $Z = -L_c/2$

CYDROP-PC also performs a sensitivity analysis to determine the amount of unbacked foam at each incremental crush depth. Additionally, this calculation is carried over to the impact force and strain energy. The code automatically prints a warning message if the foam strain exceeds 80% and the ratio of foam strain energy to kinetic energy is less than one ($SE/KE < 1$).

2.10.1.2 Oblique Impact Dynamic Analysis

Impacts at arbitrary orientation angles differ in two major respects from those that occur at angles corresponding to stable or neutral equilibrium (end, side, and center of gravity over struck corner). In the neutral and stable equilibrium conditions, the entire initial kinetic energy of drop is transformed into strain energy associated with plastic deformation of the impact limiter. At arbitrary orientation angles, only a portion of this kinetic energy is transformed into strain energy at the impacted end. The remainder of this kinetic energy becomes rotational motion of the package. The solution approach must properly reflect the continually changing transformation of initial translational kinetic energy into rotational kinetic energy and plastic deformation of the impact limiter energy absorber.

The second major difference between neutral equilibrium impacts and arbitrary angle impacts relates to the rather different load-deflection behavior of the impact limiters at low angle (<30° from horizontal) orientations. Under neutral equilibrium conditions, a major portion of the crush footprint is backed by the cylindrical body of the package, allowing strain hardening effects to stiffen the impact limiter load-deflection relation. At low angle orientations (<30° from horizontal), much of the impact limiter crush footprint is unbacked. Thus, the low angle load-deflection relations are initially quite soft, then abruptly harden as portions of the crush footprint grow into backed regions. As these low angle orientations approach horizontal attitudes, this terminal stiffening phenomena becomes more pronounced.

There are two potential solution paths to problems of this nature - a momentum formulation or a direct solution of the equations of motion. The momentum approach provides an easy and simple means to assess the transformation of translational initial velocities into rotary velocities, hence, total plastic strain energy absorbed by the impact limiter energy absorber. Unfortunately, this momentum formulation does not produce intermediate values of crush force and crush deformation needed to assess impact limiter attachment forces, nor does it conveniently provide a means to incorporate the varying load-deflection

relationships of the impact limiter as a function of orientation angle. Thus, a direct solution of the equations of motion was selected.

The three key problem variables, crush force, F , crush depth, δ , and orientation angle, θ , all vary with time, t , for a given initial orientation angle, θ_0 . The crush force is assumed to act at the centroid of the elliptical crush footprint. For the model illustrated in Figure 2.10.1-3, three independent second order differential equations of motion can be formed:

$$M(\partial^2 X / \partial t^2) = F_x$$

$$M(\partial^2 Y / \partial t^2) = F_y - Mg$$

$$\begin{aligned} I(\partial^2 \theta / \partial t^2) &= \text{sum of moments about package center of gravity} \\ &= \{[\delta(a - c)/c(\sin \theta) + T_b + L/2](\sin \theta)\}F_x \\ &\quad + \{x - [\delta(a - c)/c(\sin \theta) + T_b + L/2](\cos \theta)\}F_y \end{aligned}$$

Where:

M = the package mass = ρL

F = the crush force

g = the gravitational constant = 386.4 in/sec²

I = the rotational mass moment of inertia (as input)

r_0 = the radius of the body

L = the length of the body

T_b = impact limiter bottom thickness

θ = the instantaneous orientation angle of the package with respect to the horizon

ρ = the mass per unit length

x = distance to centroid of footprint

a, c = geometric quantities defined in Figure 2.10.1-4

FIGURE 2.10.1-3
Oblique Impact Geometry

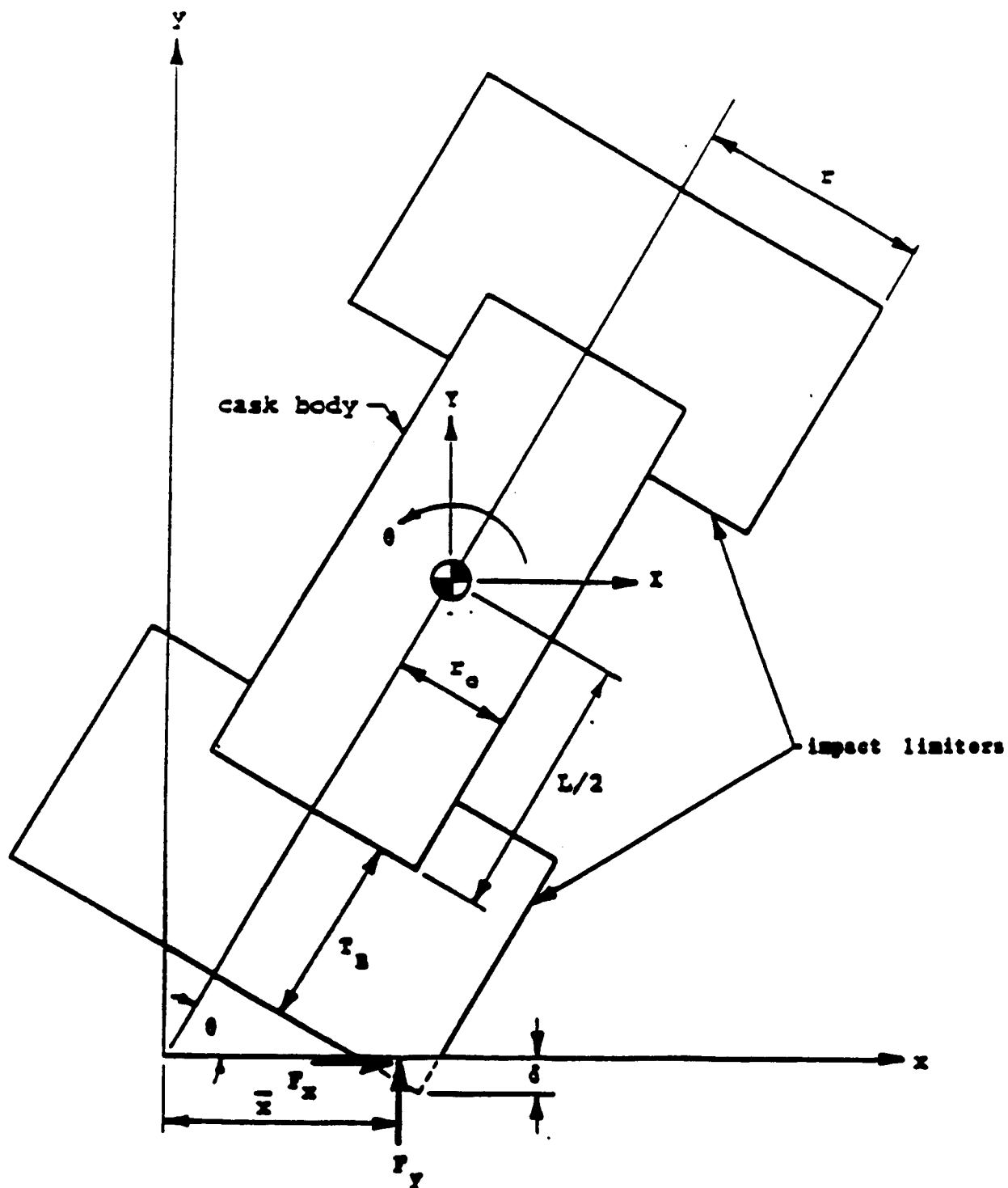
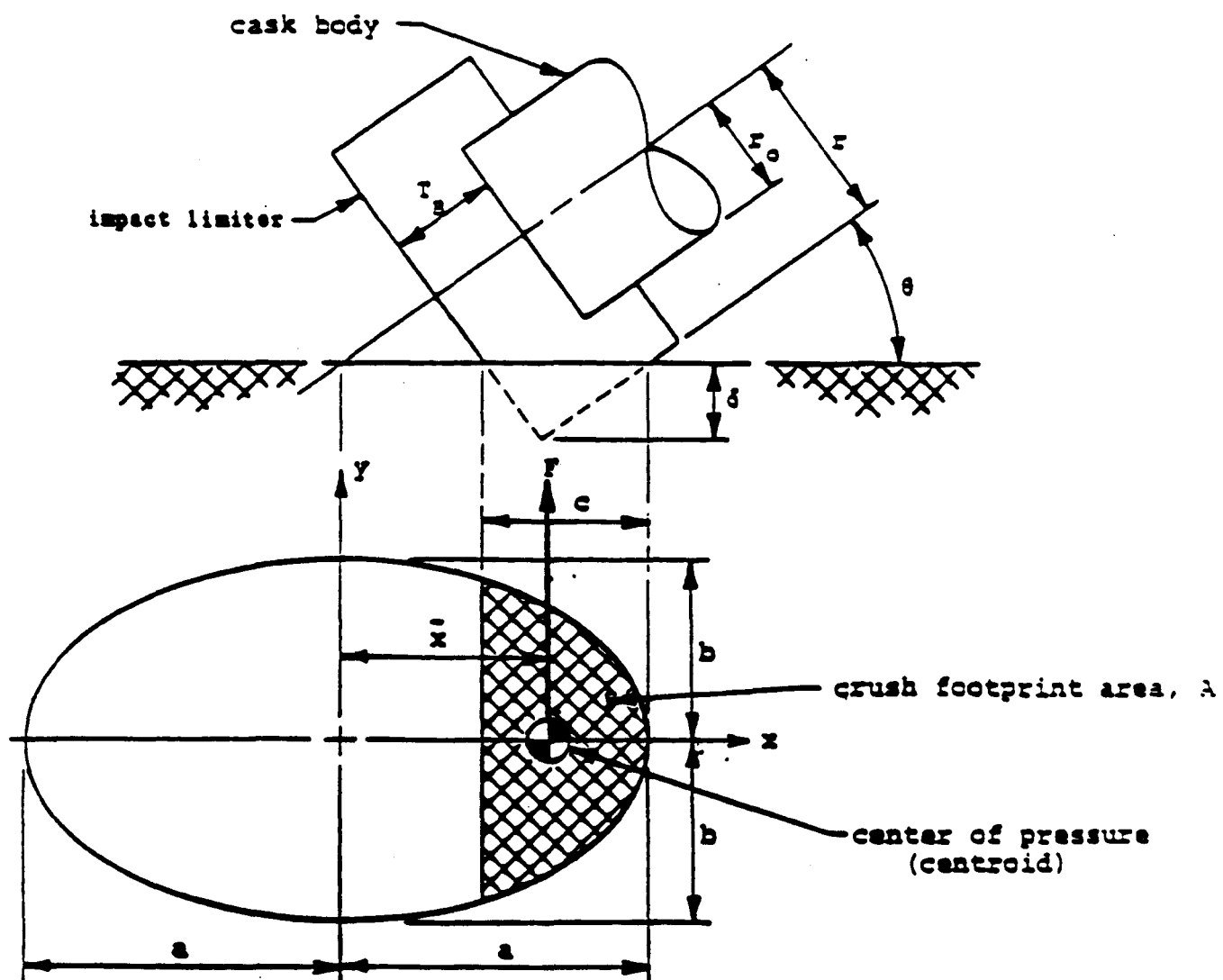


FIGURE 2.10.1-4
Elliptical Footprint Geometry



These differential equations are integrated subject to initial conditions, associated with the moment of impact when time, t , equals zero, $t=0$:

$$X = 0$$

$$Y = 0$$

$$\theta = \theta_0$$

$$\partial X / \partial t = \partial X_0 / \partial t$$

$$\partial Y / \partial t = \partial Y_0 / \partial t$$

$$\partial \theta / \partial t = \partial \theta_0 / \partial t$$

Where:

$$\theta_0 = \text{impact angle (varies with time, } t)$$

$$\partial Y_0 / \partial t = (2gh)^{0.5}$$

$$h = \text{drop height}$$

Each of the above differential equations requires a continuously updated value of the force, F , reflecting both crush depth and package orientation, or:

$$F = \zeta(\delta_y, \theta)$$

This continuously updated value of the force, F , is supplied to the integration process by means of a two dimensional linear interpolation of crush depth, δ_y , and orientation angle, θ . The tabular data used in this interpolation consist of a series of complete force-deflection relations for separate orientation angles developed via the CYDROP-PC computer program, described in Section 2.10.1.1.3. These force deflections data are input to the program via a second order Lagrangian interpolation routine. The deflection, δ_y , is expressed in terms of problem variables as:

$$\delta_y = (.5L + T_p)(\sin \theta - \sin \theta_0) + r(\cos \theta - \cos \theta_0) - Y$$

The foregoing analysis process for evaluating impacts at oblique orientations was consolidated in a Nuclear Packaging developed computer program, OBLIQUE-PC. OBLIQUE-PC integrates the equations of motion for each value of orientation angle versus time, until maximum values are found for crush force, crush deformation, shear, and body bending moment. At each incremental time step (incremental crush deformation), impact limiter attachment moments are computed, scanned for maximum values and output. By sweeping through a series of initial orientation angles, the maximum values of all internal loads are found. At each specified initial orientation angle, a solution is realized when all internal forces, moments, and deflection have reached a maximum value. Note that these internal forces, moments, and deflections do not necessarily happen at the same instantaneous angle, θ .

2.10.1.2.1 Impact Limiter Attachment Loads

For most orientations and crush depths, the impact limiter crush force is transmitted to the cask body in direct compression, hence, the forces transmitted to the circumferential impact limiter attachments are near zero. This is not true for near vertical and near horizontal orientations of the package, at very modest crush deformations and crush forces. In these very limited situations, the center of pressure of the crush force can lie beyond the outer extremities of the cask body and exert a resultant moment force upon the impact limiter attachments. Significantly, these moments exist only for very modest crush deformations and crush forces, regardless of orientation angle. This is because larger crush deformations move the center of pressure toward the cask body. At maximum crush depth and maximum crush force, for all angles of orientation, there are no impact limiter attachment moments because the impact limiter interface forces are all direct compression. The near vertical and near horizontal orientations, where attachment moments exist, are illustrated in Figure 2.10.1-5.

The diagram illustrates the effect of the angle of inclination on the stability of a masonry structure. It is divided into two parts: **NEAR VERTICAL** and **NEAR HORIZONTAL**.

In the **NEAR VERTICAL** case, the structure is shown as a rectangular block tilted at a small angle. The resultant force P acts vertically upwards from the base. The base is divided into three equal parts, each of width δ . The resultant force P acts within the middle part of the base, indicating stability.

In the **NEAR HORIZONTAL** case, the structure is shown as a rectangular block tilted at a large angle. The resultant force P acts vertically upwards from the base. The base is divided into three equal parts, each of width δ . The resultant force P acts outside the middle part of the base, indicating instability.

The diagram illustrates a mechanical system. A cylinder of radius r rotates about a horizontal axis. A block of mass m is in contact with the cylinder at point A . The block is constrained to move vertically along a guide. The distance from the axis of rotation to the point of contact is $2r$. The block is also in contact with a horizontal surface at point B . The distance from the axis of rotation to the point B is $2r$. The block is shown in two positions, one at the top and one at the bottom, with the angle of rotation indicated by θ . The diagram also shows the forces acting on the block, including the normal force N from the cylinder and the weight mg .

The impact limiter attachment moment is:

$$M = F(e_a) \text{ or } F(e_o)$$

Where:

e_a = moment arm about adjacent corner

e_o = moment arm about opposite corner

The location of the crush force can be approximated as the centroid of the crush footprint area. This approximation is consistently conservative. Specifically, for both near vertical and near horizontal orientations, foam strain-hardening effects tend to move the center of force from the geometric center of the crush footprint toward the cask body. In both instances, this tendency reduces the actual moment arm of the crush force to less than that predicted by the location of the crush footprint centroid. The moment arm, as defined by crush footprint geometry, is derived below. The location of the center of pressure relative to the opposite and adjacent corners of the cask body can be obtained from the geometry as illustrated in Figure 2.10.1-6.

Where:

$$q = r_o(\sin \theta)$$

$$f = g(\cos \theta)$$

$$g = T_b + (a - c)(\cos \theta)$$

$$e^- = x - f$$

The location of the center of pressure, measured from a normal to the crush plane passed through the intercept of package center line and body base plane (Point c), is:

$$e^- = x - [T_b + (a - c)(\cos \theta)](\cos \theta)$$

The moment arms, e_o and e_a , representing the distance from the center of pressure to the corners of the cask body, are thus given as:

$$e_o = -(\epsilon + q) = \text{moment arm about opposite corner}$$

$$e_a = (\epsilon - q) = \text{moment arm about adjacent corner}$$

Sign convention for these arms is such that the moment, $F(e_o)$, produces a clockwise (separation) moment about the opposite corner and moment, $F(e_a)$, produces a counter-clockwise (separation) moment about the adjacent corner. In other words, a positive moment must be resisted by impact limiter attachment bolts whereas a negative moment implies that the center of pressure is totally resisted by compressive interface forces and there are no attachment bolt loads.

In summary, the forces between the impact limiter and body have been derived in terms of package geometry and three problem variables: orientation angle, crush force, and crush deformation.

2.10.1.2.2 Maximum Package Loads and Locations

The package is idealized as a beam impacting on the lower end. The equations of motion are formed and used to define station-wise accelerations. These accelerations, in conjunction with the unit mass of the package, form forces which vary along the length of the package. When integrated, these forces provide a complete definition of internal thrusts, shears, and moments for the package as a function of total impact force and orientation angle.

For a planar rigid body system, the behavior is totally defined by a solution of the three equations of equilibrium written at the center of gravity of the rigid body. In the preceding figure, local coordinates are defined at the center of gravity, with axes parallel and normal to the beam. The end impact force is resolved into components parallel to these local axes. Summation of forces at the center of gravity leads to three rigid body equations of motion:

$$\text{Sum of Lateral Forces} - M(\partial^2 Y / \partial t^2) = F(\sin \alpha)$$

$$\text{Sum of Longitudinal Forces} - M(\partial^2 X / \partial t^2) = F(\cos \alpha)$$

$$\text{Sum of Moments} - I(\partial^2 \theta / \partial t^2) = -FL(\sin \alpha)/2$$

Where:

$M = \rho L$, the mass of the body

$I = \rho L^3/12$, the mass moment of inertia of the body

ρ = the mass per unit length of the body

α = the orientation angle with respect to vertical

Note that the mass moment of inertia term given above is valid only for infinitely slender beams of mass. A more accurate mass moment approximation is provided by the equation:

$$I = \rho L(3R^2 + L^2)/12$$

Where:

R = the radius of the cylindrical cask

This increased mass moment of inertia demonstrably decreases the internal moment. Thus, all moments calculated using the slender body approximation are conservative in proportion to the degree with which the cask is not slender. A very squat cask would have an internal moment predicted by OBLIQUE-PC considerably higher than reality.

Thrust, shear and moment load distribution along the length of the cask is illustrated in Figure 2.10.1-7. Corresponding equations for these load distributions and maximum/minimum values are given in Table 2.10.1-1 located below Figure 2.10.1-7.

FIGURE 2.10.1-7
Cask Thrust, Shear and Moment Distributions

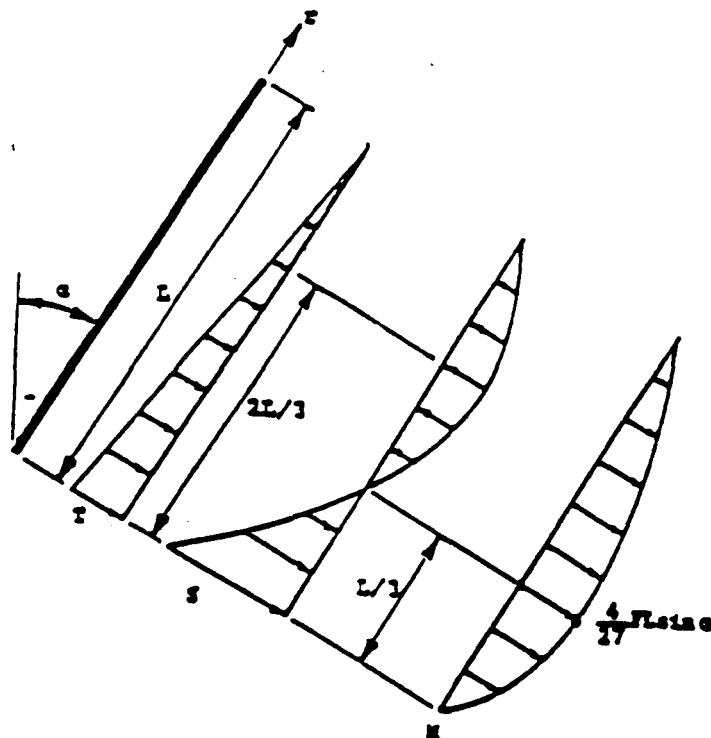


TABLE 2.10.1-1
Cask Thrust, Shear and Moment Equations

PARAMETER	EQUATION	MAXIMUM	MINIMUM
Thrust	$F(\cos \alpha)[1 - (r/L)]$	$F(\cos \alpha)$ ($r = 0$)	0 ($r = L$)
Shear	$F(\sin \alpha)[3r^2 - 4Lr + L^2]/L^2$	$F(\sin \alpha)$ ($r = 0$)	0 ($r = L/3, L$)
Moment	$F(\sin \alpha)[r^3 - 2Lr^2 + L^2r]/L^2$	$4FL(\sin \alpha)/27$ ($r = L/3$)	0 ($r = 0, L$)

APPENDIX 2.10.2

NUCLEAR PACKAGING PROPRIETARY DROP PROGRAM EVALUATIONS

2.10.2.1 Introduction

This appendix discusses the methodology utilized to obtain basic low and high temperature polyurethane foam stress/strain characteristics for input to the dynamic analysis drop programs discussed in Appendix 2.10.1. This appendix also contains the results of the corresponding drop program evaluations for the NuPac 140-B Cask.

2.10.2.2 Basic Polyurethane Foam Stress/Strain Characteristics

In order to perform drop program evaluations, impact limiter material stress/strain or load/deflection characteristics must be known. The situation is complicated for the NuPac 140-B Cask impact limiters because of large thermal gradients throughout the polyurethane foam material.

Section 3.0 of the preliminary design report shows that such thermal gradients will range 110 F to 278 F for normal conditions, and 101 F to 225 F for accident conditions (prior to the hypothetical accident thermal event). Prior Nuclear Packaging proprietary polyurethane foam stress/strain data only existed up to 180 F; thus, further testing up to 300 F was accomplished as reported in Reference (5).

Nuclear Packaging proprietary data, and data from Reference (5) utilized to obtain the specific stress/strain characteristics for the NuPac 140-B Cask impact limiters are presented in Tables 2.10.2-1 thru 2.10.2-4. Tables 2.10.2-1 and 2.10.2-2 are compressive stress data for 10 and 15 PCF perpendicular to the rise, and Tables 2.10.2-3 and 2.10.2-4 are similar data parallel to the rise.

Drop evaluations consider cold and hot conditions based upon 10 CFR 71 worst case requirements to determine corresponding worst case loads on the cask. These basic hot and cold conditions are listed below:

- o Cold conditions assume -20 F for the environment and the drop evaluations are performed assuming a uniform temperature of -20 F throughout the impact limiter foam material. Usually, the highest inertia loads are transmitted to the cask for this condition.
- o Hot conditions assume 100 F for the environment and the drop evaluations are performed based on actual equilibrium temperature gradients within the impact limiters. These gradients are determined from thermal analysis for 10 CFR 71

TABLE 2.10.2-1
LAST-A-FOAM ® FR-3700
COMPRESSIVE STRENGTH PERPENDICULAR TO RISE
Strength in (PSI)

DENSITY=10.00 Lbs/Ft³

STRAIN %	10	20	30	40	50	60	65	70	75	80
-20 °F	437	447	472	521	621	850	1088	1409	2111	3639
75 °F	330	333	355	395	474	649	822	1105	1629	2700
100 °F	279	291	314	354	425	590	748	1018	1518	2604
140 °F	246	258	277	313	378	516	655	884	1313	2260
180 °F	210	220	241	273	329	454	572	773	1148	1934
220 °F	195	198	213	241	285	389	490	664	994	1810
260 °F	150	150	166	190	227	311	393	531	791	1433
300 °F	58	65	79	100	128	185	239	330	509	898

TABLE 2.10.2-2
LAST-A-FOAM ® FR-3700
COMPRESSIVE STRENGTH PERPENDICULAR TO RISE
Strength in (PSI)

DENSITY=15.00 Lbs/Ft³

STRAIN %	10	20	30	40	50	60	65	70	75	80
-20 °F	929	971	1031	1158	1408	1891	2524	3290	4899	9338
75 °F	898	728	778	873	1081	1538	1990	2701	3984	6584
100 °F	587	623	677	772	952	1371	1774	2486	3710	6618
140 °F	512	543	591	678	833	1192	1549	2146	3270	5696
180 °F	437	486	510	584	722	1031	1334	1848	2829	4924
220 °F	388	403	439	505	611	865	1119	1558	2414	5002
260 °F	291	302	334	386	473	669	867	1203	1868	3920
300 °F	86	112	142	183	243	368	478	677	1108	2061

TABLE 2.10.2-3
LAST-A-FOAM ® FR-3700
COMPRESSIVE STRENGTH PARALLEL TO RISE
Strength in (PSI)

DENSITY=10.00 Lbs/Ft³

STRAIN %	10	20	30	40	50	60	65	70	75	80
-20 °F	507	487	517	564	688	877	1101	1481	2088	3474
75 °F	388	370	381	434	510	681	888	1094	1688	2724
100 °F	338	328	348	388	460	617	784	1047	1537	2601
140 °F	281	278	298	333	397	529	682	917	1371	2354
180 °F	258	244	264	297	353	488	587	784	1135	1844
220 °F	238	222	237	288	314	401	508	668	978	1684
260 °F	178	184	180	208	244	334	423	572	778	1383
300 °F	74	78	92	115	147	203	259	353	531	988

TABLE 2.10.2-4
LAST-A-FOAM ® FR-3700
COMPRESSIVE STRENGTH PARALLEL TO RISE
Strength in (PSI)

DENSITY=15.00 Lbs/Ft³

STRAIN %	10	20	30	40	50	60	65	70	75	80
-20 °F	983	983	1048	1178	1438	2027	2867	3438	5030	8810
75 °F	723	737	762	900	1103	1564	2026	2680	4069	6738
100 °F	624	644	698	787	987	1412	1841	2568	3907	6860
140 °F	522	548	587	662	843	1198	1572	2183	3391	6101
180 °F	487	478	522	588	737	1038	1337	1848	2821	4782
220 °F	422	427	462	527	641	880	1138	1557	2398	4608
260 °F	307	308	342	387	487	714	831	1318	1871	3620
300 °F	97	119	149	192	258	380	501	711	1147	2441

normal and accident conditions. Hot conditions definitely result in highest impact limiter deflections, which designs their overall dimensions. Usually, the lowest inertia loads are transmitted to the cask for hot conditions; however, sometimes the highest inertia loads are transmitted.

For cold conditions, higher foam stress characteristics result in higher inertia loads transmitted to the cask. Thus, the higher -20 F stress/strain foam characteristics were found from Tables 2.10.2-3 and 2.10.2-4 for data from tests parallel to the rise. Only Table 2.10.2-4 was utilized because the 15 PCF foam has higher stress characteristics than the 10 PCF material.

For hot conditions, lower foam stress characteristics result in higher impact limiter deflections. Thus, lower stress/strain foam characteristics were found from Tables 2.10.2-1 and 2.10.2-2 for data from tests perpendicular to the rise. Both tables were utilized because a foam density as low as 12.5 PCF can be expected based on manufacturing tolerances; and the lower foam density has lower stress characteristics. The manufacturing target range is 12.5 to 15 PCF; however, the primary goal is foam crush strength, not density. The foam density can be outside of this range as long as quality assurance compressive stress/strain data taken during the foam pouring process and tested at room temperature (70 to 75 F) falls between the ranges given in Table 2.10.2-7.

For hot conditions, 12.5 PCF stress/strain data were obtained by interpolation of Tables 2.10.2-1, at 10 PCF, and 2.10.2-2, at 15 PCF. The interpolation utilized normal and accident condition analytical thermal gradients obtained from the NuPac 140-B Cask thermal analysis of Section 3.0. All of the resulting data for both cold and hot conditions, including the thermal gradients, are listed in Tables 2.10.2-5 and 2.10.2-6.

TABLE 2.10.2-5
FOAM LOAD/DEFLECTION (STRESS/STRAIN) DESIGN PERFORMANCE DATA
(Normal Conditions)

O ALL TEMPERATURES IN DEG F, STRAIN & DEFLECTION, AND STRESS PSI

<u>ALL DROPS</u>			<u>END DROP</u>		<u>SIDE DROP</u>		<u>CORNER DROP</u>	
(15 PCF, MAX G'S)			(12.5 PCF, MAX DEFLECTION, MIN G'S)					
STRAIN	TEMP	STRESS	TEMP	STRESS	TEMP	STRESS	TEMP	STRESS
.10	-20	963	131	391	131	391	116	412
.20	-20	983	151	384	145	392	120	428
.30	-20	1045	168	393	159	407	128	452
.40	-20	1176	185	422	173	441	139	496
.50	-20	1435	202	483	187	512	152	581
.60	-20	2027	217	636	200	685	169	773
.70	-20	3436	232	1038	214	1141	186	1280
.80	-20	8610	247	2914	228	3261	209	3413

TABLE 2.10.2-6
FOAM LOAD/DEFLECTION (STRESS/STRAIN) DESIGN PERFORMANCE DATA
(Accident Conditions)

O ALL TEMPERATURES IN DEG F, STRAIN & DEFLECTION, AND STRESS PSI

<u>ALL DROPS</u>			<u>END DROP</u>		<u>SIDE DROP</u>		<u>CORNER DROP</u>	
(15 PCF, MAX G'S)			(12.5 PCF, MAX DEFLECTION, MIN G'S)					
STRAIN	TEMP	STRESS	TEMP	STRESS	TEMP	STRESS	TEMP	STRESS
.10	-20	963	115	413	110	420	101	432
.20	-20	983	128	417	120	428	104	451
.30	-20	1045	141	433	131	448	110	480
.40	-20	1176	153	473	142	491	118	532
.50	-20	1435	167	551	152	581	128	630
.60	-20	2027	178	748	162	793	139	857
.70	-20	3436	190	1260	172	1351	150	1464
.80	-20	8610	200	3418	183	3427	163	3713

TABLE 2.10.2-7

IMPACT LIMITER MANUFACTURING QUALITY ASSURANCE CONSIDERATIONS

- O DENSITY CRITERIA: 15 PCF max (weight consideration)
- O SAMPLING METHOD: A one cubic foot box sample will be taken from the liquid mixture of chemicals poured for each impact limiter
- O SAMPLE TEST CRITERIA:
 - o Density check shows 15 PCF max
 - o Compressive Stress/Strain at room temperature to fall within the following range

<u>STRAIN</u>	<u>STRESS</u>	<u>STRAIN</u>	<u>STRESS</u>	<u>STRAIN</u>	<u>STRESS</u>
.10	739/546	.50	1098/811	.70	2634/1868
.20	744/550	.60	1538/1132	.75	2634/1868
.30	798/590	.65	1983/1466	.80	6818/4545
.40	902/667				

2.10.2.3 Drop Program Evaluation Results

Drop analyses were performed utilizing the proprietary Nuclear Packaging computer program codes described in Appendix 2.10.1, and the impact limiter polyurethane foam design stress/strain data previously described in Section 2.10.2.2. All analyses assumed that cask weight included impact limiters and baskets; however, cold condition analyses for maximum inertia loads assumed all fuel assemblies were absent. Cask weights were obtained from Section 2.2 and are given below along with corresponding rotational mass moments of inertia:

- o Maximum Wt = 203,159 lbs (Fully loaded, 52 BWR assemblies)
- o Maximum I_{cg} = 2,794,955 lb-sec²-in (1)
- o Minimum Wt = 168,688 lbs (Empty, with PWR basket)
- o Minimum I_{cg} = 2,534,631 lb-sec²-in (1)

- (1) Details of the cask mass moment of inertia calculations will be included in the final design report and/or the SAR.

2.10.2.3.1 Cold Conditions - Maximum Inertia Loads Transmitted to Cask

Table 2.10.2-8 below gives the maximum inertia loads transmitted to the cask for all drop orientations. Cold conditions with minimum cask weight were assumed, and found to be worst case; and, for verification, corresponding hot condition (maximum cask weight, 203,159 lbs) inertia loads are listed in parenthesis. Inertia loads for cold conditions with maximum cask weight will fall somewhere in between.

TABLE 2.10.2-8
DROP ANALYSIS RESULTS FOR 140-B CASK MAXIMUM INERTIA LOADS

<u>DROP ORIENTATION</u>	<u>NORMAL (G's)</u>	<u>ACCIDENT (G's)</u>
Enddrop (90 Deg)	24.7 (18.2)	49.1 (29.1)
85	11.1 (7.3)	73.5 (29.7)
80	7.8 (6.0)	56.2 (34.1)
75	7.3 (5.7)	47.8 (30.9)
70	7.4 (5.5)	42.1 (26.7)
CG/Corner 65	7.0 (5.2)	37.5 (23.3)
60	6.2 (4.7)	33.4 (21.0)
55	5.5 (4.2)	29.6 (18.5)
50	4.9 (3.7)	26.8 (16.3)
45	4.3 (3.2)	24.1 (14.9)
40	3.9 (2.9)	22.1 (13.8)
35	3.5 (2.6)	20.8 (13.1)
30	3.3 (2.4)	19.9 (12.5)
25	3.2 (2.4)	19.8 (12.6)
20	2.6 (2.1)	19.4 (12.8)
15	2.7 (2.4)	19.9 (13.1)
10	3.1 (2.8)	22.1 (12.6)
5	4.0 (3.6)	26.6 (15.9)
Sidedrop (0 Deg)	16.3 (8.4)	55.6 (54.1)

Tables 2.10.2-9 thru 2.10.2-16 give drop analysis computer print-outs corresponding to the data in Table 2.10.2-8. The computer codes EYDROP-PC, CYDROP-PC, OBLIQUE-PC, and SYDROP-PC were utilized to produce the data for these tables. The maximum moment about the impact limiter was found for the analytical results given in Table 2.10.2-15.

TABLE 2.10.2-9
ENDDROP (90 Deg) NORMAL CONDITION OUTPUT

PROGRAM EYDROP-PC, VERSION 1.0.0, DATE 06 23/89

123456789V123456789V123456789V123456789V123456789V

COLD CASE, 15 PCF, 140-B, NORMAL (MIN WT, NO PAYLOAD, PWR BASKET)

168688.	103.	0.	40.	1.
.25	12.00	.25		
17				
.00	.00			
.05	963.00			
.10	963.00			
.15	973.00			
.20	983.00			
.25	1012.00			
.30	1045.00			
.35	1110.00			
.40	1176.00			
.45	1306.00			
.50	1435.00			
.55	1731.00			
.60	2027.00			
.65	2597.00			
.70	3436.00			
.75	6023.00			
.80	8610.00			

PROGRAM EYDROP-PC, VERSION 1.0.0, DATE 06/23/89

COLD CASE, 15 PCF, 140-B, NORMAL (MIN WT, NO PAYLOAD, PWR BASKET)

PACKAGE WEIGHT = 168688. (LBS)
 PACKAGE DIAMETER = 103.00 (IN)
 HOLE DIAMETER = .00 (IN)
 OVERPACK DEPTH = 40.00 (IN)
 DROP HEIGHT = 1.00 (FT)

		+++ IMPACT +++		++++++ ENERGY ++++++		
CRUSH DEPTH (IN)	STRAIN	FORCE (LBS)	ACCEL. (G)	KINETIC (IN-LB)	STRAIN (IN-LB)	RATIO (SE/KE)
.25	.006	1002999.	5.9	2066428.	125375.	.061
.50	.013	2005999.	11.9	2108600.	501500.	.238
.75	.019	3008998.	17.8	2150772.	1128374.	.525
1.00	.025	4011998.	23.8	2192944.	2005999.	.915
1.04	.026	4187362.	24.8	2200317.	2200317.	1.000

TABLE 2.10.2-10
ENDDROP (90 Deg) ACCIDENT CONDITION OUTPUT

PROGRAM EYDROP-PC, VERSION 1.0.0, DATE 06/23/89

COLD CASE, 15 PCF, 140-B, ACCIDENT (MIN WT, NO PAYLOAD, PWR BASKET)

PACKAGE WEIGHT = 168688. (LBS)
 PACKAGE DIAMETER = 103.00 (IN)
 HOLE DIAMETER = .00 (IN)
 OVERPACK DEPTH = 40.00 (IN)
 DROP HEIGHT = 30.00 (FT)

CRUSH DEPTH (IN)	STRAIN	+++ IMPACT +++		+++++++ ENERGY ++++++		
		FORCE (LBS)	ACCEL. (G)	KINETIC (IN-LB)	STRAIN (IN-LB)	RATIO (SE/KE)
.50	.013	2005999.	11.9	60812020.	501500.	.008
1.00	.025	4011998.	23.8	60896370.	2005999.	.033
1.50	.038	6017997.	35.7	60980710.	4513498.	.074
2.00	.050	8023996.	47.6	61065060.	8023996.	.131
2.50	.063	8016184.	47.5	61149400.	12034040.	.197
3.00	.075	8013580.	47.5	61233740.	16041480.	.262
3.50	.087	8016184.	47.5	61318090.	20048920.	.327
4.00	.100	8023996.	47.6	61402430.	24058970.	.392
4.50	.112	8053484.	47.7	61486780.	28078340.	.457
5.00	.125	8075552.	47.9	61571120.	32110590.	.522
5.50	.138	8092672.	48.0	61655460.	36152650.	.586
6.00	.150	8107318.	48.1	61739810.	40202640.	.651
6.50	.162	8106472.	48.1	61824150.	44256090.	.716
7.00	.175	8121379.	48.1	61908500.	48313060.	.780
7.50	.188	8150087.	48.3	61992840.	52380920.	.845
8.00	.200	8190641.	48.6	62077180.	56466100.	.910
8.50	.213	8260685.	49.0	62161530.	60578940.	.975
8.69	.217	8284549.	49.1	62194420.	62194420.	1.000

TABLE 2.10.2-11
SIDEDROP (0 Deg) NORMAL CONDITION OUTPUT

COLD CASE, 15 PCF, 140-B, NORMAL (MIN WT, NO PAYLOAD, PWR BASKET)
 168688. 121. 128. 77. 1.

17
 .00 .00
 .05 963.00
 .10 963.00
 .15 973.00
 .20 983.00
 .25 1012.00
 .30 1045.00
 .35 1110.00
 .40 1176.00
 .45 1306.00
 .50 1435.00
 .55 1731.00
 .60 2027.00
 .65 2597.00
 .70 3436.00
 .75 6023.00
 .80 8610.00
 150. .50 12.0 .50

PROGRAM SYDROP-PC, VERSION 1.0.0, DATE 06/23/89

COLD CASE, 15 PCF, 140-B, NORMAL (MIN WT, NO PAYLOAD, PWR BASKET)

PACKAGE WEIGHT = 168688. (LBS)
 IMPACT LMTR EFF LENGTH = 121.00 (IN)
 IMPACT LMTR EXT DIA = 128.00 (IN)
 CASK DIAMETER ADJ LMTR = 77.40 (IN)
 DROP HEIGHT = 1.00 (FT)

CRUSH DEPTH (IN)	STRAIN AT BOT	+++ IMPACT +++		++++++ ENERGY ++++++		
		FORCE (LBS)	ACCEL. (G)	KINETIC (IN-LB)	STRAIN (IN-LB)	RATIO (SE/KE)
.50	.020	489428.	2.9	2108600.	122357.	.058
1.00	.040	1379095.	8.2	2192944.	589488.	.269
1.50	.059	2367806.	14.0	2277288.	1526213.	.670
1.80	.071	2757232.	16.3	2328277.	2328277.	1.000

TABLE 2.10.2-12
SIDEDROP (0 Deg) ACCIDENT CONDITION OUTPUT

PROGRAM SYDROP-PC, VERSION 1.0.0, DATE 06/23/89

COLD CASE, 15 PCF, 140-B, ACCIDENT (MIN WT, NO PAYLOAD, PWR BASKET)

PACKAGE WEIGHT = 168688. (LBS)
 IMPACT LMTR EFF LENGTH = 121.00 (IN)
 IMPACT LMTR EXT DIA = 128.00 (IN)
 CASK DIAMETER ADJ LMTR = 77.40 (IN)
 DROP HEIGHT = 30.00 (FT)

CRUSH DEPTH (IN)	STRAIN AT BOT	+++ IMPACT +++		++++++ ENERGY ++++++		
		FORCE (LBS)	ACCEL. (G)	KINETIC (IN-LB)	STRAIN (IN-LB)	RATIO (SE/KE)
.50	.020	489428.	2.9	60812020.	122357.	.002
1.00	.040	1379095.	8.2	60896370.	589488.	.010
1.50	.059	2367806.	14.0	60980710.	1526213.	.025
2.00	.079	3011984.	17.9	61065060.	2871160.	.047
2.50	.099	3528354.	20.9	61149400.	4506245.	.074
3.00	.119	3978749.	23.6	61233740.	6383020.	.104
3.50	.138	4382829.	26.0	61318090.	8473414.	.138
4.00	.158	4749919.	28.2	61402430.	10756600.	.175
4.50	.178	5088197.	30.2	61486780.	13216130.	.215
5.00	.198	5410053.	32.1	61571120.	15840690.	.257
5.50	.217	5725108.	33.9	61655460.	18624480.	.302
6.00	.237	6031239.	35.8	61739810.	21563570.	.349
6.50	.257	6325052.	37.5	61824150.	24652640.	.399
7.00	.277	6608206.	39.2	61908500.	27885950.	.450
7.50	.296	6893944.	40.9	61992840.	31261490.	.504
8.00	.316	7193701.	42.6	62077180.	34783400.	.560
8.50	.336	7498007.	44.4	62161530.	38456330.	.619
9.00	.356	7796520.	46.2	62245870.	42279960.	.679
9.50	.375	8086594.	47.9	62330220.	46250740.	.742
10.00	.395	8393006.	49.8	62414560.	50370640.	.807
10.50	.415	8740696.	51.8	62498900.	54654070.	.874
11.00	.435	9109492.	54.0	62583250.	59116620.	.945
11.38	.450	9384989.	55.6	62647380.	62647380.	1.000

TABLE 2.10.2-13
OBLIQUE DROPS (5 to 85 Deg) NORMAL CONDITION OUTPUT

PROGRAM OBLIQUE-PC, VERSION 1.0.0, DATE 06/23/89

COLD CASE, 15 PCF, 140-B, NORMAL (MIN WT, NO PAYLOAD, PWR BASKET)

PACKAGE GEOMETRY-

LENGTH	-	203.500
RADIUS	-	38.750
OVERPACK LENGTH	-	70.000
OVERPACK SIDE THICKNESS	-	25.300
OVERPACK BOTTOM THICKNESS	-	40.000

PACKAGE MASS PROPERTIES-

MASS	-	436.600
MASS MOMENT OF INERTIA	-	2534631.000
GRAVITATIONAL CONSTANT	-	386.400

SOLUTION CHARACTERISTICS-

IMPACT VELOCITY (YDOT)	-	-96.300
(XDOT)	-	.000
(THETADOT)	-	.000
FRICTION COEFFICIENT	-	.000
ESTIMATED CRUSH DEPTH	-	20.000

THETA0 (DEG)	FMAX (LBF)	ACCL (Gs)	SHEAR (LBF)	THRUST (LBF)	MOMENT (IN-LB)	DEFLECT- ION(IN)	CLEAR- ANCE(IN)
85.00	1870829.	11.1	152853.	1864644.	4608241.	2.58	39.33
80.00	1315051.	7.8	216865.	1297204.	6538085.	4.06	39.55
75.00	1231811.	7.3	308496.	1192589.	9300588.	5.74	39.32
70.00	1252940.	7.4	425696.	1178407.	12833940.	7.36	38.86
65.00	1187401.	7.0	508594.	1072965.	15333180.	8.61	38.38
60.00	1054304.	6.2	541383.	904689.	16321680.	9.19	38.12
55.00	927756.	5.5	549302.	747661.	16560450.	9.35	37.86
50.00	820558.	4.9	542894.	615290.	16367240.	9.25	37.48
45.00	729110.	4.3	529846.	500930.	15973860.	9.03	36.84
40.00	654991.	3.9	513100.	407449.	15469010.	8.69	35.99
35.00	597946.	3.5	498483.	330772.	15028340.	8.32	34.87
30.00	557046.	3.3	488895.	267707.	14739280.	7.93	33.44
25.00	542017.	3.2	495726.	219326.	14945220.	7.56	31.71
20.00	430537.	2.6	407436.	139539.	12283450.	5.88	30.96
15.00	453490.	2.7	439582.	111968.	13252600.	4.16	30.17
10.00	530251.	3.1	523090.	87215.	15770190.	3.37	28.12
5.00	674375.	4.0	672243.	54252.	20266870.	2.54	25.85

TABLE 2.10.2-14
OBLIQUE DROPS (5 to 85 Deg) ACCIDENT CONDITION OUTPUT

PROGRAM OBLIQUE-PC, VERSION 1.0.0, DATE 06/23/89

COLD CASE, 15 PCF, 140-B, ACCIDENT (MIN WT, NO PAYLOAD, PWR BASKET)

PACKAGE GEOMETRY-

LENGTH	-	203.500
RADIUS	-	38.750
OVERPACK LENGTH	-	70.000
OVERPACK SIDE THICKNESS	-	25.300
OVERPACK BOTTOM THICKNESS	-	40.000

PACKAGE MASS PROPERTIES-

MASS	-	436.600
MASS MOMENT OF INERTIA	-	2534631.000
GRAVITATIONAL CONSTANT	-	386.400

SOLUTION CHARACTERISTICS-

IMPACT VELOCITY (YDOT)	-	-527.450
(XDOT)	-	.000
(THETADOT)	-	.000
FRICTION COEFFICIENT	-	.000
ESTIMATED CRUSH DEPTH	-	20.000

THETA0 (DEG)	FMAX (LBF)	ACCL (Gs)	SHEAR (LBF)	THRUST (LBF)	MOMENT (IN-LB)	DEFLECT- ION(IN)	CLEAR- ANCE(IN)
85.00	12395510.	73.5	1023117.	12353210.	30845080.	10.93	31.02
80.00	9480974.	56.2	1600155.	9344965.	48241710.	15.90	27.80
75.00	8056263.	47.8	2119339.	7772502.	63894140.	20.00	25.25
70.00	7109330.	42.1	2557495.	6633385.	77103730.	23.14	23.28
65.00	6320235.	37.5	2865769.	5633182.	86397620.	25.28	21.84
60.00	5637108.	33.4	3054618.	4737752.	92091090.	26.50	20.83
55.00	4989452.	29.6	3095698.	3912964.	93329570.	26.83	20.24
50.00	4515563.	26.8	3128754.	3260902.	94326140.	26.53	19.84
45.00	4057704.	24.1	3057418.	2675746.	92175500.	25.78	19.59
40.00	3732153.	22.1	3019056.	2214190.	91018940.	24.71	19.27
35.00	3506210.	20.8	3001780.	1840194.	90498100.	23.40	18.92
30.00	3354977.	19.9	3003186.	1522729.	90540500.	21.92	18.47
25.00	3333904.	19.8	3093078.	1267742.	93250560.	20.17	18.11
20.00	3278172.	19.4	3131643.	992906.	94413250.	18.08	17.85
15.00	3350336.	19.9	3270203.	751297.	98590550.	15.71	17.58
10.00	3727851.	22.1	3691948.	544059.	111305400.	13.17	17.36
5.00	4490947.	26.6	4486132.	283416.	135248600.	12.52	14.56

TABLE 2.10.2-15
OBLIQUE DROPS (5 to 85 Deg) ACCIDENT CONDITION OUTPUT
(MAXIMUM MOMENTS ABOUT IMPACT LIMITER)

PROGRAM OBLIQUE-PC, VERSION 1.0.0, DATE 06/23/89

++++ OVERPACK SEPARATION MOMENT EVALUATION +++++

COLD CASE, 15 PCF, 140-B, ACCIDENT (MIN WT, NO PAYLOAD, PWR BASKET)

INITIAL VELOCITY = -527.450

ANGLE	CRUSH DEPTH	FORCE MAJ.	+++ DG.	CRUSH WIDTH	ELLIPSE XBAR	+++ AREA	C/L DIST TO C.P. CORNER	MOMTS ABOUT CORNER ADJNCT	OPPOSIT (MOC)
(A)			(C)				(E) (Q)	(MAC)	
85.0	2.0	1302501.	64.3	22.5	51.1523.	47.1 38.6	11096210.	0.	
85.0	10.9	12397120.	64.3	132.8	0.12932.	-2.8 38.6	0.	0.	
80.0	2.9	908075.	65.0	17.2	55.1029.	46.4 38.2	7469567.	0.	
80.0	15.9	9485487.	65.0	95.5	12.10300.	6.2 38.2	0.	0.	
75.0	4.1	708529.	66.3	16.4	57.946.	42.8 37.4	3841178.	0.	
75.0	20.0	8070154.	66.4	78.7	22.8248.	12.0 37.4	0.	0.	
70.0	4.7	553189.	68.2	14.5	59.785.	39.5 36.4	1723057.	0.	
70.0	23.2	7132567.	68.7	68.9	29.6935.	14.6 36.1	0.	0.	
65.0	3.8	221634.	70.7	9.8	65.432.	37.0 35.1	421546.	0.	
65.0	25.4	6344406.	72.0	62.5	36.6034.	15.5 34.5	0.	0.	
60.0	1.9	37193.	74.0	4.3	71.125.	34.0 33.6	14809.	0.	
60.0	26.6	5654683.	76.4	58.1	42.5370.	15.3 32.5	0.	0.	
55.0	26.9	5003624.	81.9	55.2	50.4875.	14.3 30.3	0.	0.	
55.0	26.9	5003624.	81.9	55.2	50.4875.	14.3 30.3	0.	0.	
50.0	26.5	4519801.	89.2	53.1	58.4482.	12.7 27.8	0.	0.	
50.0	26.5	4519801.	89.2	53.1	58.4482.	12.7 27.8	0.	0.	
45.0	25.8	4061178.	97.9	52.1	67.4207.	10.8 25.4	0.	0.	
45.0	25.8	4061178.	97.9	52.1	67.4207.	10.8 25.4	0.	0.	
40.0	24.7	3729403.	109.6	52.1	79.4012.	8.5 22.6	0.	0.	
40.0	24.7	3729403.	109.6	52.1	79.4012.	8.5 22.6	0.	0.	
35.0	23.3	3495055.	125.1	53.1	94.3899.	6.2 19.8	0.	0.	
35.0	23.3	3495055.	125.1	53.1	94.3899.	6.2 19.8	0.	0.	
30.0	21.9	3344213.	145.3	55.3	113.3873.	4.1 17.1	0.	0.	
30.0	21.9	3344213.	145.3	55.3	113.3873.	4.1 17.1	0.	0.	
25.0	20.1	3321495.	174.2	58.9	139.3915.	2.3 14.2	0.	-54992600.	
25.0	.0	0.	151.6	.0	152.0.	-9.2 16.4	0.	0.	
20.0	18.1	3273798.	218.7	64.5	180.4037.	1.1 11.3	0.	-40869100.	
20.0	1.9	83082.	187.3	5.9	184.126.	-14.0 13.2	0.	63403.	
15.0	15.7	3343621.	299.6	75.1	254.4323.	1.1 8.3	0.	-31506580.	
15.0	4.5	496569.	248.0	18.2	237.588.	-16.0 10.0	0.	2982824.	
10.0	13.2	3725168.	469.0	97.3	404.4423.	-.6 5.3	0.	-17353050.	
10.0	5.2	885181.	370.7	30.6	352.1046.	-17.0 6.7	0.	9101936.	
5.0	12.5	4490151.	1419.4	277.8	1176.5015.	-3.4 1.7	0.	7252496.	
5.0	3.4	942483.	738.3	39.1	715.1079.	-18.9 3.4	0.	14650410.	

MAXIMUM MOMENT ABOUT ADJACENT CORNER = 11096210.
 MAXIMUM MOMENT ABOUT OPPOSITE CORNER = 14650410.

TABLE 2.10.2-16
OBLIQUE DROPS (5 to 85 Deg) ACCIDENT CONDITION OUTPUT
(Cold Condition Force/Deflection Data From CYDROP-PC)

(85.0 Deg)

.75	40.00	.75				
128684.	712398.	1626827.	2215672.	2966541.	3895847.	4906814.
5968849.	7059259.	8163296.	9293518.	10487280.	11497880.	12186190.
12501950.	12526050.	12554290.	12586380.	12625640.	12675900.	12738560.
12811520.	12890080.	12976220.	13078800.	13202430.	13350730.	13504100.
13667000.	13853620.	14088380.	14375570.	14695670.	15045700.	15443640.
15912280.	16466910.	17116230.	17913290.	18804920.	19813650.	20972070.
22426310.	24298850.	26947260.	30554320.	35039520.	40454410.	46897840.
54089660.	61860940.	70045300.	78318980.			

(80.0 Deg)

.75	40.00	.75				
44919.	251456.	666945.	925451.	1192654.	1508956.	1905988.
2355149.	2830154.	3314803.	3825994.	4359195.	4894201.	5438434.
5984308.	6540386.	7097917.	7666664.	8230399.	8788798.	9363782.
9959280.	10575690.	11112960.	11575990.	12003640.	12402290.	12763520.
13056490.	13222390.	13331760.	13463640.	13616530.	13792250.	13996210.
14224370.	14481430.	14775060.	15118870.	15512610.	15967760.	16506130.
17128520.	17817990.	18633960.	19617450.	20981490.	22686730.	24841800.
27510170.	30723730.	34471210.	38837030.			

(75.0 Deg)

.75	40.00	.75				
24471.	137473.	343151.	474108.	624520.	814885.	1054438.
1328512.	1613083.	1910108.	2217997.	2538908.	2869654.	3205281.
3552748.	3907256.	4267381.	4655022.	5027212.	5402876.	5782788.
6166662.	6563647.	6963759.	7374949.	7784605.	8194745.	8607119.
9030651.	9459995.	9898620.	10361390.	10838880.	11278350.	11701700.
12123900.	12551870.	12992030.	13429700.	13889900.	14366450.	14857390.
15359720.	15854090.	16325460.	16876300.	17524770.	18326400.	19243140.
20491580.	21959670.	23804050.	26019520.			

(70.0 Deg)

.75	40.00	.75				
7118.	43366.	122834.	220567.	350389.	510781.	691660.
883762.	1083625.	1292261.	1510929.	1737793.	1975335.	2217411.
2464937.	2718720.	2982510.	3261062.	3532732.	3809263.	4090705.
4376774.	4685076.	4983449.	5288900.	5597375.	5909600.	6226667.
6558659.	6893239.	7227610.	7578063.	7929365.	8296216.	8677202.
9061992.	9452066.	9875962.	10302230.	10751460.	11243470.	11748770.
12244140.	12772440.	13324480.	13910720.	14554320.	15267320.	16060340.
17047050.	18123050.	19385060.	20856600.			

TABLE 2.10.2-16 (CONTINUED)
OBLIQUE DROPS (5 to 85 Deg) ACCIDENT CONDITION OUTPUT
(Cold Condition Force/Deflection Data From CYDROP-PC)

(65.0 Deg)

.75	40.00	.75				
3834.	21862.	60765.	125918.	218343.	334170.	466493.
610646.	765676.	930521.	1101170.	1282914.	1466473.	1657052.
1856702.	2055950.	2262609.	2489274.	2708763.	2929651.	3158109.
3411716.	3647219.	3888860.	4134996.	4384620.	4638249.	4907870.
5174718.	5447339.	5726924.	6011847.	6311082.	6614163.	6925526.
7246304.	7586151.	7932758.	8297229.	8671332.	9071997.	9477775.
9920300.	10381090.	10865630.	11383830.	11945900.	12565430.	13258660.
13990760.	14881510.	15902540.	17107750.			

(60.0 Deg)

.75	40.00	.75				
3674.	20888.	57709.	117175.	192583.	279616.	376491.
482341.	600474.	729011.	868106.	1016818.	1166875.	1330876.
1489703.	1667351.	1835165.	2011050.	2202709.	2385432.	2575031.
2780912.	2983922.	3186516.	3396528.	3622185.	3839843.	4067559.
4301380.	4540465.	4784168.	5043443.	5313401.	5576809.	5849454.
6132290.	6430271.	6733981.	7049551.	7384729.	7730850.	8092569.
8478211.	8883947.	9310960.	9771058.	10258600.	10778000.	11379690.
12042630.	12864350.	13816710.	14862090.			

(55.0 Deg)

.75	40.00	.75				
3260.	18442.	50823.	102732.	168629.	244957.	329772.
421944.	521285.	627411.	740763.	864577.	998130.	1129205.
1275504.	1430935.	1576073.	1729231.	1893701.	2056964.	2230646.
2403113.	2579097.	2755915.	2949487.	3137703.	3333610.	3532099.
3737386.	3958710.	4173085.	4397739.	4625364.	4868523.	5126122.
5385583.	5659171.	5934739.	6234086.	6544278.	6858489.	7185413.
7539781.	7913030.	8307414.	8729464.	9176732.	9664030.	10215040.
10839620.	11587630.	12439240.	13383040.			

(50.0 Deg)

.75	40.00	.75				
2752.	15581.	42982.	87756.	146301.	215059.	292382.
377315.	469252.	567099.	670569.	780983.	899166.	1015563.
1147973.	1286668.	1418207.	1567061.	1707853.	1851858.	2014391.
2167179.	2323420.	2498540.	2665100.	2835671.	3015972.	3202969.
3388428.	3580582.	3783740.	3995220.	4205085.	4424907.	4646347.
4885612.	5139125.	5394256.	5662966.	5940865.	6229324.	6539635.
6887326.	7235489.	7599749.	7994143.	8422497.	8864800.	9344535.
9899292.	10564550.	11343860.	12248350.			

TABLE 2.10.2-16 (CONTINUED)
OBLIQUE DROPS (5 to 85 Deg) ACCIDENT CONDITION OUTPUT
(Cold Condition Force/Deflection Data From CYDROP-PC)

(45.0 Deg)
 .75 40.00 .75

2389.	13545.	37408.	76861.	130430.	194106.	266161.
345576.	431822.	523908.	622156.	724990.	835724.	948985.
1072729.	1203607.	1327555.	1467304.	1600557.	1748781.	1888921.
2032048.	2191683.	2343750.	2500154.	2673740.	2839890.	3010589.
3188989.	3377356.	3563159.	3756464.	3960798.	4169274.	4378433.
4597486.	4821872.	5058050.	5311906.	5588909.	5864874.	6154317.
6460283.	6794291.	7178729.	7553504.	7953311.	8376082.	8844258.
9354306.	9942667.	10659190.	11503360.			

(40.0 Deg)
 .75 40.00 .75

2120.	12033.	33275.	68556.	118654.	179564.	249091.
326189.	409852.	499868.	595514.	696611.	802905.	915897.
1031436.	1159295.	1294087.	1422708.	1564640.	1699129.	1838296.
1989544.	2133774.	2286206.	2445804.	2605306.	2769280.	2943658.
3119757.	3297595.	3476837.	3664053.	3862327.	4061052.	4274897.
4490221.	4709305.	4937724.	5177575.	5428483.	5702612.	5985348.
6283640.	6639567.	6997398.	7404601.	7794584.	8216739.	8679770.
9197102.	9799220.	10564940.	11500440.			

(35.0 Deg)
 .75 40.00 .75

1919.	10911.	30222.	62368.	109398.	169232.	238578.
315919.	400564.	491546.	588970.	691569.	799459.	912714.
1033876.	1159756.	1292475.	1429101.	1572774.	1712080.	1864875.
2011168.	2172119.	2324868.	2484176.	2651065.	2815999.	2983294.
3158120.	3346182.	3534225.	3722238.	3913494.	4115038.	4330846.
4551509.	4775045.	5007748.	5259630.	5517158.	5789945.	6103429.
6429152.	6772415.	7152318.	7570749.	7993599.	8452649.	8964798.
9570884.	10362600.	11331630.	12490290.			

(30.0 Deg)
 .75 40.00 .75

1769.	10077.	27969.	57839.	101815.	161133.	232295.
312973.	401783.	498110.	600836.	709952.	826962.	951264.
1082743.	1221091.	1363917.	1512346.	1662988.	1814835.	1971051.
2132701.	2295167.	2468452.	2637712.	2811135.	2982493.	3169671.
3349379.	3539307.	3731854.	3944695.	4154338.	4367927.	4594234.
4826564.	5073184.	5336867.	5631333.	5932841.	6243281.	6588058.
6943949.	7350723.	7758624.	8206809.	8693203.	9248575.	9984088.
10952980.	12108570.	13213240.	14506780.			

TABLE 2.10.2-16 (CONTINUED)
OBLIQUE DROPS (5 to 85 Deg) ACCIDENT CONDITION OUTPUT
(Cold Condition Force/Deflection Data From CYDROP-PC)

(25.0 Deg)

.75	40.00	.75				
1660.	9478.	26375.	54698.	96572.	154026.	227974.
316765.	418160.	531937.	655506.	787489.	926481.	1074704.
1226071.	1384288.	1547263.	1713668.	1882019.	2057561.	2229857.
2407280.	2594778.	2779556.	2967815.	3162423.	3355476.	3553054.
3766585.	3975045.	4188885.	4411268.	4654729.	4912589.	5180550.
5461036.	5746034.	6052024.	6380026.	6728615.	7067166.	7421217.
7785258.	8218618.	8699736.	9376257.	10298230.	11115210.	12075580.
13337930.	14860340.	16603430.	18689150.			

(20.0 Deg)

.75	40.00	.75				
12100.	55362.	1060.	149528.	203448.	269392.	349607.
446602.	561931.	692874.	835664.	989161.	1152287.	1324748.
1505917.	1688419.	1871563.	2058202.	2248199.	2441661.	2638519.
2839390.	3045687.	3256569.	3473278.	3701962.	3932180.	4167284.
4435663.	4721765.	4990472.	5275375.	5534563.	5812274.	6069314.
6359883.	6608987.	6931633.	7237167.	7647542.	8084163.	8609610.
9145728.	9848944.	10709710.	11995300.	13705130.	15593930.	17276350.
19276030.	21638700.	24290160.	27218780.			

(15.0 Deg)

.75	40.00	.75				
16513.	93239.	198111.	304461.	407110.	492197.	580428.
679228.	793204.	928580.	1094851.	1299409.	1521588.	1743222.
1962374.	2180746.	2408842.	2637991.	2874867.	3116959.	3365478.
3620296.	3917127.	4209796.	4475810.	4719302.	4976742.	5197173.
5409923.	5653389.	5904928.	6133016.	6389825.	6691927.	7011058.
7342206.	7714756.	8158953.	8761665.	9718293.	10657660.	11672790.
12959070.	14470440.	16867160.	19924670.	22974560.	25586980.	28266150.
31194840.	34551000.	37578020.	40480200.			

(10.0 Deg)

.75	40.00	.75				
26089.	147317.	286637.	440827.	615287.	761742.	894460.
1039335.	1213923.	1448729.	1731567.	2019560.	2302216.	2598150.
2896637.	3238023.	3569155.	3846276.	4078391.	4302856.	4549925.
4746299.	4954980.	5164467.	5381450.	5601908.	5808817.	6092120.
6315991.	6643691.	7003318.	7312753.	7728733.	8158692.	8928996.
10008690.	11064910.	12266620.	13734040.	15603200.	18023420.	22360200.
25963670.	29391500.	32785530.	36053520.	39414160.	40068800.	41598370.
43560530.	47179580.	51757560.	57854960.			

TABLE 2.10.2-16 (CONTINUED)
OBLIQUE DROPS (5 to 85 Deg) ACCIDENT CONDITION OUTPUT
(Cold Condition Force/Deflection Data From CYDROP-PC)

(5.0 Deg)

	.75	40.00	.75				
56637.	305899.	560378.	838702.	1052439.	1326304.	1684958.	
2094178.	2546480.	2914474.	3196864.	3466743.	3726545.	3944060.	
4164785.	4380308.	4548942.	4802176.	5002961.	5160044.	5419547.	
5629321.	5905818.	6160384.	6449526.	6845940.	7285234.	7597565.	
8070321.	8593270.	9376442.	10054530.	11989310.	15755420.	19188460.	
22873530.	26077100.	27146960.	29442120.	32592830.	35433620.	38140450.	
40806170.	43375160.	46196820.	44524700.	50366480.	51013120.	53443980.	
55850130.	57912660.	59771540.	63035340.				

2.10.2.3.2 Hot Conditions - Maximum Impact Limiter Deflections

Table 2.10.2-17 below gives the maximum impact limiter deflections and clearances for all cask drop orientations. Hot environment accident conditions with maximum cask weight were assumed, and found to be worst case. For verification, corresponding normal condition deflections and clearances are listed in parenthesis.

TABLE 2.10.2-17
DROP ANALYSIS RESULTS FOR 140-B CASK MAXIMUM
IMPACT LIMITER DEFLECTIONS

<u>DROP ORIENTATION</u>			<u>DEFLECTION (IN)</u>		<u>CLEARANCE (IN)</u>	
Enddrop	(90 Deg)	(1)	29.19	(2.64)	10.81	(37.36)
	85		20.41	(4.03)	21.93	(37.78)
	80		25.60	(6.54)	18.56	(36.96)
	75		30.67	(9.06)	14.95	(35.95)
	70		34.87	(11.33)	11.83	(34.91)
CG/Corner	65		37.61	(13.05)	9.64	(33.98)
	60		38.78	(13.95)	8.50	(33.37)
	55		39.20	(14.14)	7.61	(33.02)
	50		38.81	(13.93)	7.14	(32.63)
	45		37.51	(13.44)	7.23	(32.21)
	40		35.84	(12.80)	7.35	(31.62)
	35		33.78	(12.09)	7.68	(30.76)
	30		31.65	(11.34)	7.76	(29.71)
	25		28.98	(10.50)	8.41	(28.48)
	20		26.10	(8.76)	8.97	(27.75)
	15		22.84	(6.70)	9.73	(27.20)
	10		19.79	(5.19)	9.88	(26.01)
Sidedrop	5		20.19	(4.09)	6.77	(23.89)
	(0 Deg)	(2)	19.39	(3.31)	5.91	(21.99)

NOTES: (1) Assumes impact limiter only partially effective with the effective area being only that backed by the cask.

(2) Assumes impact limiter 86 & partially effective and neglects any additional resistance when crush plane passes through the soft outer corners into the higher density foam.

Tables 2.10.2-18 thru 2.10.2-22 give drop analysis computer print-outs corresponding to the data in Table 2.10.2-17. The computer codes EYDROP-PC, CYDROP-PC, OBLIQUE-PC, and SYDROP-PC were utilized to produce the data for these tables.

TABLE 2.10.2-18
ENDDROP (90 Deg) ACCIDENT CONDITION OUTPUT

PROGRAM EYDROP-PC, VERSION 1.0.0, DATE 06/23/89

123456789V123456789V123456789V123456789V123456789V
 HOT CASE, 12.5 PCF, 140-B, ACCIDENT, 52 BWR PAYLOAD (MAX DEFLECTION)

203159. 77. 0. 40. 30.
 .50 32.00 .50

17
 .00 .00
 .05 413.00
 .10 413.00
 .15 415.00
 .20 417.00
 .25 425.00
 .30 433.00
 .35 453.00
 .40 473.00
 .45 512.00
 .50 551.00
 .55 650.00
 .60 748.00
 .65 1004.00
 .70 1260.00
 .75 2339.00
 .80 3418.00

PROGRAM EYDROP-PC, VERSION 1.0.0, DATE 06/23/89

HOT CASE, 12.5 PCF, 140-B, ACCIDENT, 52 BWR PAYLOAD (MAX DEFLECTION)

PACKAGE WEIGHT = 203159. (LBS)
 PACKAGE DIAMETER = 77.40 (IN)
 HOLE DIAMETER = .00 (IN)
 OVERPACK DEPTH = 40.00 (IN)
 DROP HEIGHT = 30.00 (FT)

CRUSH DEPTH (IN)	STRAIN	+++ IMPACT +++		++++++ ENERGY ++++++		
		FORCE (LBS)	ACCEL. (G)	KINETIC (IN-LB)	STRAIN (IN-LB)	RATIO (SE/KE)
.50	.013	485805.	2.4	73238820.	121451.	.002
1.00	.025	971610.	4.8	73340400.	485805.	.007
1.50	.038	1457415.	7.2	73441980.	1093061.	.015
2.00	.050	1943220.	9.6	73543560.	1943220.	.026
2.50	.063	1942338.	9.6	73645140.	2914609.	.040
3.00	.075	1942044.	9.6	73746720.	3885704.	.053
3.50	.087	1942338.	9.6	73848300.	4856800.	.066
4.00	.100	1943220.	9.6	73949870.	5828189.	.079
4.50	.112	1947116.	9.6	74051460.	6800773.	.092

TABLE 2.10.2-18 (CONTINUED)
ENDDROP (90 Deg) ACCIDENT CONDITION OUTPUT

5.00	.125	1949689.	9.6	74153030.	7774975.	.105
5.50	.138	1951380.	9.6	74254620.	8750242.	.118
6.00	.150	1952630.	9.6	74356190.	9726245.	.131
6.50	.162	1950792.	9.6	74457780.	10702100.	.144
7.00	.175	1952042.	9.6	74559350.	11677810.	.157
7.50	.188	1955938.	9.6	74660940.	12654810.	.169
8.00	.200	1962040.	9.7	74762510.	13634300.	.182
8.50	.213	1974538.	9.7	74864090.	14618450.	.195
9.00	.225	1984390.	9.8	74965670.	15608180.	.208
9.50	.237	1992477.	9.8	75067250.	16602390.	.221
10.00	.250	1999682.	9.8	75168830.	17600430.	.234
10.50	.262	2000711.	9.8	75270410.	18600530.	.247
11.00	.275	2007915.	9.9	75371990.	19602690.	.260
11.50	.287	2020413.	9.9	75473570.	20609770.	.273
12.00	.300	2037323.	10.0	75575150.	21624200.	.286
12.50	.313	2065737.	10.2	75676730.	22649970.	.299
13.00	.325	2089961.	10.3	75778300.	23688890.	.313
13.50	.338	2111392.	10.4	75879890.	24739230.	.326
14.00	.350	2131425.	10.5	75981460.	25799930.	.340
14.50	.363	2141681.	10.5	76083050.	26868210.	.353
15.00	.375	2161715.	10.6	76184620.	27944060.	.367
15.50	.387	2190129.	10.8	76286210.	29032020.	.381
16.00	.400	2225528.	11.0	76387780.	30135930.	.395
16.50	.412	2286842.	11.3	76489360.	31264020.	.409
17.00	.425	2334922.	11.5	76590940.	32419460.	.423
17.50	.438	2374181.	11.7	76692520.	33596740.	.438
18.00	.450	2409028.	11.9	76794100.	34792540.	.453
18.50	.463	2412741.	11.9	76895680.	35997990.	.468
19.00	.475	2447551.	12.0	76997260.	37213060.	.483
19.50	.488	2508975.	12.3	77098840.	38452190.	.499
20.00	.500	2592528.	12.8	77200420.	39727570.	.515
20.50	.512	2750334.	13.5	77302000.	41063280.	.531
21.00	.525	2872778.	14.1	77403580.	42469060.	.549
21.50	.538	2971549.	14.6	77505160.	43930140.	.567
22.00	.550	3058336.	15.1	77606740.	45437610.	.585
22.50	.563	3063262.	15.1	77708320.	46968010.	.604
23.00	.575	3149498.	15.5	77809900.	48521200.	.624
23.50	.587	3305429.	16.3	77911480.	50134930.	.643
24.00	.600	3519439.	17.3	78013060.	51841150.	.665
24.50	.613	4032335.	19.8	78114630.	53700000.	.688
25.00	.625	4363717.	21.5	78216220.	55000000.	.714
25.50	.637	4574087.	22.5	78317790.	58000000.	.741
26.00	.650	4723954.	23.3	78419380.	60000000.	.770
26.50	.663	4450284.	21.9	78520950.	62600000.	.798
27.00	.675	4600151.	22.6	78622540.	64940000.	.826
27.50	.688	5113047.	25.2	78724110.	67371540.	.856
28.00	.700	5928468.	29.2	78825700.	70131910.	.890
28.50	.712	7197675.	35.4	78927270.	73413450.	.930
29.00	.725	8466889.	41.7	79028850.	77329590.	.978
29.19	.730	8952019.	44.1	79067680.	79067680.	1.000

TABLE 2.10.2-19
SIDEDROP (0 Deg) ACCIDENT CONDITION OUTPUT

PROGRAM SYDROP-PC, VERSION 1.0.0, DATE 06/23/89

123456789V123456789V123456789V123456789V123456789V

HOT CASE, 12.5 PCF, 140-B, ACCIDENT, 52 BWR PAYLOAD (MAX DEFLECTION)

203159. 121. 128. 77. 30.
 17

.00	.00
.05	420.00
.10	420.00
.15	424.00
.20	428.00
.25	438.00
.30	448.00
.35	470.00
.40	491.00
.45	536.00
.50	581.00
.55	687.00
.60	793.00
.65	1072.00
.70	1351.00
.75	2389.00
.80	3427.00
150. .50	21.0 .50

PROGRAM SYDROP-PC, VERSION 1.0.0, DATE 06/23/89

HOT CASE, 12.5 PCF, 140-B, ACCIDENT, 52 BWR PAYLOAD (MAX DEFLECTION)

PACKAGE WEIGHT	= 203159. (LBS)
IMPACT LMTR EFF LENGTH	= 121.00 (IN)
IMPACT LMTR EXT DIA	= 128.00 (IN)
CASK DIAMETER ADJ LMTR	= 77.40 (IN)
DROP HEIGHT	= 30.00 (FT)

CRUSH DEPTH (IN)	STRAIN AT BOT	+++ IMPACT +++		++++++ ENERGY ++++++		
		FORCE (LBS)	ACCEL. (G)	KINETIC (IN-LB)	STRAIN (IN-LB)	RATIO (SE/KE)
.50	.020	213458.	1.1	73238820.	53364.	.001
1.00	.040	601475.	3.0	73340400.	257098.	.004
1.50	.059	1032712.	5.1	73441980.	665644.	.009
2.00	.079	1313724.	6.5	73543560.	1252253.	.017
2.50	.099	1538924.	7.6	73645140.	1965415.	.027
3.00	.119	1734963.	8.5	73746720.	2783887.	.038
3.50	.138	1910791.	9.4	73848300.	3695325.	.050
4.00	.158	2070728.	10.2	73949870.	4690705.	.063

TABLE 2.10.2-19 (CONTINUED)
SILLEDROP (0 Deg) ACCIDENT CONDITION OUTPUT

4.50	.178	2218246.	10.9	74051460.	5762949.	.078
5.00	.198	2358060.	11.6	74153030.	6907025.	.093
5.50	.217	2494005.	12.3	74254620.	8120041.	.109
6.00	.237	2625086.	12.9	74356190.	9399814.	.126
6.50	.257	2749786.	13.5	74457780.	10743530.	.144
7.00	.277	2868760.	14.1	74559350.	12148170.	.163
7.50	.296	2987566.	14.7	74660940.	13612250.	.182
8.00	.316	3110873.	15.3	74762510.	15136860.	.202
8.50	.336	3234824.	15.9	74864090.	16723280.	.223
9.00	.356	3355250.	16.5	74965670.	18370800.	.245
9.50	.375	3471154.	17.1	75067250.	20077400.	.267
10.00	.395	3591993.	17.7	75168830.	21843190.	.291
10.50	.415	3727199.	18.3	75270410.	23672990.	.315
11.00	.435	3869489.	19.0	75371990.	25572160.	.339
11.50	.455	4008870.	19.7	75473570.	27541750.	.365
12.00	.474	4140542.	20.4	75575150.	29579100.	.391
12.50	.494	4288956.	21.1	75676730.	31686470.	.419
13.00	.514	4475632.	22.0	75778300.	33877620.	.447
13.50	.534	4691687.	23.1	75879890.	36169450.	.477
14.00	.553	4927797.	24.3	75981460.	38574320.	.508
14.50	.573	5145651.	25.3	76083050.	41092680.	.540
15.00	.593	5424414.	26.7	76184620.	43735200.	.574
15.50	.613	5815432.	28.6	76286210.	46545160.	.610
16.00	.632	6247374.	30.8	76387780.	49560860.	.649
16.50	.652	6650317.	32.7	76489360.	52785280.	.690
17.00	.672	6930514.	34.1	76590940.	56180490.	.734
17.50	.692	7382792.	36.3	76692520.	59758820.	.779
18.00	.711	8158566.	40.2	76794100.	63644160.	.829
18.50	.731	9247404.	45.5	76895680.	67995650.	.884
19.00	.751	10293230.	50.7	76997260.	72880810.	.947
19.39	.766	10995640.	54.1	77076630.	77076630.	1.000

TABLE 2.10.2-20
OBLIQUE DROPS (5 to 85 Deg) ACCIDENT CONDITION OUTPUT

PROGRAM OBLIQUE-PC, VERSION 1.0.0, DATE 06/23/89

HOT CASE, 12.5 PCF, 140-B, ACCIDENT, 52 BWR PAYLOAD (MAX DEFLECTION)

PACKAGE GEOMETRY-

LENGTH	-	203.500
RADIUS	-	38.750
OVERPACK LENGTH	-	70.000
OVERPACK SIDE THICKNESS	-	25.300
OVERPACK BOTTOM THICKNESS	-	40.000

PACKAGE MASS PROPERTIES-

MASS	-	525.800
MASS MOMENT OF INERTIA	-	2794955.000
GRAVITATIONAL CONSTANT	-	386.400

SOLUTION CHARACTERISTICS-

IMPACT VELOCITY (YDOT)	-	-527.450
(XDOT)	-	.000
(THETADOT)	-	.000
FRICTION COEFFICIENT	-	.000
ESTIMATED CRUSH DEPTH	-	20.000

THETA0 (DEG)	FMAX (LBF)	ACCL (Gs)	SHEAR (LBF)	THRUST (LBF)	MOMENT (IN-LB)	DEFLECT- ION(IN)	CLEAR- ANCE(IN)
85.00	6030153.	29.7	607563.	5999468.	18316900.	20.41	21.93
80.00	6922955.	34.1	1348167.	6790416.	40644740.	25.60	18.56
75.00	6284382.	30.9	1821575.	6014593.	54917100.	30.67	14.95
55.00	5427303.	26.7	2117799.	4997054.	63847720.	34.87	11.83
65.00	4734015.	23.3	2300626.	4137393.	69359620.	37.61	9.64
60.00	4258753.	21.0	2440138.	3490373.	73565630.	38.78	8.50
55.00	3752200.	18.5	2450603.	2848289.	73881150.	39.20	7.61
50.00	3301951.	16.3	2382126.	2303948.	71816670.	38.81	7.14
45.00	3021275.	14.9	2360292.	1918528.	71158420.	37.51	7.23
40.00	2800230.	13.8	2333471.	1597664.	70349830.	35.84	7.35
35.00	2661380.	13.1	2329592.	1340487.	70232880.	33.78	7.68
30.00	2536557.	12.5	2308703.	1100275.	69603120.	31.65	7.76
25.00	2554174.	12.6	2397715.	925613.	72286650.	28.98	8.41
20.00	2601403.	12.8	2503380.	748421.	75472260.	26.10	8.97
15.00	2669231.	13.1	2616916.	567771.	78895170.	22.84	9.73
10.00	2560543.	12.6	2542902.	353368.	76663790.	19.79	9.88
5.00	3237272.	15.9	3234422.	167141.	97511820.	20.19	6.77

TABLE 2.10.2-21
OBLIQUE DROPS (5 to 85 Deg) ACCIDENT CONDITION OUTPUT
(MAXIMUM MOMENTS ABOUT IMPACT LIMITER)

PROGRAM OBLIQUE-PC, VERSION 1.0.0, DATE 06/23/89

++++ OVERPACK SEPARATION MOMENT EVALUATION +++++

HOT CASE, 12.5 PCF, 140-B, ACCIDENT, 52 BWR PAYLOAD (MAX DEFLECTION)

INITIAL VELOCITY = -527.450

ANGLE	CRUSH DEPTH	FORCE	MAJ. (A)	CRUSH DG. (C)	ELLIPSE WIDTH (C)	AREA	C/L DIST TO C.P. CORNER (E) (Q)	MOMTS ABOUT CORNER ADJNCT (MAC)	OPPOSIT (MOC)
85.0	2.1	654822.	64.3	24.3	50.	1702.	46.1 38.6	4894248.	0.
85.0	16.8	5814335.	64.3	191.8	0.	12938.	-2.5 38.6	0.	0.
80.0	3.2	438010.	65.0	18.4	54.	1131.	45.7 38.2	3316283.	0.
80.0	25.7	6955448.	65.3	133.4	0.	13145.	-5.2 38.0	0.	0.
75.0	3.7	273865.	66.3	14.7	58.	809.	43.7 37.4	1722627.	0.
75.0	30.8	6326652.	67.0	110.2	7.	11861.	-.8 37.1	0.	0.
70.0	4.1	190571.	68.2	12.8	61.	648.	40.4 36.4	753780.	0.
70.0	35.0	5458797.	69.7	96.8	16.	10390.	4.1 35.6	0.	0.
65.0	3.4	78489.	70.7	8.9	65.	375.	37.4 35.1	178662.	0.
65.0	37.8	4757085.	73.5	88.3	23.	9283.	7.3 33.8	0.	0.
60.0	38.9	4274693.	78.5	82.5	31.	8407.	9.4 31.6	0.	0.
60.0	38.9	4274693.	78.5	82.5	31.	8407.	9.4 31.6	0.	0.
55.0	39.3	3759915.	85.1	79.2	39.	7814.	10.4 29.2	0.	0.
55.0	39.3	3759915.	85.1	79.2	39.	7814.	10.4 29.2	0.	0.
50.0	38.8	3305829.	93.2	77.8	48.	7416.	10.8 26.6	0.	0.
50.0	38.8	3305829.	93.2	77.8	48.	7416.	10.8 26.6	0.	0.
45.0	37.5	3020631.	103.5	77.1	58.	7074.	10.8 24.0	0.	0.
45.0	37.5	3020631.	103.5	77.1	58.	7074.	10.8 24.0	0.	0.
40.0	35.8	2793508.	117.2	78.2	71.	6885.	10.5 21.2	0.	0.
40.0	35.8	2793508.	117.2	78.2	71.	6885.	10.5 21.2	0.	0.
35.0	33.6	2647571.	134.9	80.5	88.	6789.	10.2 18.4	0.	0.
35.0	33.6	2647571.	134.9	80.5	88.	6789.	10.2 18.4	0.	0.
30.0	31.5	2517491.	159.5	85.6	107.	6666.	8.7 15.6	0.	0.
30.0	31.5	2517491.	159.5	85.6	107.	6666.	8.7 15.6	0.	0.
25.0	28.9	2545565.	189.8	91.0	132.	6449.	6.9 13.1	0.	0.
25.0	28.9	2545565.	189.8	91.0	132.	6449.	6.9 13.1	0.	0.
20.0	26.0	2594542.	241.1	101.7	173.	6254.	4.6 10.3	0.	-38644580.
20.0	1.9	35731.	187.3	5.8	184.	122.	-14.1 13.2	0.	28745.
15.0	22.8	2664933.	332.3	120.6	245.	6041.	2.1 7.5	0.	-25485100.
15.0	4.6	227435.	247.9	18.5	237.	606.	-15.9 10.0	0.	1336588.
10.0	19.8	2557706.	569.6	176.9	427.	5935.	-.8 4.4	0.	-9042285.
10.0	5.1	388111.	369.8	29.8	352.	1011.	-17.2 6.7	0.	4079155.
5.0	20.3	3259823.	1597.5	506.	11126.	6361.	-3.5 1.6	0.	6486710.
5.0	3.6	449734.	738.5	41.5	714.	1179.	-18.0 3.4	0.	6566211.

MAXIMUM MOMENT ABOUT ADJACENT CORNER = 4944550.
 MAXIMUM MOMENT ABOUT OPPOSITE CORNER = 6567916.

2.10.2.3.3 Hot And Cold Conditions - Maximum Impact Limiter Moments

Table 2.10.2-22 below gives the maximum impact limiter moments for design of attachment bolts. The normal condition worst case corresponds to the cold environment with maximum cask weight. The accident condition worst case corresponds to the cold environment with minimum cask weight. All of the data in Table 2.10.2-22 were obtained from the OBLIQUE-PC dynamic analysis computer code.

TABLE 2.10.2-22
DROP ANALYSIS RESULTS FOR 140-B CASK MAXIMUM
IMPACT LIMITER MOMENTS

	NORMAL (in-lbs)	ACCIDENT (in-lbs)
Cold, Min Wt.	12,903,160	14,650,410
Cold, Max Wt.	13,326,240	14,648,130
Hot, Max Wt.	6,490,408	6,567,916

APPENDIX 2.10.3

BUCKLING DESIGN CRITERIA

AND DETAILED EVALUATION

2.10.3 Buckling Design Criteria and Detailed Evaluation

The buckling criteria selected for use in analyzing the NuPac 140-B Cask is that presented in Code Case N-284 of the ASME Boiler and Pressure Vessel Code (Metal Containment Shell Buckling Design Methods, Reference (10)). By demonstrating that the requirements specified in N-284 are met, the buckling requirement of paragraph C.5, of Nuclear Regulatory Guide 7.6 (i.e., Buckling should not occur as the result of any normal or accident condition loading) is satisfied. Consistent with Regulatory Guide 7.6 philosophy, ASME Code Level A Service Limits correspond to Regulatory Guide 7.6 defined "normal conditions" and ASME Level D Service Limits correspond to "accident conditions" in making the buckling assessment.

The N-284 Code Case addresses both elastic and inelastic buckling and employs a factor of safety of 2.0 for Level A (normal) conditions and a factor of safety of 1.34 for Level D (accident) conditions. Interaction equations are employed which account for the effect of combined hoop compressive, axial compressive, and in-plane shear loadings. The basic analytic approach is summarized as follows.

1. Theoretical elastic buckling stresses are determined using classical theory.
2. Capacity reduction factors are applied which account for the difference between classical theory and predicted instability stresses for fabricated shells.
3. Plasticity reduction factors are applied for those cases where elastically determined buckling stresses are above the proportional limit.
4. Elastic and inelastic buckling checks which employ appropriate factors of safety and appropriate interaction equations are made using worst case applied compressive and in-plane shear stresses.

All terminology used herein is consistent with that in Code Case N-284, Reference (10).

2.10.3.1 Theoretical Elastic Buckling Stresses

The geometry and geometric parameters and the material property data required to carry out the buckling evaluation is presented in Table 2.10.3-1. Using this data, Tables 2.10.3-2 and 2.10.3-3 are developed. The basic equations used in the development of these two tables are directly available from -1712 of N-284, and are summarized below (Note: C_1 are geometric dependent parameters):

- a) For axial compression

$$\sigma_{\theta\theta L} = C_{\theta} E t / R = \text{theoretical axial buckling stress.}$$

Note: The theoretical elastic buckling stress for axial compression of cylinders is also directly applicable for the case of bending of cylinders. (Per Reference (39), pp. 230 and 235)

- (b) For external pressure

$$\sigma_{\theta\theta L} = \sigma_{rr L} = C_{\theta r} E t / R = \text{theoretical hoop buckling stress for case of no end pressure}$$

$$\sigma_{\theta\theta L} = C_{\theta h} E t / R = \text{theoretical hoop buckling stress for case with end pressure included}$$

- c) For shear

$$\sigma_{\phi\phi L} = C_{\phi} E t / R = \text{theoretical in-plane shear buckling stress.}$$

TABLE 2.10.3-1

Geometry, Geometric Parameters, and Material Data Used in Buckling Evaluations

Parameter	Outer Cask	
	Outer Shell	Inner Shell
R = radius(in)	33.875	29.063
t = thickness(in)	1.5	1.125
$(Rt)^{0.5}$	7.128	5.718
l_o = length(in)	180.5	180.5
$l_o = 2\pi R$ = circumf.(in)	212.8	182.6
$M_o = l_o / (Rt)^{0.5}$	25.32	31.57
$M_i = l_i / (Rt)^{0.5}$	29.85	31.93
M = lesser of M_o , M_i	25.32	31.57
μ = Poisson's Ratio	0.3	0.3
C_o = coefficient	0.605	0.605
$C_{oz} = 0.92 / (M_o - 1.17)$	0.03810	0.03026
$C_{oi} = 0.92 / (M_i - 0.636)$	0.03727	0.02974
C_{os} (see Note 1)	0.1483	0.1328

Notes:

- 1) $C_{os} = 4.82 (1 + 0.0239 M_o^3)^{0.5} / M_o^2$ if $1.5 \leq M_o < 26$
 $C_{os} = 0.746 / (M_o)^{0.5}$ if $26 \leq M_o < 8.69 R/t$

Values of Young's modulus (E) and yield stress (σ_y) are temperature dependent and are tabulated below for potential temperatures of interest.

Parameter	Temperature (°F)		
	-20	250	300
E(psi)	28.7(10) ⁶	27.3(10) ⁶	27.0(10) ⁶
σ_y (psi)	55,000	45,200	43,400

TABLE 2.10.3-2
Theoretical Elastic Buckling Stress Values
(General Temperature Independent Form)

Elastic Buckling Stress *	Shell	
	Outer Cask	
	Outer Shell	Inner Shell
$\sigma_{\theta eL}$ (axial)	0.026790 E	0.023419 E
$\sigma_{\theta eL} = \sigma_{reL}$ (hoop, without end pressure)	0.001687 E	0.001171 E
σ_{heL} (hoop, end pressure included)	0.001650 E	0.001151 E
$\sigma_{\phi eL}$ (shear)	0.006567 E	0.005141 E

* Elastic Buckling Stress Formulas are as follows per Appendix 2.10.3.1. The tabulated values are the resultant numerical values with modulus of elasticity, E, retained as a temperature dependent parameter.

$$\sigma_{\theta eL} = C_{\theta} Et/R$$

$$\sigma_{\theta eL} = \sigma_{reL} = C_{\theta r} Et/R$$

$$\sigma_{heL} = C_{\theta h} Et/R$$

$$\sigma_{\phi eL} = C_{\phi} Et/R$$

where E = modulus of elasticity

C_1 , t, R = geometric parameters

TABLE 2.10.3-3
Theoretical Elastic Buckling Stresses (psi) Versus Temperature for
Each Shell of 140-B Cask

Shell		-20°F	250°F	300°F
$\sigma_{\theta\theta L}$	Cask	768,873	731,367	723,330
	Outer Shell			
$\sigma_{\theta\theta L}$	Cask	672,125	639,339	632,313
	Inner Shell			
$\sigma_{\theta\theta L}$ - $\sigma_{\theta\theta L}$	Cask	48,417	46,055	45,549
	Outer Shell			
$\sigma_{\theta\theta L}$ - $\sigma_{\theta\theta L}$	Cask	33,608	31,968	31,617
	Inner Shell			
$\sigma_{\theta\theta L}$	Cask	47,355	45,045	44,550
	Outer Shell			
$\sigma_{\theta\theta L}$	Cask	33,034	31,422	31,077
	Inner Shell			
$\sigma_{\theta\theta L}$	Cask	188,473	179,279	177,309
	Outer Shell			
$\sigma_{\theta\theta L}$	Cask	147,547	140,349	138,807
	Inner Shell			

2.10.3.2 Capacity Reduction Factors

Capacity reduction factors which account for the difference between classical theory and predicted instability stresses for fabricated shells are directly determined using -1500 of Code Case N-284. For the NuPac 140-B Cask shell geometries, the basic governing equations are as follows.

- a) For axial compression
 $\alpha_{\phi L} = 1.0(10)^{-5}\sigma_y - 0.033$

Note: The capacity reduction factor for cylinders under axial compression is also employed for the case of bending of cylinders. This is slightly conservative based on an inspection of Figures 10-9 and 10-13 of Reference (39). In other words, advantage is not taken of the fact that the critical buckling stress for cylinders under axial compression is slightly less than the critical buckling stress for cylinders in bending.

- b) For hoop compression
 $\alpha_{\phi L} = 0.8$

- c) For shear
 $\alpha_{\phi L} = 0.8$

Direct application of these equations results in the following capacity reduction factors as a function of temperature.

Capacity Reduction Factor	Temperature (°F)		
	-20	250	300
$\alpha_{\phi L}$ (axial)	0.517	0.419	0.401
$\alpha_{\phi L}$ (hoop)	0.8	0.8	0.8
$\alpha_{\phi L}$ (shear)	0.8	0.8	0.8

In order to directly use the above defined capacity reduction factors, the tolerance requirements of NE-4220 must be met. Per NE-4221.1 and NE-4221.2, limits on "maximum difference in cross-sectional diameters" and "maximum deviation from true theoretical form for external pressure" must be met. Per drawing 2111-201, sheet 1, Flagnote 20, these requirements are met for both the cask outer shell and inner shell.

2.10.3.3 Plasticity Reduction Factors

Plasticity reduction factors for cases where the elastically determined buckling stresses are above the proportional limit are directly determined using -1600 of Code Case N-284. The magnitude of the reduction factor is directly related to the magnitude of the applied compressive or in-plane shear stress, σ_1 , as indicated by the applicable equations below.

a) For axial compression

$$\eta_a = 1.0 \text{ if } \sigma_a(\text{FS})/\sigma_y \leq 0.55$$

$$\eta_a = 0.18 / \{1 - 0.45 \sigma_y / [\sigma_a(\text{FS})]\} \text{ if } 0.55 < \sigma_a(\text{FS})/\sigma_y \leq 0.738$$

$$\eta_a = 1.31 - 1.15(\sigma_a(\text{FS})/\sigma_y) \text{ if } 0.738 < \sigma_a(\text{FS})/\sigma_y \leq 1.0$$

b) For hoop compression

$$\eta_h = 1.0 \text{ if } \sigma_h(\text{FS})/\sigma_y \leq 0.67$$

$$\eta_h = 2.53 - 2.29(\sigma_h(\text{FS})/\sigma_y) \text{ if } 0.67 < \sigma_h(\text{FS})/\sigma_y \leq 1.0$$

c) For shear

$$\eta_{ss} = 1.0 \text{ if } \sigma_{ss}(\text{FS})/\sigma_y \leq 0.48$$

$$\eta_{ss} = 0.1 / \{1 - 0.43\sigma_y / [\sigma_{ss}(\text{FS})]\} \text{ if } 0.48 < \sigma_{ss}(\text{FS})/\sigma_y \leq 0.6$$

As plasticity reduction factors depend directly on applied compressive or in-plane shear stress magnitudes, numerical values for reduction factors cannot be presented at this point in the evaluation.

2.10.3.4 Development of Upper Bound Magnitudes for Compressive Stresses And In Plane Shear Stresses

Per -1600 of Code Case N-284, as an upper limit, σ_i ($i = \phi$ or θ) must be less than σ_y divided by the applicable Factor of Safety. Similarly for shear, $\sigma_{\phi\theta}$ must be less than or equal to $0.6\sigma_y$ divided by the applicable Factor of Safety. With a factor of safety of 2.0 for normal conditions, and 1.34 for accident conditions, the maximum (upper bound) permitted values for elastically determined axial, σ_{ϕ} , or hoop, σ_{θ} , compressive stresses and for shear, $\sigma_{\phi\theta}$, are as follows. Under no circumstances can these upper bound limits be exceeded. However, satisfying these limits alone is not sufficient to demonstrate that buckling will not occur. As discussed in this Appendix, Section 2.10.3.5, interaction equations must also be satisfied.

		Temperature °F		
Load Condition		-20	250	300
Upper bound permitted compressive stress	Normal	27,500	22,600	21,700
σ_{ϕ} or σ_{θ} (psi)	Accident	41,045	33,731	32,388
Upper bound Permitted	Normal	16,500	13,560	13,020
In-Plane Shear Stress $\sigma_{\phi\theta}$ (psi)	Accident	24,627	20,239	19,433

2.10.3.5 Interaction Equations

Elastic and inelastic interaction equations must be satisfied for all states of compressive and in-plane shear stress. The interaction equations for cylindrical shells are directly available from -1713.1.1 and -1713.2.1 of Code Case N-284. Once a stress state is established for a specific shell, plasticity reduction factors can be determined and all appropriate interaction equations checked. Elastic interaction equations must be satisfied, and if any of the uniaxial critical stress values exceed the proportional limit of the fabricated material, the inelastic interaction equations must also be satisfied.

2.10.3.6 Stress Summary

Applied compressive membrane and in-plane shear stresses for each load condition and for each shell of the 140-B cask are summarized in Tables 2.10.3-4 and 2.10.3-5. Appropriate worst case combinations of these stresses are provided in Table 2.10.3-6 thus allowing plasticity reduction factors to be explicitly determined and interaction equations to be checked.

TABLE 2.10.3-4
Compressive Membrane and In-Plane Shear Stress Summary for
Cask - Inner Shell

<u>Load Condition</u>		σ_{axial}	σ_{hoop}	r	Reference Section
a	normal diff. expansion	234	704	0	2.6.1.2
fabrication (lead shrinkage)					
b	at -20°F	3,174	3,099	0	2.6.2
c	at 300°F	1,161	1,134	0	2.6.7.1(6)
(extrapolated)					
normal end drop + fabrication*					
d	at -20°F	7,195	3,099	0	2.6.7.1(6)
e	at 300°F	7,195	1,134	0	2.6.7.1(6)
normal end drop + fabrication**					
f	at -20°F	4,719	4,994	0	2.6.7.1(7)
g	at 300°F	4,719	2,744	0	2.6.7.1(7)
h	normal oblique drop (85° from horizontal)	(1)	(1)	(1)	2.6.7.2
normal oblique drop + fabrication					
i	at -20°F	(1)	(1)	(1)	2.6.7.2
j	at 300°F	(1)	(1)	(1)	2.6.7.2
normal side drop					
k	maximum moment location	6331	0	0	2.6.7.3(3)
l	maximum shear location	0	0	3,137	2.6.7.3(3)

(1) To be considered during final design.

TABLE 2.10.3-4 (Con't.)
Compressive Membrane and In-Plane Shear Stress Summary for
Cask - Inner Shell

<u>Load Condition</u>		σ_{axial}	σ_{hoop}	r	Reference Section
accident end drop + fabrication*					
m	at -20°F	14,318	3,099	0	2.7.1.1(6)
n	at 250°F	14,318	1,462	0	2.7.1.1(6)
accident end drop + fab.**					
o	at -20°F	9,391	5,058	0	2.7.1.1(7)
p	at 250°F	9,391	4,087	0	2.7.1.1(7)
accident oblique drop (85° from horizontal)					
q		26,928	0	3,900	2.7.1.2(4)
accident oblique drop + fab.*					
r	at -20°F	30,102	3,099	3,900	2.7.1.2(4)
s	at 250°F	28,425	1,462	3,900	2.7.1.1(4)
accident side drop					
t	maximum moment location	21,588	0	0	2.7.1.3(3)
u	maximum shear location	0	0	10,703	2.7.1.3(3)
pin punch on cask side					
v	maximum shear location	0	0	2,839	2.7.2
w	maximum moment location	6,821	0	0	2.7.2

* lead assumed to be initially shrunk onto inner shell

** lead assumed to be initially stress free

TABLE 2.10.3-5
Compressive Membrane and In-plane Shear Stress Summary for
Cask- Outer Shell

<u>Load Condition</u>		σ_{axial}	σ_{hoop}	r	Reference Section
a	normal differential expansion	0	1,164	0	2.6.1.2
normal end drop + fabrication*					
b	at -20°F	4,050	0	0	2.6.7.1(6)
c	at 300°F	4,050	0	0	2.6.7.1(6)
normal end drop + fabrication**					
d	at -20°F	3,036	4,365	0	2.6.7.1(7)
e	at 300°F	3,036	2,398	0	2.6.7.1(7)
normal oblique drop (85° from horizontal)					
f		(1)	(1)	(1)	2.6.7.2
normal side drop					
g	maximum moment location	7,399	0	0	2.6.7.3(2)
h	maximum shear location	0	0	3,137	2.6.7.3(3)
accident end drop + fabrication*					
i	at -20°F	8,060	0	0	2.7.1.1(6)
j	at 250°F	8,060	0	0	2.7.1.1(6)
accident end drop + fab.**					
k	at -20°F	6,042	0	0	2.7.1.1(7)
l	at 250°F	6,042	0	0	2.7.1.1(7)

(1) To be considered during final design.

TABLE 2.10.3-5 (Con't)
Compressive Membrane and In-plane Shear Stress Summary for
Cask- Outer Shell

<u>Load Condition</u>	σ_{axial}	σ_{hoop}	r	Reference Page
m accident oblique drop (85° from horizontal)	27,499	0	3,900	2.7.1.2(4)
accident side drop				
n maximum moment location	25,231	0	0	2.7.1.3(2)
o maximum shear location	0	0	10,703	2.7.1.3(3)
pin punch on cask side				
p maximum moment location	7,967	0	0	2.7.2
q maximum shear location	0	0	2,839	2.7.2

* lead assumed to be initially shrunk onto inner shell

** lead assumed to be initially stress free

TABLE 2.10.3-6
Worst Case Load Combinations
Cask - Inner Shell

Load Combination		Normal or Accident	Shell Temperature (°F)	σ_{θ} (psi)	σ_{ϕ} (psi)	σ_{xx} (psi)
(From Table 2.10.3-4)						
1	a + d	normal	-20	7,429	3,803	0
2	a + e	normal	300	7,429	1,838	0
3	a + f	normal	-20	4,953	5,698	0
4	a + g	normal	300	4,953	3,448	0
5	a + l	normal	300	234	704	3,137
6	a + r	accident	-20	30,336	3,803	3,900
7	a + s	accident	250	28,659	2,166	3,900
8	a + u	accident	250	234	704	10,703

Cask - Outer Shell

Load Combination		Normal or Accident	Shell Temperature	σ_{θ}	σ_{ϕ}	σ_{xx}
2.10.3-5)						
1	a + f	normal	300	(1)	(1)	(1)
2	a + g	normal	300	7,399	0	0
3	a + h	normal	300	0	1,164	3,137
4	a + j	accident	250	8,060	1,164	0
5	a + m	accident	250	27,499	1,164	3,900
6	a + o	accident	250	0	1,164	10,703

(1) To be considered during final design

2.10.3.7 Detailed Buckling Evaluations - Plasticity Reduction Factor Determinations and Interaction Equation Checks

By using the worst case load combinations available in Table 2.10.3-6 of this Appendix, Section 2.10.3.6, and the various data available in Sections 2.10.3.1 through 2.10.3.5, the methods outlined in code case N-284 can be directly applied to the NuPac 140-B Cask shells. Worst case load combinations will be taken as 7 for the inner shell and 5 for the outer shell (Table 2.10.3-6). The step by step procedure is that specified in paragraph -1800 of Code Case N-284.

Step 1 Stresses for each specified loading are per Table 2.10.3-4 and 2.10.3-5 herein. The specified loadings for load combinations 5 and 7 are normal differential expansion, accident oblique drop and fabrication.

Step 2 Combined stresses are per Table 2.10.3-6 herein. For load combination 5 and 7 the combined stresses are:

	<u>Outer Shell</u>	<u>Inner Shell</u>
σ_x	= 27,499 psi	= 28,659 psi
σ_y	= 1,164 psi	= 2,166 psi
σ_{xy}	= 3,900 psi	= 3,900 psi

Step 3 For accident conditions, the factor of Safety (FS) is 1.34. Multiplying the stress components by this Factor of Safety yields:

	<u>Outer Shell</u>	<u>Inner Shell</u>
FS(σ_x)	= 36,849 psi	= 38,403 psi
FS(σ_y)	= 1,560 psi	= 2,902 psi
FS(σ_{xy})	= 5,226 psi	= 5,226 psi

Step 4 Capacity reduction factors are available from Section 2.10.3.2 and are as follows for load combination 5 and 7 temperature of 250°F.

α_{xL}	= 0.419
α_{yL}	= 0.8
α_{xyL}	= 0.8

Step 5 Plasticity reduction factors are determined using the equations presented in Section 2.10.3.3 as follows, (σ_y available from Section 2.10.3.1).

	<u>Outer Shell</u>	<u>Inner Shell</u>
a) For axial compression		
$\sigma_a(\text{FS})/\sigma_y = 36849/45,200 =$	0.8152	0.8496
$\eta_a = 1.31 - 1.15 (0.8152) =$	0.3725	0.3329
b) For hoop compression		
$\sigma_h(\text{FS})/\sigma_y = 1560/55000 =$	0.0345	0.0528
$\eta_h =$	1.0	1.0
c) For shear		
$\sigma_{ss}(\text{FS})/\sigma_y = 5226/55000 =$	0.1156	0.1156
$\eta_{ss} =$	1.0	1.0

Step 6 Compute elastic stress components per the following equation:

$$\sigma_{is} = \sigma_i(\text{FS})/\alpha_{is}$$

	<u>Outer Shell</u>	<u>Inner Shell</u>
$\sigma_{as} = \sigma_a(\text{FS})/\alpha_{as} = 36849/0.419 =$	87,945 psi	91,654 psi
$\sigma_{hs} = \sigma_h(\text{FS})/\alpha_{hs} = 1560/0.8 =$	1,950 psi	3,628 psi
$\sigma_{ss} = \sigma_{ss}(\text{FS})/\alpha_{ss} = 5226/0.8 =$	6,533 psi	6,533 psi

Step 7 Compute inelastic stress components per the following equation:

$$\sigma_{ip} = \sigma_{is}/\eta_i$$

	<u>Outer Shell</u>	<u>Inner Shell</u>
$\sigma_{ap} = \sigma_{as}/\eta_a = 87,945/0.3725 =$	236,094 psi	275,320 psi
$\sigma_{hp} = \sigma_{hs}/\eta_h = 1950/1.0 =$	1,950 psi	3,628 psi
$\sigma_{sp} = \sigma_{ss}/\eta_{ss} = 6533/1.0 =$	6,533 psi	6,533 psi

Step 8 For the 140-B cask, the buckling evaluation approach consistent with the vessel design and method of analysis is that of paragraph - 1710 of N-284 (By Formulae).

Step 9

- a) discontinuity stresses from normal differential expansion are included and conservatively used without any modification.
- b) no corrections are made for discontinuity stresses
- c) Classical uniaxial buckling values are available from Appendix 2.10.3.1. For the outer cask inner shell at 250°F, these classical values are as follows:

	<u>Outer Shell</u>	<u>Inner Shell</u>
$\sigma_{\theta\theta L}$ =	731,367	639,339
$\sigma_{\theta\theta L}$ = $\sigma_{rr L}$ =	46,055	31,968
$\sigma_{ho L}$ =	45,045	31,422
$\sigma_{\theta\theta L}$ =	179,279	140,349

- d) Applicable elastic and inelastic interaction equations in paragraphs -1713.1.1 and -1713.2.1 of Code Case N-284 are checked as follows:

Elastic buckling (-1713.1.1)

- (a) Axial compression plus hoop compression ($\sigma_{\theta\theta} < 0.5\sigma_{\theta\theta}$)
Not applicable

- (b) Axial compression plus hoop compression ($\sigma_{\theta\theta} \geq 0.5 \sigma_{\theta\theta}$)

$$[(\sigma_{\theta\theta} - 0.5\sigma_{ho L})/(\sigma_{\theta\theta L} - 0.5\sigma_{ho L})] + (\sigma_{\theta\theta}/\sigma_{ho L})^2 \leq 1.0$$

For Outer Shell

$$[(87,945 - 0.5(45,045))/(731,367 - 0.5(45,045))] + (1,950/45,045)^2 = 0.094 < 1.0$$

For Inner Shell

$$[(91,654 - 0.5(31,422))/(639,339 - 0.5(31,422))] + (3,628/31,422)^2 = 0.135 < 1.0$$

- (c) Axial compression plus shear

$$(\sigma_{\theta\theta}/\sigma_{\theta\theta L}) + (\sigma_{\theta\theta}/\sigma_{\theta\theta L})^2 \leq 1.0$$

For Outer Shell

$$(87,945/731,367) + (6,533/179,279)^2 = 0.122 < 1.0$$

For Inner Shell

$$(91,654/639,339) + (6,533/140,349)^2 = 0.145 < 1.0$$

(d) Hoop compression plus shear

$$(\sigma_{\theta\theta}/\sigma_{\theta\theta L}) = (\sigma_{\theta\theta}/\sigma_{\theta\theta L})^2 \leq 1.0$$

For Outer Shell

$$(1,950/46,055) + (6,533/179,279)^2 = 0.044 < 1.0$$

For Inner Shell

$$(3,628/31,968) + (6,533/140,349)^2 = 0.115 < 1.0$$

(e) Axial compression plus hoop compression plus shear

For Outer Shell

$$K = 1 - (\sigma_{\theta\theta}/\sigma_{\theta\theta L})^2 = 1 - (6,533/179,279)^2 = 1.00$$

For Inner Shell

$$K = 1 - (6,533/140,349)^2 = 1.00$$

and equations (a) and (b) above remain unchanged.

Inelastic buckling (-1713.2.1)

For axial compression plus hoop compression,

$$\sigma_{ip}/\sigma_{iel} \leq 1.0$$

For Outer Shell

$$\sigma_{ip}/\sigma_{iel} = 236,094/731,367 = 0.323 < 1.0$$

For Inner Shell

$$= 275,320/639,339 = 0.431 < 1.0$$

For Outer Shell

$$\sigma_{ip}/\sigma_{iel} = 1,950/46,055 = 0.042 < 1.0$$

For Inner Shell

$$= 3,628/31,968 = 0.113 < 1.0$$

(a) Axial compression plus shear

$$(\sigma_{ip}/\sigma_{iel})^2 + (\sigma_{\theta\theta}/\sigma_{\theta\theta L})^2 \leq 1.0$$

For Outer Shell

$$(236,094/731,367)^2 + (6,533/179,279)^2 = 0.106 < 1.0$$

For Inner Shell

$$(275,320/639,339)^2 + (6,533/140,349)^2 = 0.188 < 1.0$$

(b) Hoop compression plus shear

$$(\sigma_{\theta\theta}/\sigma_{\theta\theta L})^2 + (\sigma_{\theta\theta}/\sigma_{\theta\theta L})^2 \leq 1.0$$

For Outer Shell

$$(1,950/46,055)^2 + (6,533/179,279)^2 = 0.003 < 1.0$$

For Inner Shell

$$(3,628/31,968)^2 + (6,533/140,349)^2 = 0.015 < 1.0$$

e) Stiffeners not needed

As all interaction equations are satisfied and there are no concentrated loads on the shell which lead to localized buckling, it is concluded that buckling of the outer cask inner and outer shells will not occur. Of course this conclusion is based upon the assumption that load combinations 7 for the inner shell, and 5 for the outer shell are truly the worst case. Other load cases remain to be analyzed, such as normal condition oblique drops and 200 meter deep immersion. However, it is expected that margins of safety against buckling will be positive for these cases, also.

APPENDIX 2.10.4 - FINITE ELEMENT ANALYSIS

- 2.10.4.1.....PWR Basket Finite Element Analysis - Side Drop
 - 2.10.4.1.1.....Input Data For PWR Basket
 - 2.10.4.1.2.....Output Data For Normal Drop
 - 2.10.4.1.3.....Output Data For 1.4 G Transportation
 - 2.10.4.1.4.....Output Data For 1.4 G Handling/Flat Base
 - 2.10.4.1.5.....Input Data For Accident Drop
- 2.10.4.2.....BWR Basket Finite Element Analysis - Side Drop
 - 2.10.4.2.1.....Input Data For BWR Basket
 - 2.10.4.2.2.....Output Data For Normal Drop
 - 2.10.4.2.3.....Output Data For Accident Drop
- 2.10.4.3.....Cask Tiedown Load Finite Element Analysis
 - 2.10.4.3.1.....Input/Output Data For Cask Tiedown Model
 - 2.10.4.3.2.....Input/Output Data For Cask Vertical
Transportation Loads

NOTE: The above information will be supplied in the final design report and/or the SAR.

2.11 References

1. Cask Systems Development, Initiative 1, Statement Of Work For NuPac 100-Ton Rail/Barge Cask Development, Contract No. DE-AC07-88ID12700, May 10, 1988.
2. 10 CFR Part 71, 1983.
3. U.S. Nuclear Regulatory Commission Regulatory Guide 7.6, Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels, Revision 1, March, 1978.
4. NuPac Report No. L-9775, "O-ring Seal Test Program Report", Prepared by, Nuclear Packaging Inc., Federal Way, WA, July 1989.
5. NuPac Report No. IL-001-NP, "Impact Limiter Polyurethane Foam High Temperature Structural Performance Test Results", Prepared by, Nuclear Packaging Inc., Federal Way, WA, July 19, 1989.
6. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1, 1986 Edition
7. U.S. Nuclear Regulatory Commission Regulatory Guide 7.8, Load Combinations for the Structural Analysis of Shipping Casks, May, 1977.
8. NUREG/CR-1815, "Recommendations for Protecting Against Failure By Brittle Fracture in Ferritic Steel Shipping Containers Up To Four Inches Thick", August, 1981, Lawrence Livermore Laboratory.
9. Bickford, John H., "An Introduction To The Design And Behavior of Bolted Joints", Marcell Dekker, Inc., 1981, Appendix H.
10. ASME Code Case N-284, Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC, Approval Date: August 25, 1980.

11. UCID- 21246 "Dynamic Impact Effects On Spent Fuel Assemblies", by R. Chun, Lawrence Livermore National Laboratory, Oct 20, 1987.
12. WADC Technical Report 57-695, ASTIA Document No. 151165, "Determination of the Mechanical Properties of a High Purity Lead and a 0.058% Copper-Lead Alloy", by Thomas Tietz, Stanford Research Institute, April, 1958.
13. NUREG/CR-0481, SAND77-1872, "An Assessment of Stress-Strain Data Suitable for Finite Element Elastic-Plastic Analysis of Shipping Containers", by H.J. Rack and G.A. Knorovsky, September, 1978.
14. Bulletin S-83N, "Boro-Silicone Shielding", Prepared by, Reactor Experiments, Inc., San Carlos, CA, October 1, 1987. (Include update on thermal expansion and conductivity per telecon October 19, 1988)
15. Roark and Young, "Formulas for Stress and Strain", McGraw Hill Fifth Edition.
16. NuPac Doc. No. CPV89-007-NP, "EYDROP-PC Verification", Sept, 1989
17. NuPac Doc. No. CPV89-008-NP, "EYDROP-PC Users Manual", Sept, 1989
18. NuPac Doc. No. CPV89-009-NP, "SYDROP-PC Verification", Sept, 1989
19. NuPac Doc. No. CPV89-010-NP, "SYDROP-PC Users Manual", Sept, 1989
20. NuPac Doc. No. CPV89-011-NP, "CYDROP-PC Verification", Sept, 1989
21. NuPac Doc. No. CPV89-012-NP, "CYDROP-PC Users Manual", Sept, 1989
22. NuPac Doc. No. CPV89-013-NP, "OBLIQUE-PC Verification", Sept, 1989
23. NuPac Doc. No. CPV89-014-NP, "OBLIQUE-PC Users Manual", Sept, 1989
24. ORNL-NSIC-68, "Cask Designers Guide", by L.B. Shappert, Feb, 1970.

25. I. Sakamoto, "An Experimental Study of Puncture Resistance of Spent Fuel Shipping Casks by Drop Impact Tests", 4th International Symposium on Packaging and Transportation of Radioactive Materials, pp. 262-276, Sept 22-27, 1984, Miami Beach, FL.
26. R.C. Shieh, "Semi-empirical Equations for Puncture Analysis of Lead-Shielded Spent Fuel Shipping Casks".
27. "Safety Analysis Report For The NuPac 125-B Fuel Shipping Cask", Docket Number 71-9200, Prepared by Nuclear Packaging Inc., Federal Way, WA, May, 1985.
28. Shigley, "Mechanical Engineering Design", Third Edition, McGraw-Hill.
29. S. Timoshenko, "Strength of Materials", Part II, Advanced Theory and Problems, Third Edition, 1958.
30. "Design of Welded Structures", The James F. Lincoln Arc Welding Foundation, Part Seven, Section 7.7-5, Transverse Shrinkage, by O. W. Blodgett, 1975.
31. "Handbook of Stainless Steels", D Peckner and I. M. Bernstein Ed., McGraw-Hill, NY, 1977.
32. "Table Speeds Calculation of Strength of Threads", Product Engineering, November 27, 1961, pgs 41 - 49
33. "Mark's Standard Handbook For Mechanical Engineers", Eighth Edition, 1978.
34. "Manual of Steel Construction", 8th Edition, American Institute of Steel Construction.
35. Shigley, "Mechanical Engineering Design", Second Edition, McGraw-Hill, 1972, Sect 7-4 & 7-5, pp. 302-309.

36. "Handbook of Tables for Applied Engineering Science", 2nd Edition, Edited by R. Boltz and G. Tuve, CRC Press.
37. "Safety Analysis Report For The IF-300 Fuel Shipping Cask", Document Number NEDO-10084-3, September, 1984.
38. NUREG/CR-3966, UCID-20639, "Method for Impact Analysis of Shipping Containers", by T.A. Nelson and R.C. Chun, LLNL, Nov, 1987.
39. "Structural Analysis of Shells", by E.H. Baker, et. all, Krieger Pub. Co., 1981.
40. "Engineering Considerations of Stress, Strain, and Strength", by R. C. Juvinall, McGraw-Hill, N.Y., NY, 1967.
41. "Machinery's Handbook", 18th Edition, Industrial Press (pg 1132, Table 6).
42. "Foundation Engineering", by R.B. Peck & W.E. Hanson, Wiley & Sons, N.Y., NY, 1954.
43. "Shock Loading Environments", by F.D. Irani, American Association of Railroads, Pueblo, Colorado, Prepared for EG & G, Idaho, Contract No. DE-AC07-76ID01570, Subcontract No. C85-100785.
44. Internal Boeing Company Document No D-10914, "Finish and Surface Metals Requirements for Aircraft Equipment", Revision September, 1963.

NUPAC 140-B RAIL/BARGE
SPENT FUEL SHIPPING CASK

PRELIMINARY DESIGN THERMAL ANALYSIS REPORT

OCTOBER 1989

DOE CONTRACT NUMBER
DE-AC07-88ID12700

	<u>PAGE</u>
3.0 THERMAL EVALUATION	
3.1 Discussion	3-3
3.2 Summary of Thermal Properties of Materials	3-8
3.3 Technical Specifications of Components	3-12
3.4 Thermal Evaluation for Normal Conditions of Transport	3-13
3.4.1 Thermal Model	3-14
3.4.2 Maximum Temperatures	3-34
3.4.3 Minimum Temperatures	3-40
3.4.4 Maximum Internal Pressures	3-40
3.4.5 Maximum Thermal Stresses	3-42
3.4.6 Evaluation of Package Performance for Normal Conditions of Transport	3-53
3.5 Hypothetical Accident Thermal Evaluation	3-54
3.5.1 Thermal Model	3-54
3.5.2 Package Conditions and Environment	3-57
3.5.3 Package Temperatures	3-57
3.5.4 Maximum Internal Pressures	3-74
3.5.5 Maximum Thermal Stresses	3-77
3.5.6 Evaluation of Package Performance for Hypothetical Accident Thermal Conditions	3-81
3.6 Appendix	3-82
3.7 References	3-156

3.0 THERMAL EVALUATION

3.1 Discussion

This section describes the thermal management of the NuPac 140-B Cask and Fuel Package and details the supporting thermal analyses. The contractual requirements, from Reference (1), that dictate the thermal aspects of the cask design are listed in Appendix 3.6.1.

Thermal management of the NuPac 140-B Cask and Fuel Package employs all passive techniques to keep critical components from exceeding maximum temperature limits. Thermal problems regarding cask component temperature limits do not exist for cold environmental extremes. Passive thermal management techniques are listed below in Table 3.1-1, and illustrated in Figure 3.1-1.

TABLE 3.1-1
NuPac 140-B Cask Passive Thermal Management Features

- A. Copper fins are attached to the cask outer shell and extend through the neutron shield to the thermal shield.
- B. Copper liners are attached to the impact limiters at locations adjacent to the cask and extending radially outward to the impact limiter outer diameter.
- C. Impact limiter exterior surfaces are coated with a white paint that has a low solar absorptance and a high surface emittance.
- D. Interior surfaces of the thermal shield and attached load collar are painted with a high emittance black paint. Adjacent surfaces of the neutron shield are painted with the same material.
- E. The personnel barrier is generously ventilated to allow rising heated air to efficiently escape and be replaced by cooler outside air.
- F. The polyurethane foam impact limiters and thermal shield act as insulators during the hypothetical accident thermal condition.

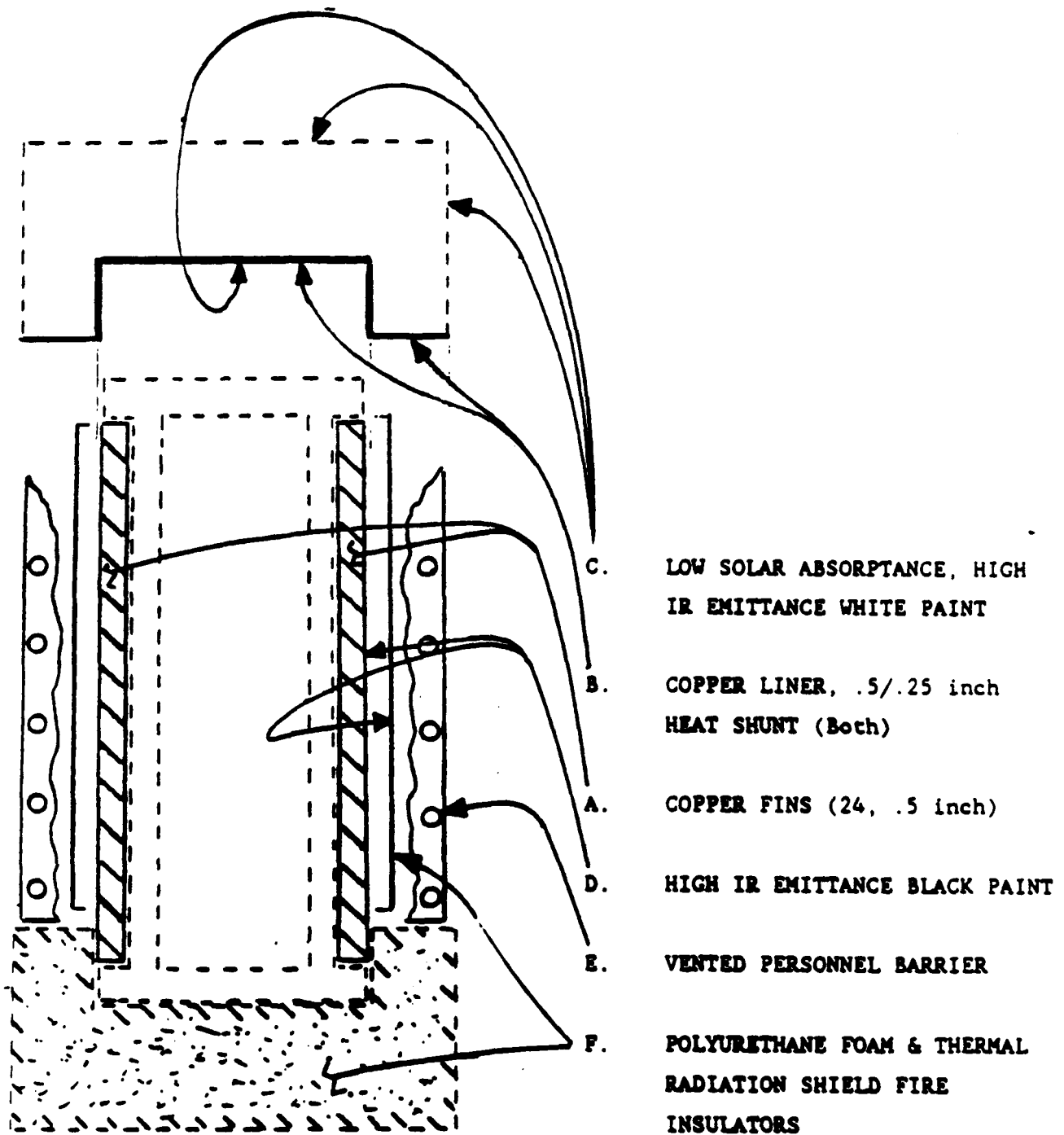


FIGURE 3.1-1

NuPac 140-B Cask Passive Thermal Management Techniques

Critical cask components and their maximum temperatures are listed and discussed in Table 3.1-2 below.

TABLE 3.1-2
Maximum Temperatures

CASK COMPONENT	NORMAL CONDITIONS	ACCIDENT CONDITIONS	SOURCE FOR LIMITS	COMMENTS
0 LID TO CASK SEALS	300°F	300°F	NuPac Tests	(1)
0 NEUTRON SHIELD	400°F	600°F	<u>MFG DATA</u>	(2)
0 IMPACT LIMITER	300°F	N/A	NuPac Tests	(3)
0 CASK EXTERIOR	180°F	N/A	10 CFR 71	(4)
0 GAMMA SHIELD	620°F	620°F	<u>MFG DATA</u>	(5)
0 FUEL CLADDING	716°F	716°F	Customer <u>SPEC</u>	

- (1) Eypel-F elastomeric O-ring seals have been test verified for this application as reported in Reference (2). Compression set was acceptable to 350°F and leak testing was accomplished successfully to 300°F. Additional testing will be performed to verify leak tightness @ 350°F.
- (2) Boro-Silicone, a product of Reactor Experiments Inc. has been extensively tested. The radiation shielding results show greater than 96% Hydrogen remaining below 300°F, and the heat transfer results show negligible material degradation for intermittent operation up to 600°F for about an hour. These test data are reported in Reference (3).
- (3) Nuclear Packaging test results reported in reference (4) show that meaningful polyurethane foam impact resistant structural characteristics exist below 300°F. The material thermally decomposes at about 400°F, but Nuclear Packaging experience and test results, such as those presented in Reference (5), show that such decomposition terminates after the 30 minute, 1,475°F hypothetical thermal test condition. In addition, thermal decomposition occurs only locally at external surfaces on the impact limiters, and penetrates into the foam an insignificant extent. Thus, for the accident condition, a maximum temperature limit is not applicable.

- (4) The cask exterior consists of the exterior surface of the personnel barrier and those regions of the impact limiters outside of its envelope. In the hypothetical accident thermal event, such surface temperature limits are not applicable.
- (5) This limit is the lead melting temperature.

It will be shown in subsequent sections that none of the cask component temperature limits given in Table 3.1-2 above are exceeded. The payload heat source analyzed was 11 KW for both PWR and BWR fuel assemblies. Payloads of 21 PWR assemblies and 52 BWR assemblies were assumed. Per Reference (6), a typical PWR fuel assembly was assumed to contain .4614 metric tons of heavy metal, and a typical BWR, .1833. Data obtained from Reference (7) calculated for 10 year decay and 35000 MWD/MTU shows 1163 Watts/MTU. Significant thermal data from Reference (7) were obtained utilizing the well recognized 'ORIGEN2' code, and are presented below in Table 3.1-3 for information. The data in Table 3.1-3 were generated for 4.5% initial enrichment. The effects of lower enrichment will be studied during the final design phase.

TABLE 3.1-3
ORIGEN2 Results for Decay Versus Burnup and Ageing Period

<u>Burnup</u> <u>(MWD/MTU)</u>	<u>DECAY(*)</u> <u>(5 YR)</u>	<u>DECAY(*)</u> <u>(10 YR)</u>
35000	1787	1163
40000	2092	1360
50000	2776	1807
60000	3580	2347

(*) For one metric ton of heavy metal, thermal power
(watts/MTU)

Five thermal models were formulated to engineer thermal management design aspects of the NuPac 140-B Cask. In brief, they are topically listed below.

1. Half axisymmetric model of cask with impact limiter
2. Thirty degree 2-D model of cask wall
3. Quarter 2-D model of PWR basket
4. Quarter 2-D model of BWR basket
5. Full 2-D model of center PWR fuel assembly

The " ANSYS-PC THERMAL " finite element computer code, version 4.3A4, was utilized for all thermal models mentioned above. Normal and accident boundary conditions were taken from 10 CFR 71. Damage conditions, such as pin puncture of the thermal and neutron shields, side drop indentations of the impact limiters, and general drop damage to the copper fins will be considered during the final design.

3.2 Summary of Thermal Properties of Materials

Of primary concern to thermal analyses are the following five material and surface properties.

k , thermal conductivity, btu/ft-hr-F

ρ , density, lbm/cu ft

c , specific heat, btu/lbm-F

ϵ , surface emittance

α , surface solar absorptance

Values of the cask material properties k , ρ , and c are given in Table 3.2-1 below.

TABLE 3.2-1
140-B Cask Material Properties for Thermal Analyses

Material	k (Btu/ft-hr-F)	ρ (lbm/cu ft)	c (Btu/lbm-F)	Reference
Stainless	8.30 @ 100°F	501 (2)	.120	(8)
Steels (1)	13.0 @ 1100°F			
Lead	20.0 @ 100°F 18.0 @ 600°F	708	.030	(10)
Copper	220	556	.092	(10)
Borated Copper	150	556	.092	(9)
Boro-Silicone	.29	98	.460	(3)
Depleted Uranium	14.0	1180	.028	(9)
Aluminum	128	169	.230	(10)
Polyurethane Foam	.012 @ 100°F .030 @ 300°F	15	.300	(4)
Nitrogen or Air	.015 @ 100°F .030 @ 800°F	N/A (3)	N/A (3)	(10)
Notes:	(1)	Typical values used for austenitic stainless steels as XM-19.		
	(2)	Where temperatures are not given, the material property was assumed constant.		
	(3)	These properties were not used in the analysis.		

Values of surface emittance at various locations internal and external to the cask are listed in Table 3.2-2 below. Only two surfaces of the cask were affected by solar absorptance assumptions, the personnel barrier and the impact limiters. Corresponding, solar absorptance values assumed were .5 and .2 respectively, which were taken from References (9), (10), and (13). The aluminum personnel barrier was assumed to be anodized and the impact limiter surfaces were all assumed to be painted white.

TABLE 3.2-2
140-B Cask Surface Properties for Thermal Analyses

CASK COMPONENT & PAYLOAD SURFACES	EMITTANCE	REFERENCES
Fuel Rods & Fuel Assy Tie plates (Heavily oxidized Zircoloy & Steels)	.80	(11) & (13)
Fuel Assy Spacers (Moderate to heavily oxidized Steels)	.50 to .80 (1)	(10) & (11)
Basket Tubes (Stainless Steels, as received)	.20 to .30 (2)	(11)
Basket OD & ends (Moderately oxidized Steel)	.50	(11)
Cask ID (Light to moderately oxidized Steel)	.30 to .50 (3)	(11)
Cask Lid OD (Moderately oxidized Steel)	.50	(11)
Neutron Shield OD (Black painted copper & boro-silicone)	.80	(9),(10) & (12)
Thermal Shield & load collar ID's (Black painted Steels)	.80	(9) & (12)
All Cask External Surfaces (Decon cleaned Steel)	.25 (4)	(11)
All Impact Limiter External surfaces (White painted Steel & Copper)	.80	(10) & (12)
All Personnel Barrier Surfaces (Anodized Aluminum)	.50	(9) & (10)

- Notes:
- (1) The .50 value was used on the cask lid and end closure side of the spacers.
 - (2) A bounding value of .20 was used for the basket tube wall boundary of the FWR fuel assembly
 - (3) A bounding value of .30 was used for the cask ID boundary of both FWR and BWR basket models.
 - (4) A value of .80 was utilized for all exposed surfaces outside of the impact limiter/cask interface for accident condition analyses.

It should be mentioned that engineering judgement and experience was exercised to select emittance and absorptance values in a realistic and practical manner without compounding conservatism. However, where such data were not readily available in the literature, bounding values were chosen for unknown surface conditions such as .25 for the outer surfaces of the cask after decontamination cleaning. Furthermore, values assumed in notes (2) & (3) above were used to simulate the "first put in service" situation.

3.3 Technical Specifications of Components

The materials used in the NuPac 140-B Cask which are considered to be temperature sensitive are the O-ring seals, the borated silicone neutron shield segments, and the impact limiter polyurethane foam. Critical cask component temperature limits have already been mentioned in Table 3.2-1 which includes the contractual fuel cladding limit, and the 10 CFR 71 external surface limit.

The Nuclear Packaging engineering testing described in Reference (2) shows that the elastomeric O-ring seals are available which are acceptable for compression set to 350°F, and leak testing was accomplished successfully to 300°F. This testing also shows that the seals are acceptable at the other extreme of -40°F. Further leak testing is planned to demonstrate leak tightness at 350°F.

Boro-Silicone, a product of Reactor Experiments Inc. has been extensively tested, and is being evaluated for use in the NuPac 140-B Cask. The radiation shielding results show greater than 96% Hydrogen remaining below 300°F, and the heat transfer results show negligible material degradation for intermittent operation up to 600°F for about an hour. These test data are reported in Reference (3).

Nuclear Packaging test results reported in reference (4) show that meaningful polyurethane foam impact resistant structural characteristics exist below 300°F. The material decomposes at about 400°F, but Nuclear Packaging experience and test results, such as those presented in Reference (5), show that such decomposition terminates after the 30 minute 1,475°F hypothetical thermal test condition. In addition, thermal decomposition occurs only locally at external surfaces on the impact limiters, and penetrates into the foam an insignificant extent. Nuclear Packaging testing and experience show that this FR3700 foam formulation is also acceptable down to -20°F.

Other cask materials are Stainless steel, Copper, Aluminum, and Lead with respective melting points of 2600°F, 1980°F, 1215°F, and 620°F.

For all cask components, the temperatures resulting from 10 CFR 71 normal conditions fall within the above temperature limits. In the 10 CFR 71 accident condition hypothetical thermal 1,475°F test transient, the polyurethane foam will see temperatures locally in excess of 300°F, but the foam will still perform its function as an insulator.

3.4 Thermal Evaluation for Normal Conditions of Transport

This section presents the thermal analyses of the NuPac 140-B Cask for normal conditions of transport. The thermal conditions considered are those specified in Reference (14), namely, 10 CFR 71, Section 71.71. Per paragraph 71.71 (c)(1), a 100 F ambient temperature with the insolation values used for heat input to the exterior cask surfaces shown in Table 3.4-1, shall be assumed.

TABLE 3.4-1
10 CFR 71 Insolation Data

<u>FORM AND LOCATION OF SURFACE</u>	<u>TOTAL INSOLATION FOR A 12 HR PERIOD (Btu/sq ft)</u>
• Flat surfaces transported horizontally	
- Base	None
- Other surfaces	2,950
• Flat surfaces not transported horizontally	738
• Curved surfaces	1,475

The payload decay heat source analyzed was 11 KW for both PWR and BWR fuel assemblies. Payloads of 21 PWR assemblies and 52 BWR assemblies were assumed. Per Reference (6), a typical PWR fuel assembly was assumed to contain .4614 metric tons of heavy metal, and a typical BWR, .1833. Data obtained from Reference (7) calculated for 10 year decay and 35000 MWD/MTU shows 1163 Watts/MTU.

The only analysis performed for normal conditions was the hot day, 100°F, with solar insolation. As found in Reference (5), normal condition analyses with no solar insolation, or for cold day -40°F environments were not significant.

3.4.1 Thermal Model

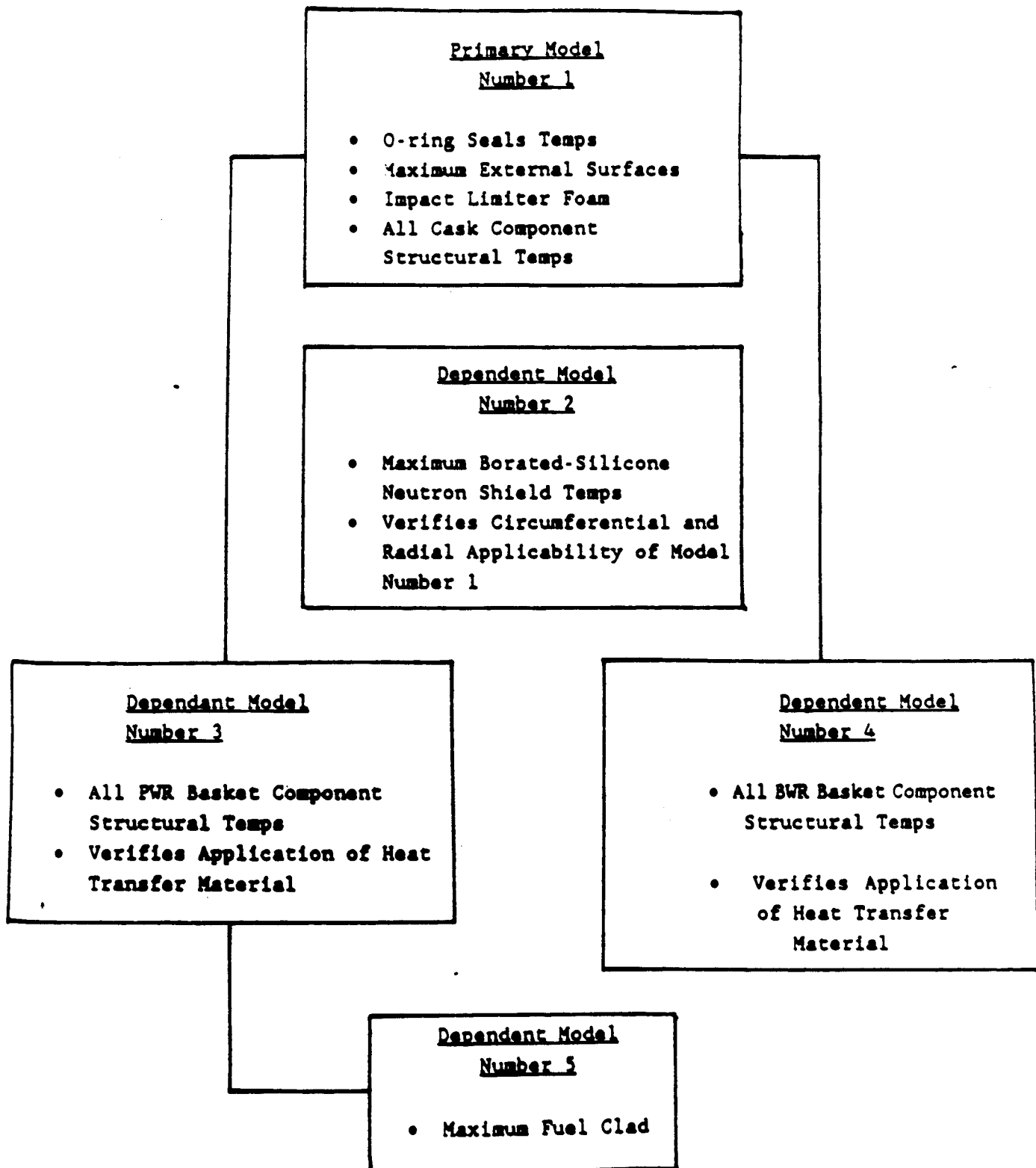
Five thermal models were formulated to engineer thermal management design aspects of the 140-B cask. In brief, they are topically listed below.

1. Half axisymmetric model of cask with impact limiter
2. Thirty degree 2-D model of cask wall
3. Quarter 2-D model of PWR basket
4. Quarter 2-D model of BWR basket
5. Full 2-D model of center PWR fuel assembly

The "ANSYS-PC THERMAL" finite element computer code, version 4.3A4, was utilized for all thermal models mentioned above, 10 CFR 71 normal boundary conditions were assumed. Typical finite element formulations utilized in the above five thermal models are described in Appendix 3.6.9.

The cask thermal analysis was implemented with five interdependent thermal models, as mentioned above, to reduce the complexity that would have existed had all five been combined into one. The interdependency of the five thermal models is depicted in Table 3.4.1-1 along with the specific critical cask component temperatures that were determined by each specific model.

TABLE 3.4.1-1
Interdependency of 140-B Cask Thermal Models



3.4.1.1 Half Axisymmetric model of cask with impact limiter

The finite element model is depicted in an exploded view in Figure 3.4.1-1.

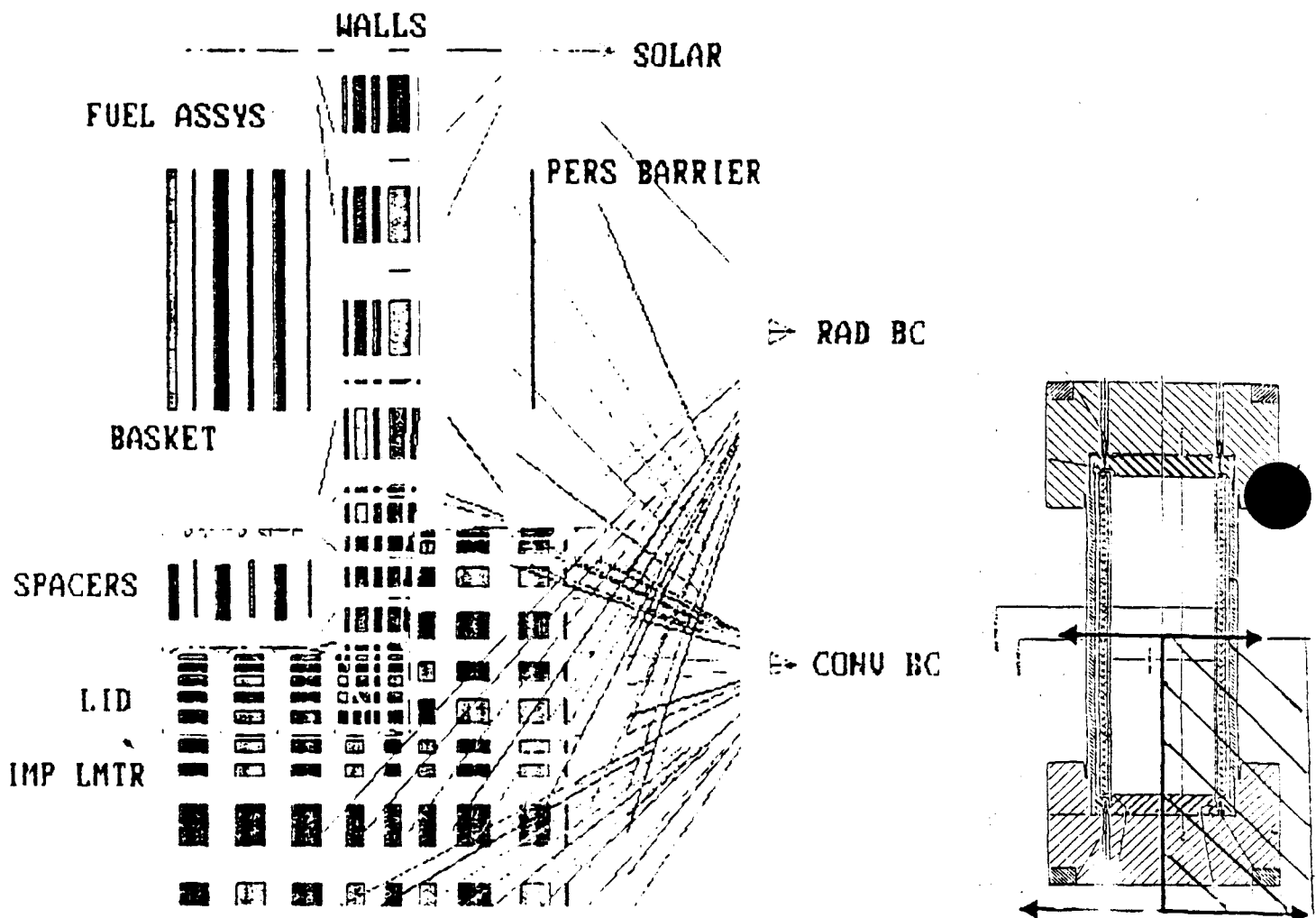


FIGURE 3.4.1-1
Exploded View of Half Axisymmetric Model of Cask
with Impact Limiter

The thermal model shown in Figure 3.4.1-1 consists of 410 elements with 282 corresponding nodes. As depicted in Table 3.4.1-1, this is the primary cask thermal model since all other models utilize the results of this model as boundary conditions. Significant cask component temperatures utilized from this model are also listed in Table 3.4.1-1. The shaded portion to the right of the element breakdown represents the physical section of the cask simulated by the finite elements. Per the ANSYS-PC THERMAL code notation, elements utilized are listed below.

- o 2-D Axisymmetric Isoparametric Thermal Solid, "STIF55"
- o 2-D Conducting Bar, "STIF32"
- o Radiation Link, "STIF31"
- o Convection Link, "STIF34"

The thermal solid elements can be recognized by the square, rectangular, and triangular shapes. All such shapes are really just the cross sections of corresponding solid rings. Radiation and convection elements can be recognized by jagged lines, and conduction elements by solid lines and dots. Typical components of the cask are listed in the figure along with boundary conditions (open triangles) and solar heat inputs (arrows). The decay heat is input to homogenized fuel assembly thermal solid elements via heat generation commands which are not shown graphically by the ANSYS code. Input, nodal definition, and output nodal temperatures are given in Appendix 3.6.2.

The following assumptions are inherent in the half axisymmetric model.

- A. Material thermal conductivities are varied linearly with temperature or held constant per Table 3.2-1.
- B. Surface emittance and absorptance values are held constant per Table 3.2-2. All thermal radiation from or between cask surfaces is assumed to be diffuse and grey per Reference (12).
- C. The cask is assumed to be horizontal and convection inside is assumed to be negligible per Reference (13).

- D. All thermal radiation between adjacent parallel cask surfaces is assumed to be between infinite parallel surfaces according to the equation below:

$$F_{12} = 1 / [(1/\epsilon_1) + (1/\epsilon_2) - 1] \quad (1)$$

- E. All thermal radiation from cask external surfaces is assumed to be between a small body within a large body according to the equation below:

$$F_{12} = \epsilon_1 F_{12} A_1 \quad (2)$$

Black body view factors are set equal to 1.0 at the thermal shield, except for a value of .5 applied locally between the cask lift collar and the impact limiter.

- F. Heat transfer within the basket is assumed to flow via radiation and solid conduction. Heat transfer from the outer basket walls to the cask ID and ends is via radiation and gas (nitrogen) conduction. A radial spacing of .25 inches and end spacings of 1.0 inch are assumed. Contact conductance is conservatively not covered between the basket and cask because of the nebulous assumptions that would have to be employed for contact surface areas and pressures.
- G. Heat transfer from the outer neutron shield surface to the ID of the thermal shield is via radiation and gas (air) conduction. A radial spacing of .13 inches is assumed. Contact conductance for the steel wire in between and spaced every three inches is assumed negligible due to thermal expansion effects.

- H. Heat transfer from the outer cask surface to the ID of impact limiter is via radiation and gas (air) conduction. A radial spacing of .25 inches is assumed. High contact conductance at the impact limiter bolt circle attachment interface is assumed due to high attachment bolt torque and a coefficient of 1000 Btu/hr-sqft-F (curve #10) was used. Contact conductance coefficients are obtained from Reference (o) from the figure reproduced below as Figure 3.4.1-2. Curves #9 & #10 for stainless steel are utilized.

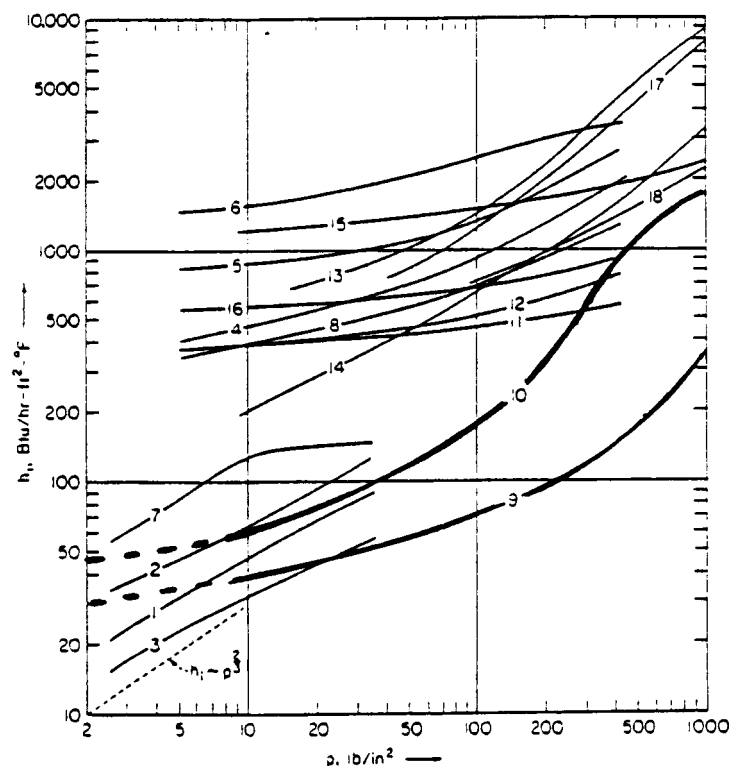


FIGURE 3.4.1-2
Contact Conductance Coefficients as a Function of
Surface Contact Pressure

- I. Heat transfer from the outer cask forging surface to the OD of the cask lid along the bore seal region is via radiation and gas (air) conduction. A radial spacing of .012 inches is assumed. High contact conductance at the cask lid bolt circle attachment interface is assumed due to high attachment bolt torque and a coefficient of 1000 Btu/hr-sqft-F was used. Contact conductance coefficients are obtained from Figure 3.3, curve #10, taken from Reference (15).
- J. Cask and impact limiter external convection is based on standard free convection relations for horizontal cylinders as given below and taken from Reference (12).

$$Nu_D = 0.53(Gr_D Pr)^{1/4}$$

(3)

At 150 F mean film temperature, the above equation yields free convection coefficients of .45, .45 and .49 Btu/hr-sqft-F on impact limiter, personnel barrier, and cask surfaces respectively.

- K. The neutron shield borated silicone sections and the copper fins were modeled in parallel. Thus, the external neutron shield temperature will be an average value. The thirty degree cask wall 2-D model to be described subsequently in Section 3.4.1.2 shall be utilized to optimize this interface.
- L. The payload and basket structure, including fuel assembly spacers, were all modeled in a homogenized fashion, and contact conductance was not accounted for at basket panel junctions. Thus basket and payload temperatures are only average values and are not used for design. The basket and fuel assembly 2-D models described subsequently in Sections 3.4.1.3, 3.4.1.4, and 3.4.1.5 are utilized for design.

3.4.1.2 Thirty Degree 2-D Model of Cask Wall

The finite element model is depicted in an exploded view in Figure 3.4.1-3.

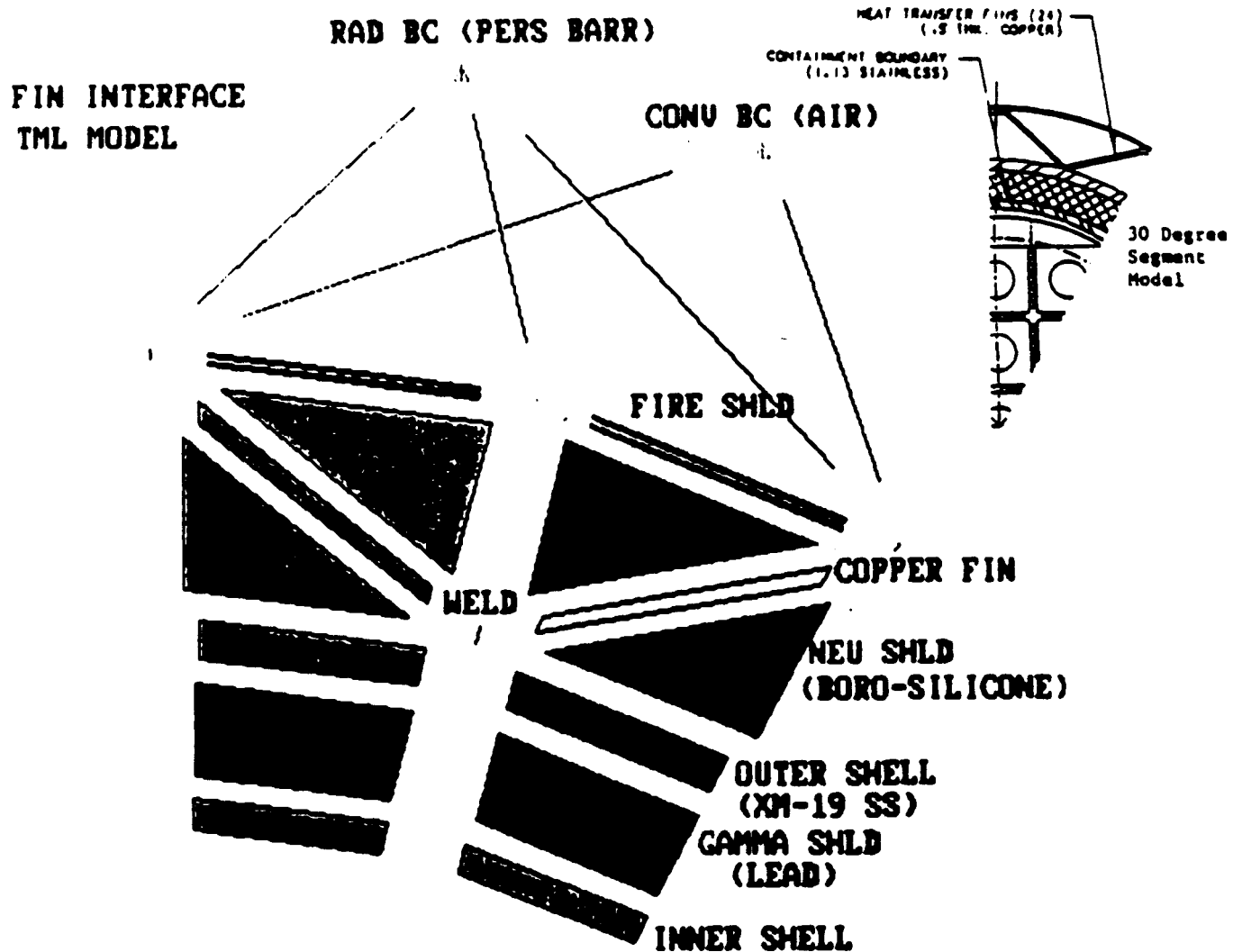


FIGURE 3.4.1-3
Exploded View of Thirty Degree 2-D Model of Cask Wall

The thermal model shown in Figure 3.4.1-3 consists of 31 elements with 31 corresponding nodes. As depicted in Table 3.4.1-1, this model depends upon the primary model of Section 3.4.1-1 because the personnel barrier temperature is a boundary condition. Significant cask component temperature utilized from this model are also listed in Table 3.4.1-1. The segment portion to the right of the element breakdown represents the physical section of the cask simulated by the finite elements. The physical section of the cask simulated is taken from a cross sectional cut perpendicular to the cask center-line in the center of the cask midway between the impact limiters.

Per the ANSYS-PC THERMAL code notation, elements utilized are listed below.

- o 2-D Isoparametric Thermal Solid, "STIF55"
- o 2-D Conducting Bar, "STIF32"
- o Radiation Link, "STIF31"
- o Convection Link, "STIF34"

The thermal solid elements can be recognized by the square, rectangular, and triangular shapes. Radiation, convection, and conduction elements can be recognized by jagged lines. Conduction elements also show as dots and solid lines. Typical components of the cask are listed in the figure along with boundary conditions (open triangles). Solar heat input is not present because the personnel barrier blocks the sun. The corresponding personnel barrier temperature of 177 °F, found from the analysis of Section 3.4.1.1, is input as the radiation boundary condition.

The decay heat is input to inner containment shell nodes in a uniform manner via heat flow commands which are shown as arrows. The actual input information utilized for the thermal model in Figure 3.4.1-3 is given in Appendix 3.6.3.

The following assumptions are inherent in the thirty degree 2-D model.

- A. Material thermal conductivities, same as Section 3.4.1.1.

- B. Surface emittance and absorptance values, same as Section 3.4.1.1.
- C. The cask is assumed to be horizontal and convection inside is assumed to be negligible, same as Section 3.4.1.1.
- D. All thermal radiation between adjacent parallel cask surfaces, same as Section 3.4.1.1.
- E. All thermal radiation from cask external surfaces, same as Section 3.4.1.1. All black body view factors are assumed = 1.0.
- F. Heat transfer from the payload/basket is applied uniformly to the three cask inner wall nodes.
- G. Heat transfer from the outer neutron shield surface to the ID of the thermal shield is the same as Section 3.4.1.1.
- H. Assumption H from Section 3.4.1.1 is not applicable because the impact limiter is not included in the model.
- I. Assumption I from Section 3.4.1.1 is not applicable because the cask lid is not included in the model.
- J. Cask thermal shield external convection is the same as Section 3.4.1.1.

3.4.1.3 Quarter 2-D Model of PWR Basket

The finite element model is depicted in Figure 3.4.1-4

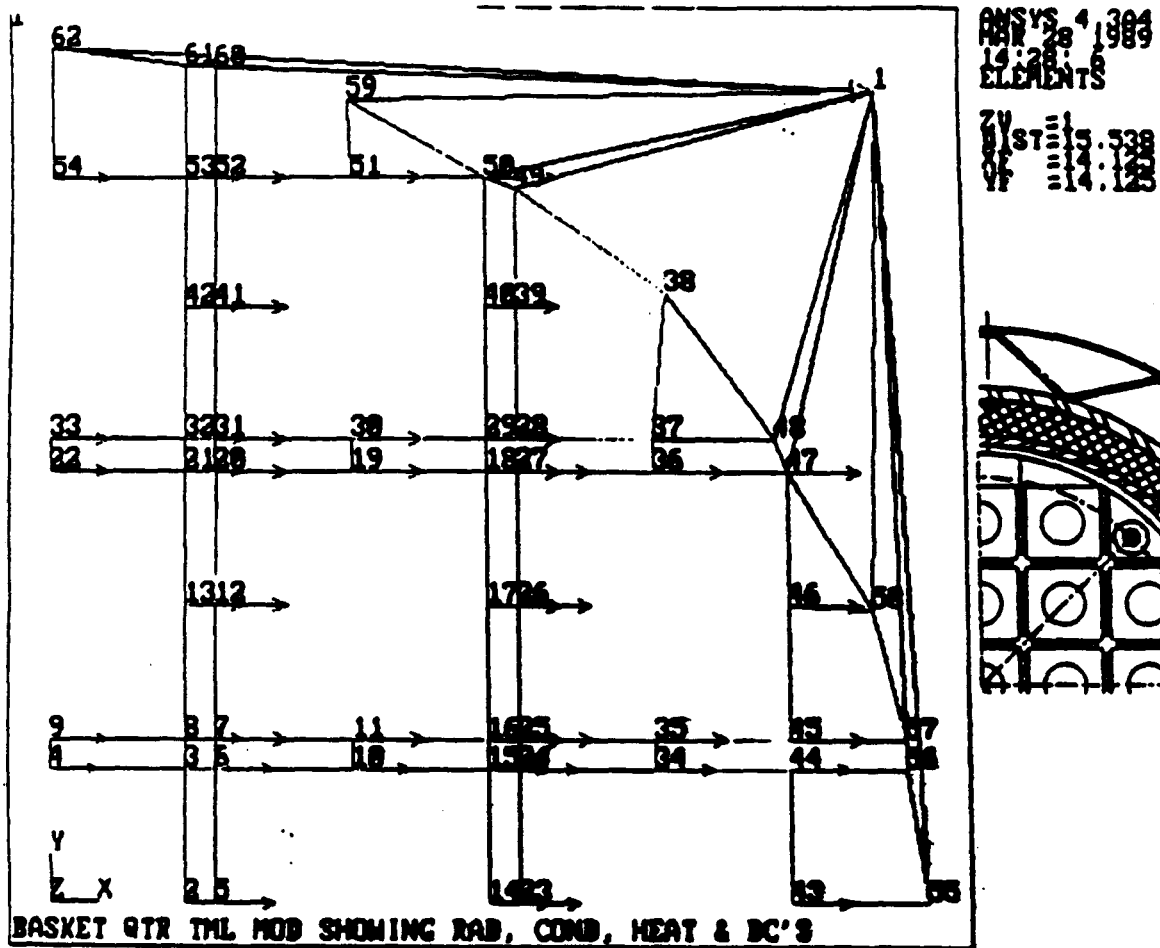


FIGURE 3.4.1-4
Quarter 2-D Model of PWR Basket

The thermal model shown in Figure 3.4.1-4 consists of 140 elements with 82 corresponding nodes. As depicted in Table 3.4.1-1, this model depends upon the primary model of Section 3.4.1-1 because the inner cask containment shell temperature is a boundary condition. Significant component temperature utilized from this model are also listed in Table 3.4.1-1. The segment portion to the right of the element breakdown represents the physical section of the basket simulated by the finite elements. The physical section of the basket simulated is taken from a cross sectional cut perpendicular to the cask center-line in the center of the cask midway between the impact limiters.

Per the ANSYS-PC THERMAL code notation, elements utilized are listed below.

- o 2-D Conducting Bar, "STIF32"
- o Radiation Link, "STIF31"
- o Convection Link, "STIF34"

Radiation and gas conduction can be recognized by jagged lines. Basket structure conduction elements show as solid lines. The basket panels are obviously recognized as the straight line conduction link elements and the boundary condition of the adjacent cask wall by the open triangles. Cask external heat transfer effects are not directly present because the adjacent cask wall temperature, which is found from the analysis of Section 3.4.1.1, is input as a constant temperature boundary condition of 291 °F.

The decay heat is input at inner basket panel nodes, also shown in Figure 3.4.1-4, in a uniform manner via heat flow commands which are shown by arrows. The actual input information utilized for the thermal model in Figure 3.4.1-4 is given in Appendix 3.6.4.

The following assumptions are inherent in the quarter 2-D PWR basket model.

- A. Material thermal conductivities, same as Section 3.4.1.1.

- B. Surface emittance and absorptance values, same as Section 3.4.1.1.
- C. The cask is assumed to be horizontal and convection inside is assumed to be negligible, same as Section 3.4.1.1.
- D. All thermal radiation between adjacent parallel cask surfaces, same as Section 3.4.1.1.
- E. All thermal radiation from cask external surfaces, same as Section 3.4.1.1. All black body view factors are assumed = 1.0.
- F. Decay heat transfer from the fuel assemblies is applied uniformly to all corresponding inner panel wall nodes.
- G. Assumption G from Section 3.4.1.1 is not applicable for this model because the inner cask wall temperature is input as the boundary condition.
- H. Assumption H from Section 3.4.1.1 is not applicable because the impact limiter is not included in the model. However, contact conductance was considered at the panel junction keys. A low value of 30 Btu/hr-sqft-F was assumed, and as shown in Figure 3.4.1-2, curve #9 (extrapolated), this corresponds to very low joint contact surface pressure (about 2 psi which is determined from Figure 3.4.1-2). A "slight press fit" will be specified for manufacture.
- I. Assumption I from Section 3.4.1.1 is not applicable because the cask lid is not included in the model. Contact conductance was mentioned in item H above.
- J. Assumption J from Section 3.4.1.1 is not applicable for this model because the inner cask wall temperature is input as the boundary condition.

3.4.1.4 Quarter 2-D Model of BWR Basket

The finite element model is depicted in Figure 3.4.1-5.

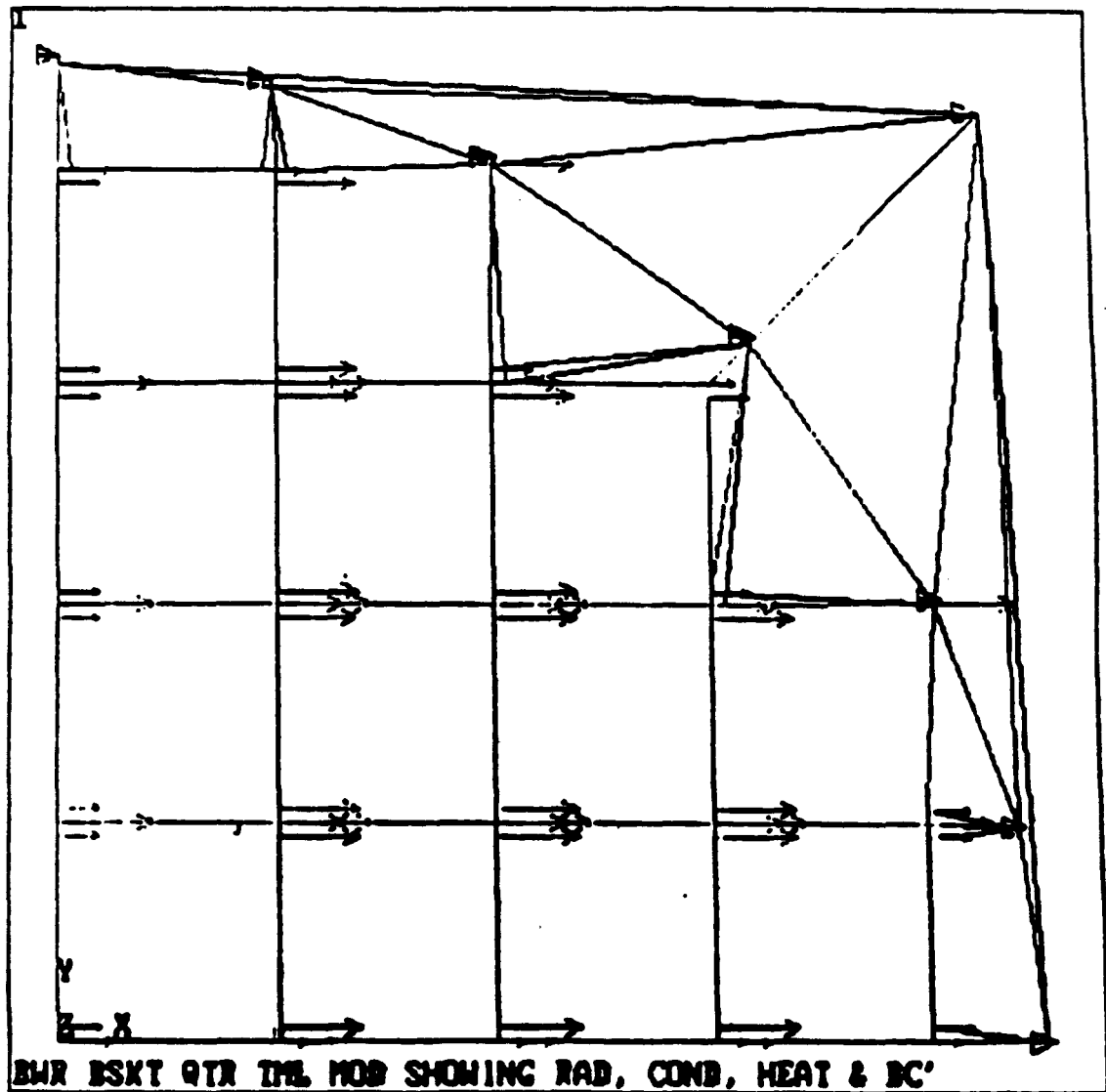


FIGURE 3.4.1-5
Quarter 2-D Model of BWR Basket

The thermal model shown in Figure 3.4.1-5 consists of 151 elements with 102 corresponding nodes. As depicted in Table 3.4.1-1, this model depends upon the primary model of Section 3.4.1-1 because the inner cask containment shell temperature is a boundary condition. Significant component temperatures utilized from this model are also listed in Table 3.4.1-1. The physical section of the basket simulated is similar to the PWR basket of Figure 3.4.1-4, and is taken from a cross sectional cut perpendicular to the cask centerline in the center of the cask midway between the impact limiters.

Per the ANSYS-PC THERMAL code notation, elements utilized are listed below.

- o 2-D Conducting Bar, "STIF32"
- o Radiation Link, "STIF31"
- o Convection Link, "STIF34"

Radiation and gas conduction can be recognized by jagged lines. Basket structure conduction elements show as solid lines. The basket panels are obviously recognized as the straight line conduction link elements and the boundary condition of the adjacent cask wall by the open triangles. Cask external heat transfer effects are not directly present because the adjacent cask wall temperature, which is found from the analysis of Section 3.4.1.1, is input as a constant temperature boundary condition of 291 °F.

The decay heat is input at inner basket panel nodes, also shown in Figure 3.4.1-5, in a uniform manner via heat flow commands which are shown by arrows. The actual input information utilized for the thermal model in Figure 3.4.1-5 is given in Appendix 3.6.5.

The following assumptions are inherent in the quarter 2-D BWR basket model.

- A. Material thermal conductivities, same as Section 3.4.1.1.

- B. Surface emittance and absorptance values, same as Section 3.4.1.1.
- C. The cask is assumed to be horizontal and convection inside is assumed to be negligible, same as Section 3.4.1.1.
- D. All thermal radiation between adjacent parallel cask surfaces, same as Section 3.4.1.1.
- E. All thermal radiation from cask external surfaces, same as Section 3.4.1.1. All black body view factors are assumed = 1.0.
- F. Decay heat transfer from the fuel assemblies is applied uniformly to all corresponding inner panel wall nodes.
- G. Assumption G from Section 3.4.1.1 is not applicable for this model because the inner cask wall temperature is input as the boundary condition.
- H. Assumption H from Section 3.4.1.1 is not applicable because the impact limiter is not included in the model. However, contact conductance was considered at the panel junction keys. A moderate value of 300 Btu/hr-sqft-F was assumed, and as shown in Figure 3.4.1-2, curve #10, this corresponds to moderate joint contact surface pressure (about 200 psi which is determined from Figure 3.4.1-2). An interference fit of .0015 inch minimum will be specified for manufacture.
- I. Assumption I from Section 3.4.1.1 is not applicable because the cask lid is not included in the model. Contact conductance was mentioned in item H above.
- J. Assumption J from Section 3.4.1.1 is not applicable for this model because the inner cask wall temperature is input as the boundary condition.

3.4.1.5 Full 2-D Model of Center PWR Fuel Assembly

The finite element model is depicted in Figure 3.4.1-6.

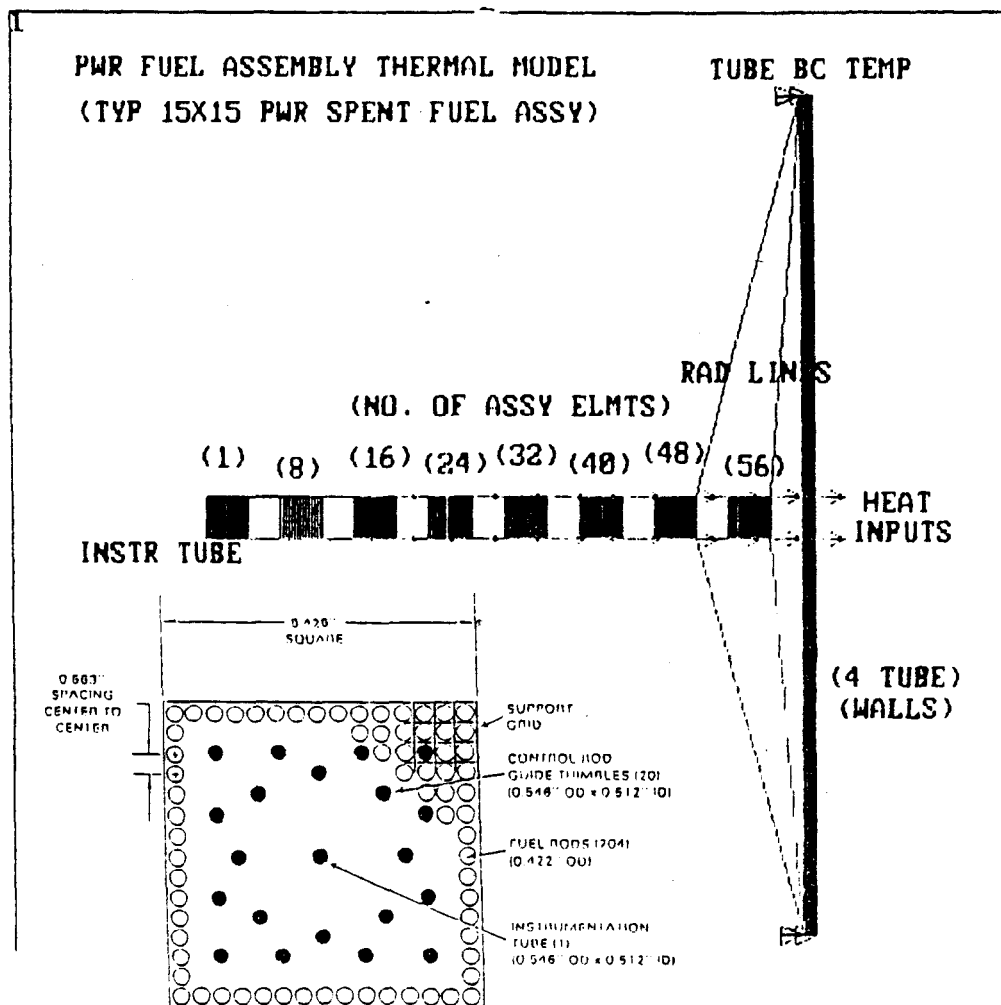


FIGURE 3.4.1-6
Full 2-D Model of Center PWR Fuel Assembly

The thermal model shown in Figure 3.4.1-6 consists of 39 elements with 36 corresponding nodes. The segment portion below the element break-down represents the physical section of the fuel assembly simulated by the finite elements. The physical section simulated is taken from a cross sectional cut perpendicular to the cask centerline in the center of the cask midway between the impact limiters. Furthermore, the physical section simulated is bounded by the center tube of the PWR basket.

Per the ANSYS-PC THERMAL code notation, elements utilized are listed below.

- o 2-D Isoparametric Thermal Solid, "STIF55"
- o Radiation Link, STIF31"

The thermal solid elements can be recognized by the square and rectangular shapes. Radiation elements can be recognized by jagged lines. There are no conduction or convection elements. The number of fuel assembly rods and tube walls represented by each element are listed in the figure along with boundary conditions (open triangles). The far left element is an instrumentation tube and lies at the center of the fuel assembly. The next element represents the eight fuel rods surrounding the instrumentation tube, etc. Finally, the far right element represents the four walls of the center basket tube surrounding the fuel assembly. The temperature of this far right element is found from the analysis of Section 3.4.1.3, and is input as the radiation boundary condition.

The decay heat is input to each square fuel element node in a uniform manner via heat flow commands which are shown as arrows. The actual input information utilized for the thermal model in Figure 3.4.1-6 is given in Appendix 3.6.6.

The following assumptions are inherent in the PWR fuel assembly model.

- A. Material thermal conductivities, same as Section 3.4.1.1. The only material of interest is depleted uranium and its conductivity was assumed to be quite larger than actual in order that the exterior of the fuel rods would have nearly a constant temperature.

- B. Surface emittance and absorptance values, same as Section 3.4.1.1.
- C. The cask is assumed to be horizontal and convection is assumed to be negligible, same as Section 3.4.1.1.
- D. Thermal radiation between fuel assembly rod surfaces, and between the fuel rods and center basket tube walls is assumed to be between two gray surfaces, A_1 & A_2 according to the equation below:

$$F_{12} = 1 / \{ [(1-\epsilon_1)/\epsilon_1] + (1/F_{12}) + (A_1/A_2) [(1-\epsilon_2)/\epsilon_2] \} \quad (4)$$

- E. Black body view factors used in the above equation were taken from Reference (12) and reproduced in Figure 3.4.1-7 and correspond to values of .83 between first and second rows of rods, .16 between first and third rows of rods, and .01 between first and fourth rows of rods. A view factor of 1.0 is assumed between the outside row of rods and the tube walls.
- F. Decay heat is input to each row of fuel rods uniformly at each corresponding element node. Since only one element shows for each row of tubes, the corresponding heat rates applied increase according to the number of rods in the row.
- G. Assumption G from Section 3.4.1.1 is not applicable for this model because the central basket tube wall temperature is input as the boundary condition.
- H. Assumption H from Section 3.4.1.1 is not applicable because the impact limiter is not included in the model.
- I. Assumption I from Section 3.4.1.1 is not applicable because the cask lid is not included in the model.
- J. Assumption J from Section 3.4.1.1 is not applicable for this model because the central basket tube wall temperature is input as the boundary condition.

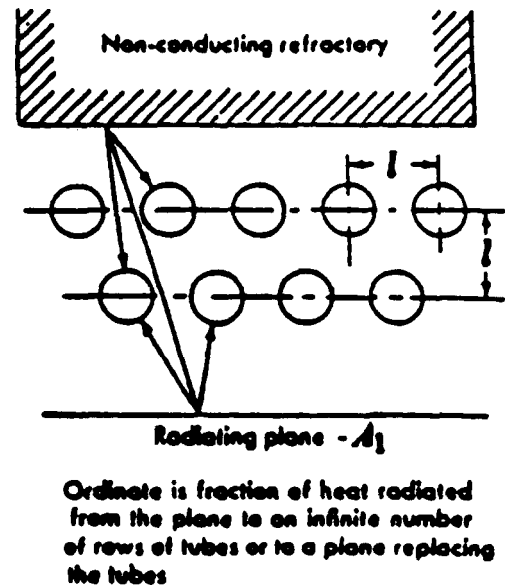
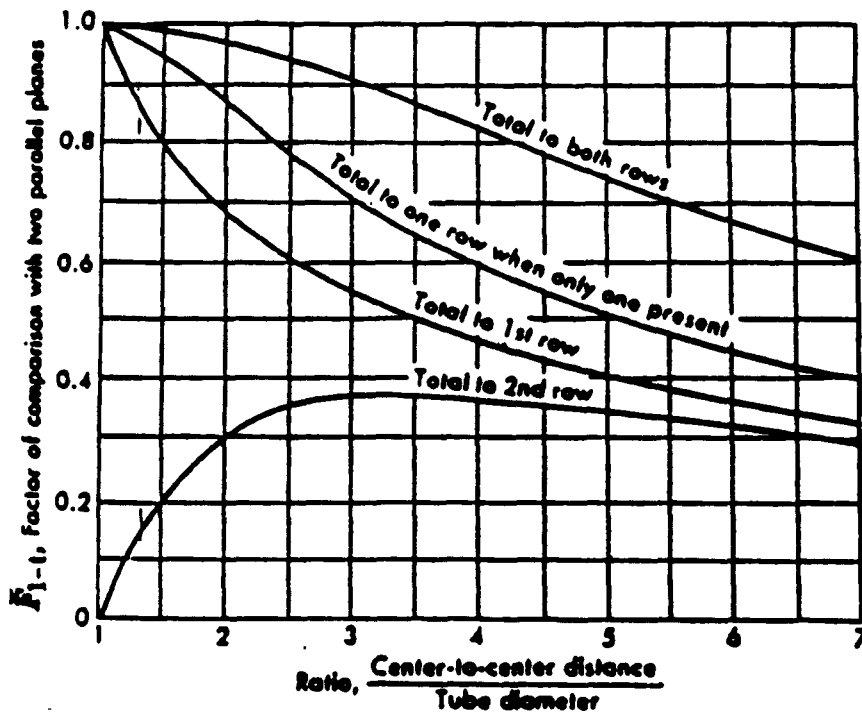


FIGURE 3.4.1-7
Shape Factor for a Plane and One or Two Rows of Tubes
Above and Parallel to it

3.4.2 Maximum Temperatures

The maximum temperatures noted for normal conditions of transportation (i.e. 100°F ambient, insolation, and 11 KW payload decay heat) are presented in Table 3.4.2-1 below for the critical cask components listed in Table 3.1-2.

TABLE 3.4.2-1
Normal Condition Thermal Analysis Results for Critical
Cask Components

	O-RING SEAL TEMP	LIMITER FOAM TEMP	NEUTRON SHIELD TEMP	FUEL CLAD TEMP	PERSONNEL BARRIER TEMP
CASK DESIGN BASIC CONFIGURATION					
(TEMPERATURE LIMITS FOR CASK APPLICATION)	350 °F	300 °F	400 °F	716 °F	180 °F
*****10 CFR 71 NORMAL CONDITIONS*****					
CONFIGURATION EMPLOYS	300 °F	278 °F	294 °F	660 °F	177 °F
PASSIVE THERMAL MANAG- (NODE)	(NODE)	(NODE)	(NODE)	(SEE)	(NODE)
EMENT PER TABLE 3.1-1 (138)	(138)	(200)	(79)	(NOTE)	(5)

NOTE: This value was obtained from the thermal model of Section 3.4.1.5.
All other values in this table were obtained from the thermal model
of Section 3.4.1.1.

Other cask component temperatures of significance are listed with their ranges in Table 3.4.2-2. A corresponding cask thermal plot is given in Figure 3.4.2-1.

TABLE 3.4.2-2
Normal Condition Thermal Analysis Results for Other
Cask Components

THERMAL MODEL			TEMPERATURE RANGE (DEG, F)		
o	Inner steel shell	Section 3.4.1.1	298	to	288
o	Lead gamma shield	"	296	to	283
o	Outer steel shell	"	294	to	280
o	Copper fins	"	293	to	271
o	Boro-silicone neutron shield	"	293	to	271
o	Thermal shield	"	246	to	241
o	Load collar	"	290	to	245
o	End closure forgings	"	296	to	273
o	Cask lid	"	333	to	294
o	O-ring seals	"	300	to	295
o	Impact limiter	"	278	to	110
o	Fuel assy spacers	"	387	to	312
o	PWR basket	Section 3.4.1.3	580	to	356
o	BWR basket	Section 3.4.1.4	594	to	403
o	Fuel rods	Section 3.4.1.5	660	to	440

Figure 3.4.2-2 shows equilibrium temperatures at various locations within the thirty degree cask wall segment model discussed in Section 3.4.1.2. These temperature results are presented to check on the limitations of assumption 'K' of Section 3.4.1.1. As shown, the axisymmetric model predicts inside cask wall and exterior thermal shield temperatures of 291° F and 241° F. The thirty degree cask wall model shows good agreement with corresponding temperatures ranging 287 to 292° F and 232 to 240° F respectively. Furthermore, the thirty degree cask wall model gives local borated-silicone temperature variations, as 260, 270 and 277° F that can be used for thermal expansion design tolerance calculations.

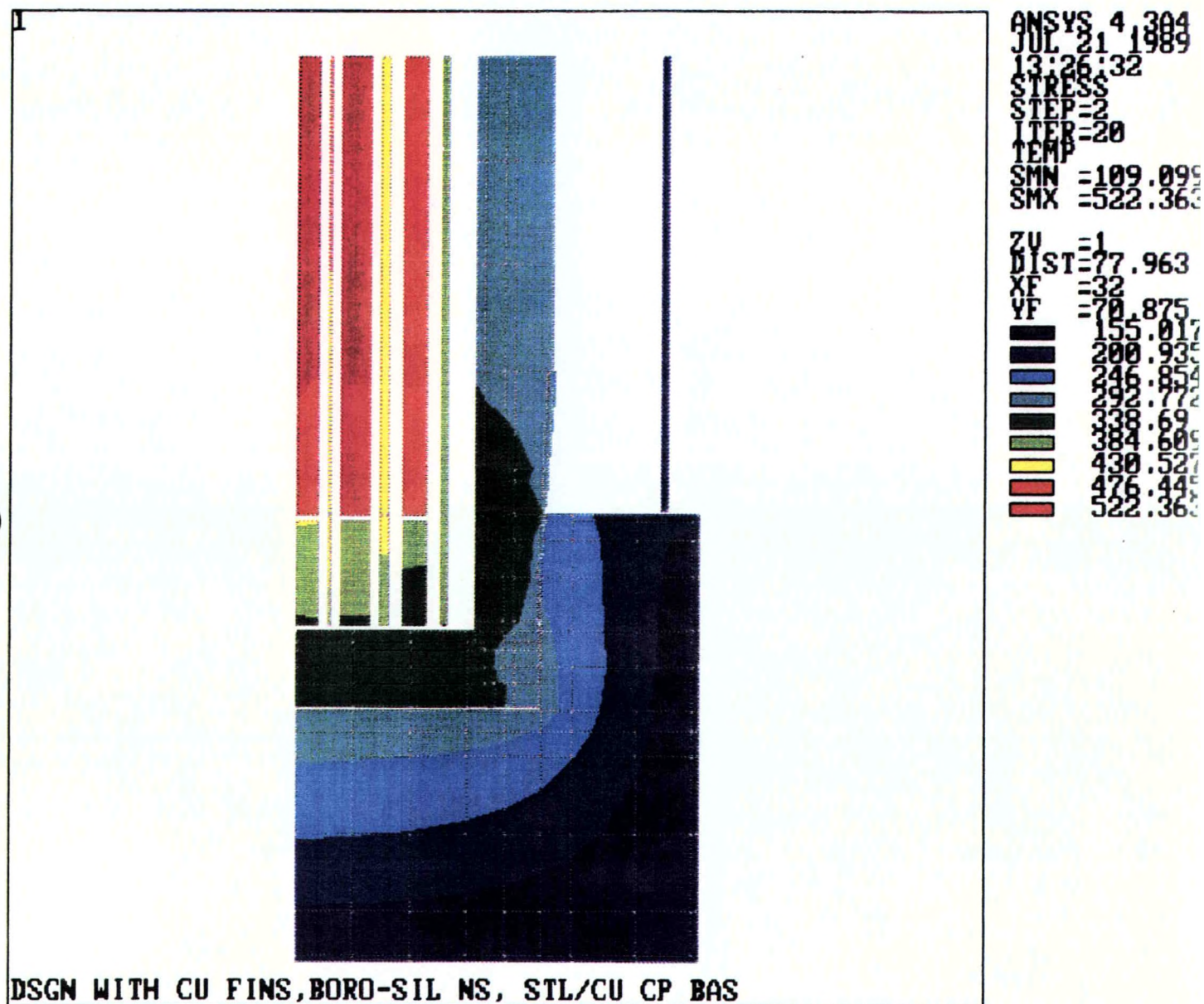


FIGURE 3.4.2-1

140-B Cask Temperature Distribution Corresponding to
 10 CFR 71 Normal Conditions of Transport

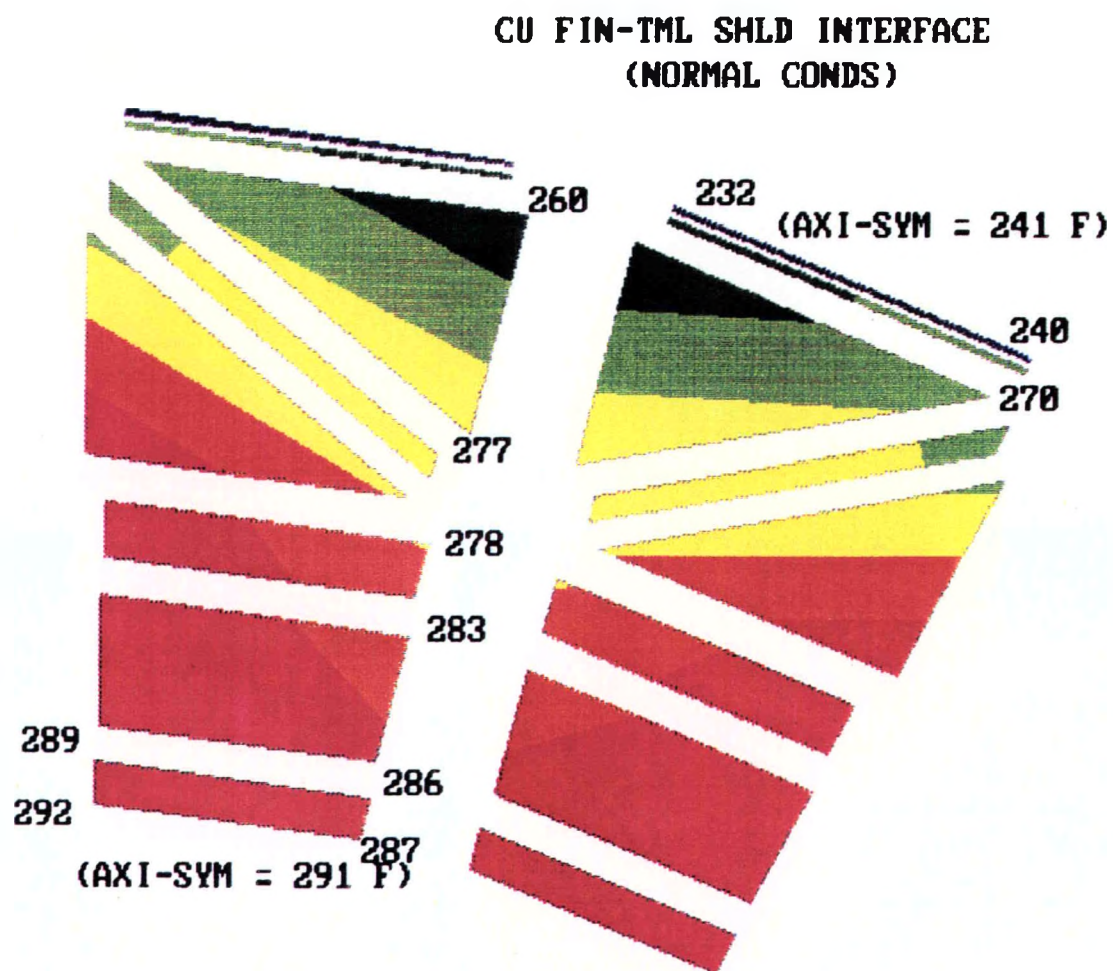


FIGURE 3.4.2-2
Normal Condition Temperature Distribution in Cask
Wall Between Impact Limiters

Figure 3.4.2-3 shows equilibrium temperatures at various locations within the PWR basket thermal model discussed in Section 3.4.1.3. These temperatures are presented for PWR basket design. Note that only half of the model temperatures are presented because the other half are symmetrically identical.

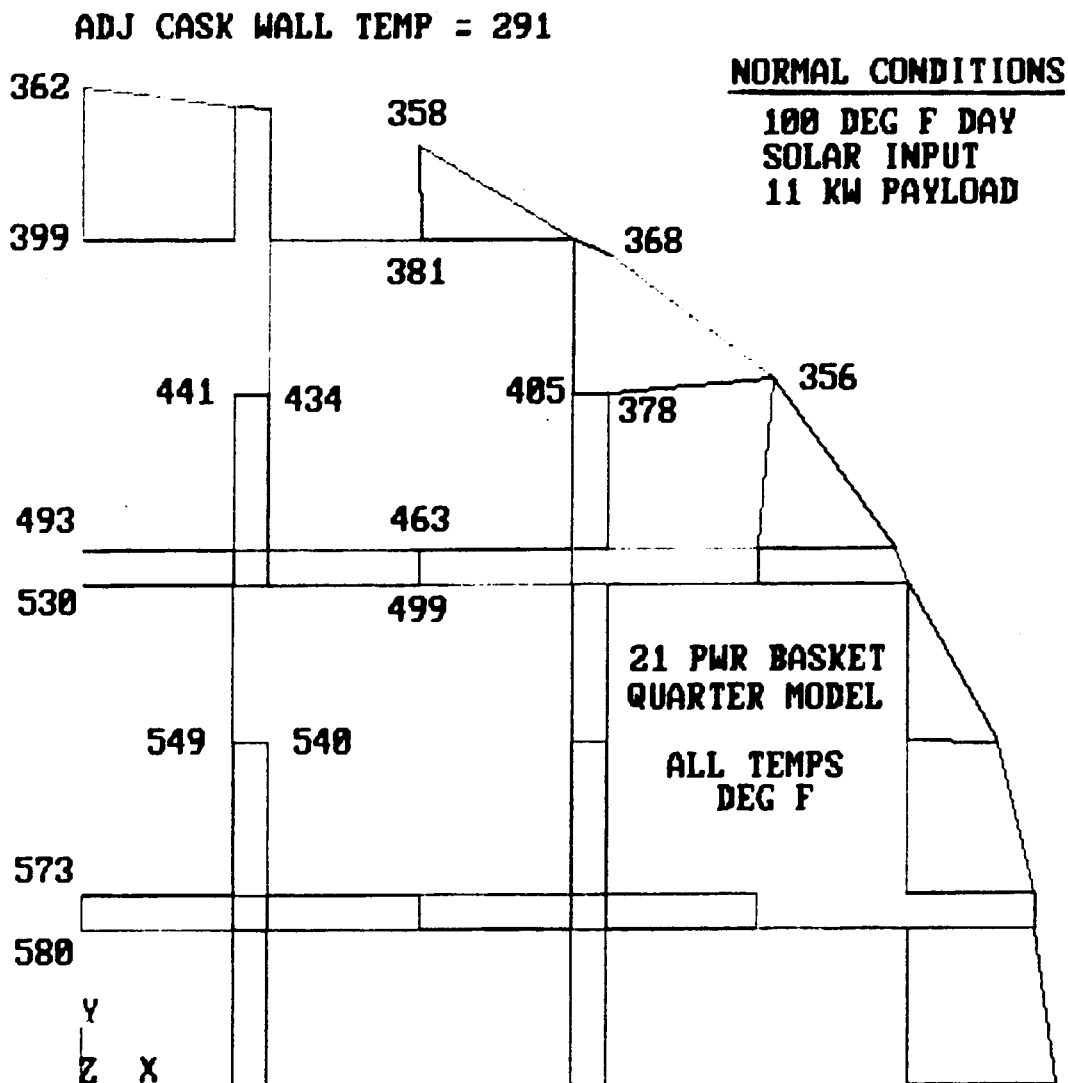


FIGURE 3.4.2-3

Normal Condition Temperature Distribution in PWR
 Basket at a Location Between Impact Limiters

Figure 3.4.2-4 shows equilibrium temperatures at various locations within the BWR basket thermal model discussed in Section 3.4.1.4. These temperatures are presented for BWR basket design. Note that only half of the model temperatures are presented because the other half are symmetrically identical.

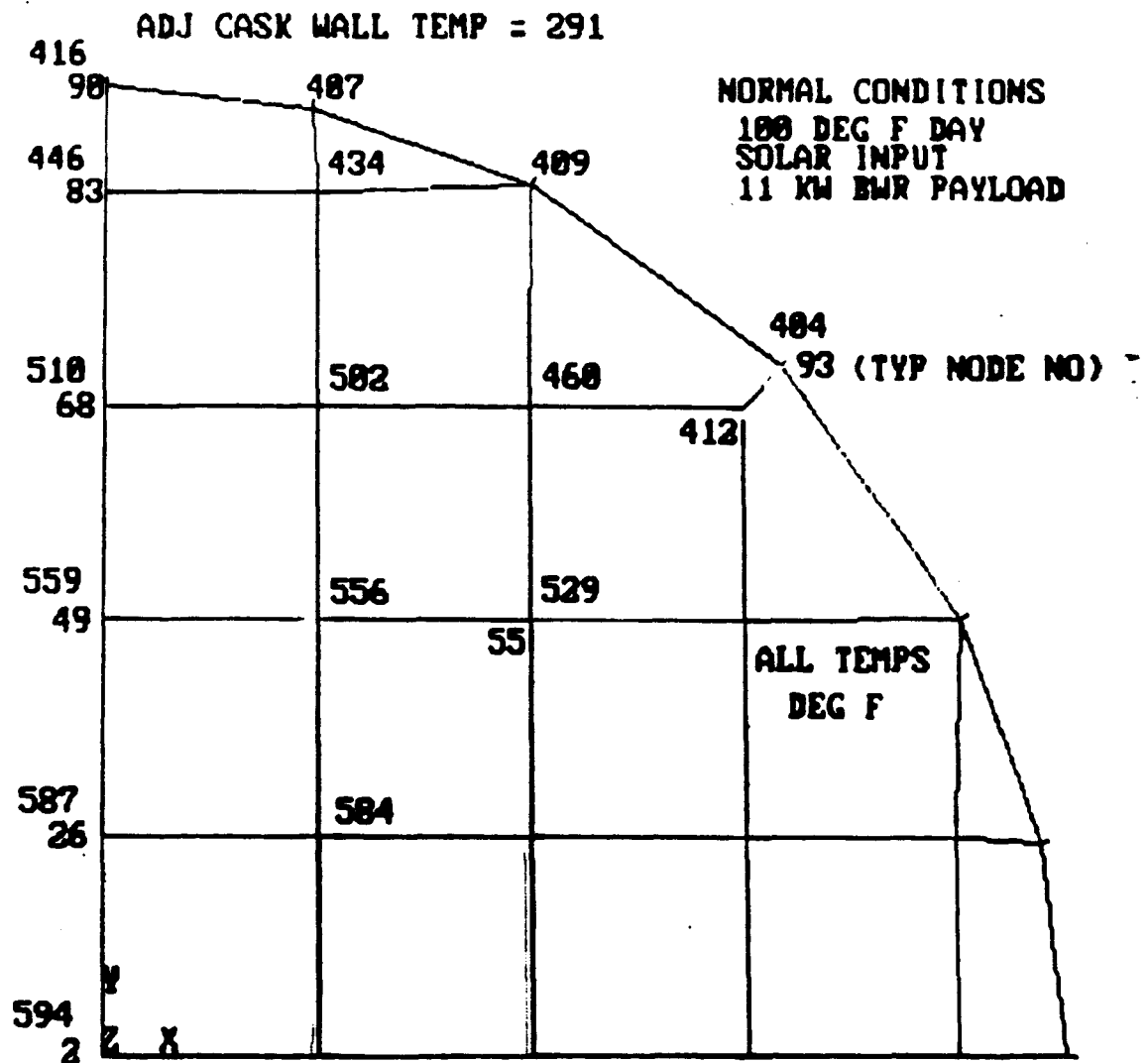


FIGURE 3.4.2-4
Normal Condition Temperature Distribution in BWR
Basket at a Location Between Impact Limiters

The temperature distribution in the fuel assembly thermal model discussed in Section 3.4.1.5 shall not be shown for normal conditions because the only contractual requirement for the spent fuel assemblies concerns the maximum fuel rod temperature, which will be subsequently shown to occur for accident conditions. Furthermore, only PWR fuel assemblies need be considered because the overall thermal resistance between the hotter central assembly fuel rods and the basket wall is significantly greater than that of the BWR assemblies. A typical 15 X 15 PWR fuel assembly has nearly twice as many rows of rods in between versus a typical 8 X 8 BWR fuel assembly.

3.4.3 Minimum Temperatures

The minimum temperature distribution for the NuPac 140-B Cask will occur with no decay heat load (empty basket) and an ambient temperature of -40°F (per 10 CFR 71.71(c)(2)). Since the steady state analysis of these conditions represents a trivial case, no computerized thermal calculations were performed. Instead, it was assumed that all cask components would reach the -40°F temperature under steady state conditions.

3.4.4 Maximum Internal Pressures

Maximum cask internal pressure for normal conditions is the sum of the fill gas pressure and residual water that may remain after drain operations. Fuel rod internal gas leakage effects will be considered for accident conditions. The fill gas will be assumed to be nitrogen initially at 70°F and 14.7 psia. From Nuclear Packaging's experience with the IF-300 cask, 1.0 cubic feet of residual water will be assumed.

Assuming ideal gas mixtures, Dalton's law of partial pressures says the total pressure of the mixture of steam and nitrogen inside the cask is the sum of the individual gas pressures evaluated at the temperature and volume of the mixture. Thus, the temperature and volume of the mixture are calculated as follows:

- o Mixture volume = 135.5 cubic feet, which assumes a 50% fill factor approximation for the payload.
- o Mixture temperature = 529°F (989 R), which is based on average internal surface temperatures of cask, basket and payload. The payload was assumed to be 21 PWR 17 X 17 fuel assemblies. The average surface temperatures of these three internal components were obtained from Table 3.4.2-2. Table 3.4.4-1 below summarizes the calculations for mixture temperature.

TABLE 3.4.4-1
CASK INTERNAL GAS MIXTURE TEMPERATURE CALCULATION

Internal Component	Area (sq-ft)	Temp (Deg F)	Product
o Cask Walls	263	290	760,000
o Basket Walls	1,139	468	533,000
o Fuel Rod Walls	6,514	550	3,583,000
SUMS	7,916		4,192,000

$$\text{MIXTURE TEMP} = 4,192,000 / 7,916 = 529^{\circ}\text{F} = 989 \text{ R}$$

The steam and nitrogen pressures are calculated from ideal gas laws as follows:

$$P_{\text{steam}} = \frac{(62.4 \text{ lbm})(85.8 \text{ ft-lbf/lbm-R})(989 \text{ R})}{[(144 \text{ sqin/sqft})(135.5 \text{ cuft})]} = 271.4 \text{ psia}$$

$$P_{\text{nitrogen}} = \frac{(14.7 \text{ psia})(989 \text{ R})}{(530 \text{ R})} = 27.4 \text{ psia}$$

Applying Dalton's law of partial pressures, the gas mixture pressure inside of the cask is:

$$P_{\text{internal}} = P_{\text{steam}} + P_{\text{nitrogen}} = 299 \text{ psia}$$

3.4.5 Maximum Thermal Stresses

Experience from Reference (5) for a similar cask shows that the primary regions of concern for thermal stress are at the cask cylindrical shell to end closure interfaces. For the NucPac 140-B Cask, the load collar also terminates at this interface and will be investigated for thermal stress. In addition, lid bolt thermal stress and thermal stress at the copper fin to outer cask wall attachment interface will be investigated. Thermal stresses calculated in this section will be mentioned in Section 2.6.1. Thermal stress in the baskets will be minimal due to inherent flexibility of the 'keyed panel' design. Detailed basket thermal stress analyses will be considered during the final design.

Differential thermal expansion between the cask and the baskets will be considered during the final design. Differential expansion between the borated-silicone neutron shield and its surrounding walls will be considered in this section and mentioned in Section 2.6.1. Finally, cask pressure stresses due to thermally induced internal pressures will be considered in Section 2.6.1. Table 3.4.5-1 below lists the mean coefficients of thermal expansion from 70°F for the materials of concern.

TABLE 3.4.5-1
Mean Coefficients of Expansion for Steel Cask Shells, Neutron
Shield, and Copper Fins ($\times 10^{-6}$ in/in $^{-\circ}\text{F}$)

Temperature (Deg F) (1)	XM-19 Steel Shells (1)	ASTM A-320, L43 Steel Bolts (2)	Boro-Silicone Neutron Shield (3)	Copper Fins
70	8.24	6.20	104	8.87
150	8.40	6.40	104	9.01
300	8.65	6.78	104	9.28
450	8.86	7.08	104	9.55

- (1) Reference (8)
(2) Reference (3)
(3) Reference (10)

3.4.5.1 Cask Shell and Load Collar to End Closures

The temperature distribution in a representative section of the cask shell to end closure interface is shown below in Figure 3.4.5-1.

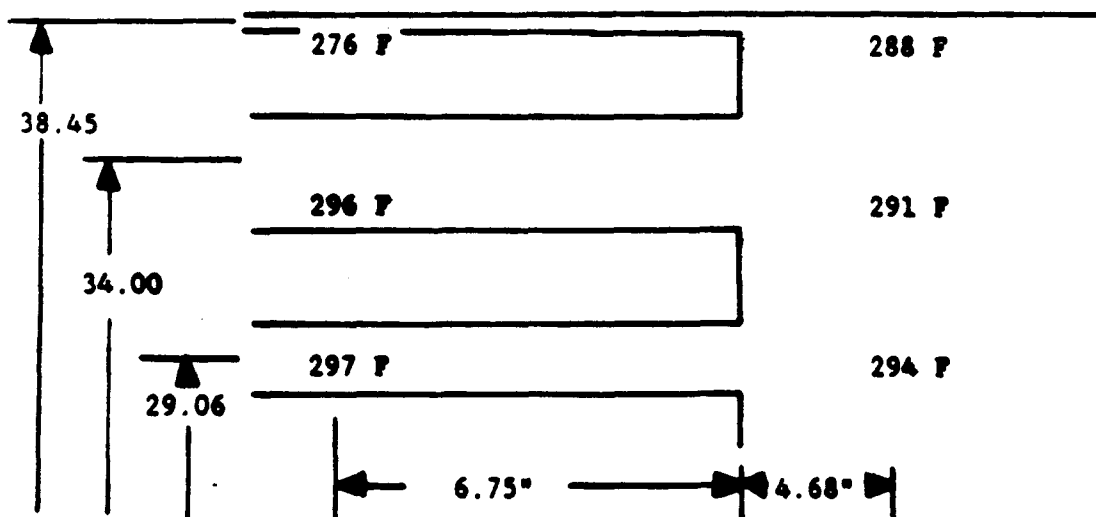


FIGURE 3.4.5-1
Cask Shell and Load Collar to End Closure Temperatures

Figure 3.4.5-1 shows three significant temperature differences for cask shells and load collar, namely:

- o Inner shell to end closure = $297 - 294 = 3^{\circ}\text{F}$
- o Outer shell to end closure = $296 - 291 = 5^{\circ}\text{F}$
- o Load collar to end closure = $288 - 276 = 15^{\circ}\text{F}$

Choosing a design temperature of 300°F , imposed radial deflections for the above three cask components are:

- o $\delta = R \alpha \Delta T = (29.06)(8.65 \times 10^{-6})(3) = -.000754$
- o $\delta = R \alpha \Delta T = (34.00)(8.65 \times 10^{-6})(5) = -.00147$
- o $\delta = R \alpha \Delta T = (38.45)(8.65 \times 10^{-6})(15) = +.00499$

The simplified free body diagram is depicted in Figure 3.4.5-2 below.

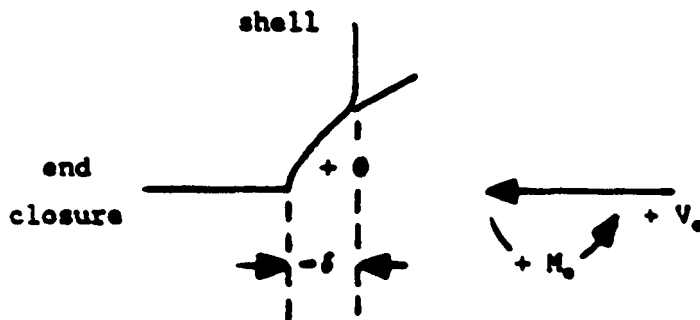


FIGURE 3.4.5-2
Free Body Diagram of Cask Shell/End Closure Interfaces

Rotations, Θ , at the shell ends are set equal to zero. From Reference (16),

$$\delta = -V_o/2D\lambda^3 + M_o/2D\lambda^2 \quad (5)$$

$$\Theta = -V_o/2D\lambda^2 + M_o/2D\lambda \quad (6)$$

where:

$$D = Et^3/12(1 - \mu^2) \quad (7)$$

$$\lambda = [3(1 - \mu^2)/R^2 t^2]^{.25} \quad (8)$$

$$D = (27.0 (10)^{-6}) (1.13)^3 / 12 [1 - (.3)^2]$$

$$= 3.57 (10)^{-6} \text{ lb-in}$$

$$\lambda = [3(1 - (.3)^2) / (29.06)^2 (1.13)^2]^{.25}$$

$$= .2243^{-1} \text{ in}$$

$$R = 29.06 \text{ in}$$

$$t = 1.13 \text{ in}$$

$$E = 27.0 (10)^6 \text{ psi @ 300 F (Reference (h))}$$

$$\mu = 0.3$$

$$\delta = 0.000793 \text{ in}$$

$$\Theta = 0 \text{ radians}$$

Simultaneously solving for V_o and M_o from equations (5) and (6) yields:

$$V_o = 122 \text{ lb/in}$$

$$M_o = 272 \text{ in-lb/in}$$

From Reference (16):

$$\sigma_1 = E \alpha \Delta T \quad (\text{meridional membrane})$$

between shells

$$= (27.0 (10)^6) (8.65 (10)^{-6}) (297 - 296 \text{ F}) = 234 \text{ psi}$$

$$\sigma_2 = -2V_o \lambda R/t + 2M_o \lambda^2 R/t = -704 \text{ psi (hoop membrane)}$$

$$\sigma_1' = -6M/t^2 = -1445 \text{ psi (meridional bending)}$$

$$\sigma_2' = \mu \sigma_1' = -433 \text{ psi (hoop bending)}$$

Similar calculations were performed for the outer cask shell and the load collar and the results summarized in Table 3.4.5-2.

TABLE 3.4.5-2
Thermal Stress Summary for Cask Shells and Load Collar

Cask Component	Wall Thickness	Membrane Stress		Bending Stress		
		<u>Meridional</u>	<u>Hoop</u>	<u>Meridional</u>	<u>Hoop</u>	<u>Hoop</u>
o Inner Shell	1.13 in	-234 psi	-704	-1,445	-433	
o Outer Shell	1.50	234	-1,164	-2,109	-633	
o Load Collar	0.25	4,671	3,524	6,384	1,915	

3.4.5.2 Copper Fin to Outer Cask Shell Interface

As shown in Table 3.4.5-1, there is a noticeable difference in coefficients of thermal expansion between cask shell and fin materials. Choosing a design temperature of 300°F, the relative radial deflection between the cask wall and attached copper fin imposed upon the fin is:

$$\begin{aligned}
 \delta &= R(\alpha_{cu} - \alpha_{ss})(T_{intf} - T_{ref}) \\
 &= (35 \text{ in})(9.28(10)^{-6} - 8.65(10)^{-6} \text{ in/in-F})(300^\circ\text{F} - 70^\circ\text{F}) \\
 &= .00507 \text{ in}
 \end{aligned}$$

The simplified free body diagram is depicted in Figure 3.4.5-3 below. It shows, based on symmetry, that the fin is not allowed to expand out from the steel shell attachment point, which imposes both bending and compression in the fin. The portion of the fin affected can then be modeled as a simple cantilever beam with forces imposed as shown:

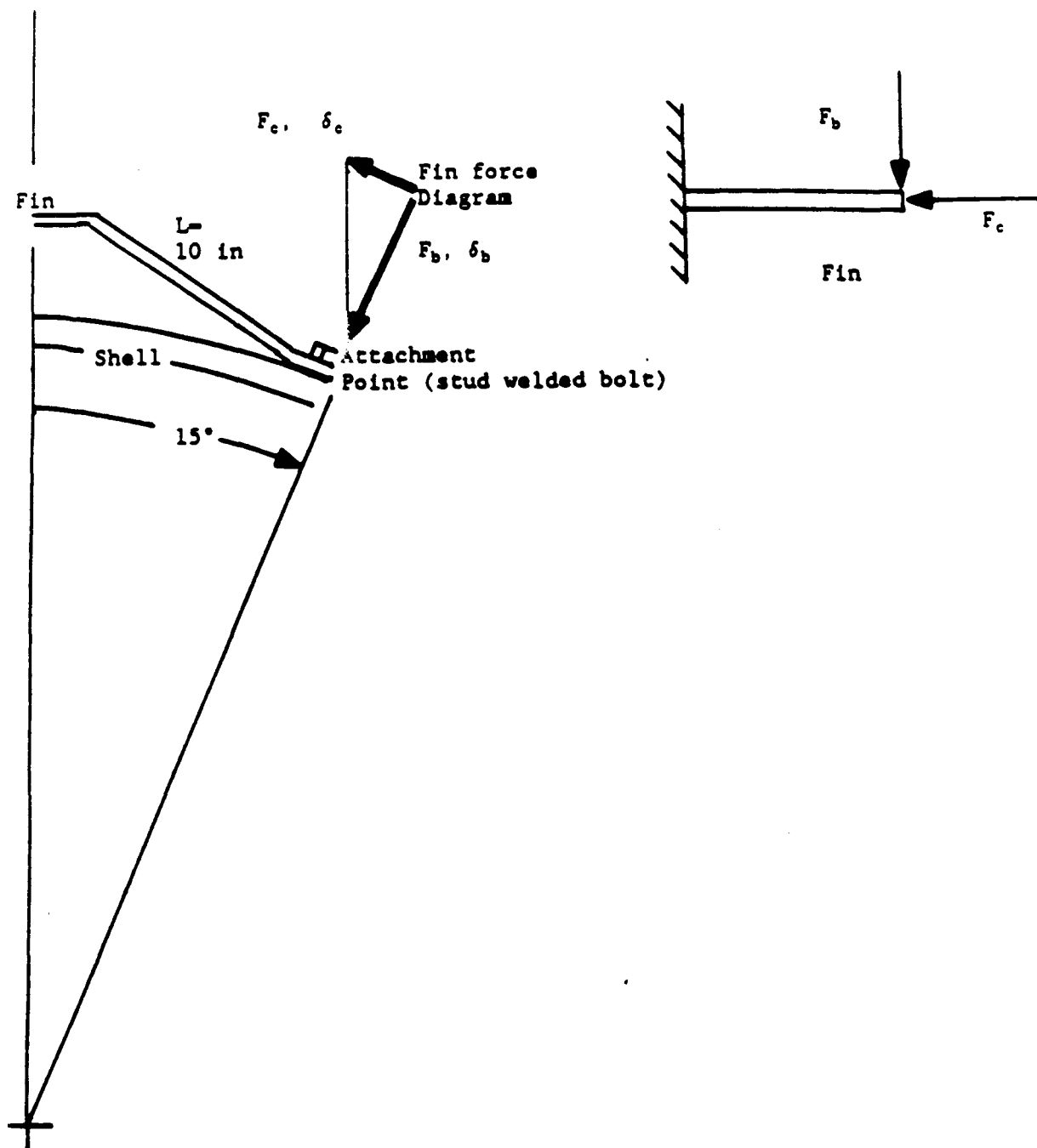


FIGURE 3.4.5-3
Free Body Diagram of Fin/Shell Interface

The thermal stress due to compression is calculated as follows:

$$\sigma_c = (.00507 \text{ in}) (\tan 15^\circ)$$

$$= .00136 \text{ in}$$

$$\sigma_c = \epsilon E$$

$$\sigma_c = (.00136 \text{ in}) (16.3 \times 10^6 \text{ psi}) / (10 \text{ in})$$

$$= 2,216 \text{ psi}$$

$$\epsilon = \text{strain in symmetrical section of fin, } \delta_c / L$$

$$E = \text{modulus of copper at 300 F (Reference (8))}$$

The thermal stress due to bending is calculated as follows:

$$\begin{aligned} \sigma_b &= Mc/I = (2560 \text{ in-lbs}) (.25 \text{ in}) / (1.03 \text{ in}^4) \\ &= 621 \text{ psi} \end{aligned}$$

$$\begin{aligned} I &= bh^3/12 = (99 \text{ in fin length}) (.5 \text{ in})^3 / 12 \\ &= 1.03 \text{ in}^4 \end{aligned}$$

$$\begin{aligned} M &= F_b L = (.00507 \text{ in}) (3) (16.3 \times 10^6 \text{ psi}) (1.03 \text{ in}^4) / (10 \text{ in})^2 \\ &= 2,560 \text{ in-lbs} \end{aligned}$$

$$c = .25 \text{ in}$$

The thermal stress imposed on the half inch copper fins at the "stud welded bolt" attachment point shown in Figure 3.4.5-3 above is calculated as follows:

$$\delta = t(\alpha_{cu} - \alpha_{ss})(T_{intf} - T_{ref})$$

$$\epsilon = \delta/t = (.63 (10)^{-6}) (300 - 70 \text{ F}) = 1.449 (10)^{-4}$$

$$\sigma = \epsilon E = (1.449 (10)^{-4}) (16.3 (10)^6 \text{ psi})$$

$$= 2,362 \text{ psi}$$

In sum, the copper fins will stay well within their elastic limit for normal conditions because Reference (8) gives an allowable stress of 5000 psi at 300°F, which is 33% lower than actual yield stress. Thus, for normal conditions the heat dissipation function of the copper fins will not be degraded due to loosening at the "stud welded bolt" attachment point as a result of thermal cycling.

3.4.5.3 Cask Lid Bolted Interface

As shown in Table 3.4.5-1, there is a noticeable difference in coefficients of thermal expansion between cask XM-19 and bolt ASTM A-320 L43 steel materials. Choosing a design temperature of 300°F, the relative longitudinal deflection between the cask lid flange and bolts imposes additional tension in the bolts and the resulting stress is calculated below.

$$\begin{aligned} \sigma &= (\alpha_{\text{cask}} - \alpha_{\text{bolt}}) (T_{\text{intf}} - T_{\text{ref}}) (E_{\text{bolt}} @ 300 \text{ F}) \\ &= (8.65(10)^{-6} - 6.78 (10)^{-6} \text{ in/in-F}) (300 \text{ F} - 70 \text{ F}) (26.7 (10)^6) \\ &= 11,484 \text{ psi} \end{aligned}$$

where it is assumed that the bolt and cask at the interface are at the same temperature. Analysis verifies this assumption with respective bolt (node 135) and cask interface (node 117) both being at 293°F.

3.4.5.4 Relative Expansion of Neutron Shield

As shown in Table 3.4.5-1, there is a significant difference in coefficients of thermal expansion between cask shell and fin, and the borated-silicone neutron shield materials. With the borated-silicone

expansion coefficient being ten times greater, segments of the material will have to be designed to fit loose when initially installed within the copper fins. The effects of this initial loose fit on radiation shielding will be considered during the final design phase of the program. Choosing a design temperature of 300 °F, radial tolerance and longitudinal tolerance recommendations are calculated as follows:

Radial tolerances	= $\alpha L(T_{intf} - T_{ref})$
	= $(104 (10)^{-6} \text{ in/in } ^\circ\text{F})(5.9 \text{ in})(300 - 70^\circ \text{ F})$
	= .141 in, for height of triangular segments
	= $(104 (10)^{-6} \text{ in/in } ^\circ\text{F})(49.5)(300 - 70^\circ \text{ F})$
	= 1.18 in, for diameter of end closure segments
	= $(104 (10)^{-6})(18.2)(230^\circ\text{F})$
	= .435 in, for circumferential growth along long side of triangular segments

Longitudinal tolerance	= $\alpha L(T_{intf} - T_{ref})$
	= $(104 (10)^{-6})(177)(230^\circ\text{F})$
	= 4.23 in, for total length along cask of triangular segments
	= $(104 (10)^{-6} \text{ in/in } ^\circ\text{F})(2.5)(300 - 70^\circ \text{ F})$
	= .060 in, for thickness of end closure segments

The segment tolerances are depicted in Figure 3.4.5-4. To obtain the longitudinal tolerance, the borated-silicone segments will be broken into a number of short lengths with stepped overlapping joints.

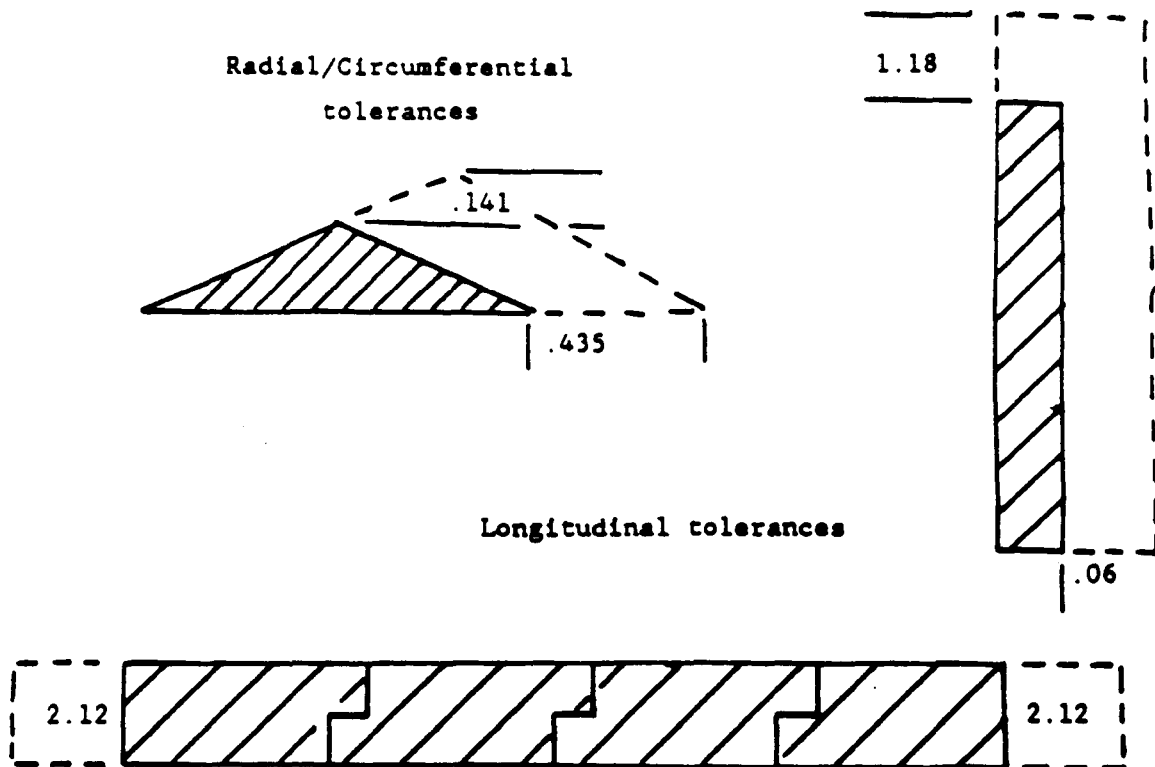


FIGURE 3.4.5-4
Borated-Silicone Neutron Shield Tolerances

3.4.6 Evaluation of Package Performance For Normal Conditions Of Transport

The component temperatures associated with the more extreme normal thermal conditions presented in Sections 3.4.2 and 3.4.3 are all within allowable limits for the respective materials (see Section 3.3). As input to Section 2.0, the minimum temperature for any cask component is taken as -40°F (-20°F when combined with other load cases), and the maximums as 300°F for cask shells, 335°F for cask end closures, 280°F for impact limiters, 580°F for PWR basket, and 595°F for BWR basket.

Regarding thermal expansion and stress, the heat dissipation function of the copper fins will not be degraded due to loosening at their "stud welded bolt" attachment points as a result of thermal cycling. Additional tension in the lid attachment bolts will not be excessive. Also, the borated-silicone neutron shield segments will be toleranced for proper fit.

Regarding pressure, a 299 psia internal pressure is considered maximum for any cask component.

3.5 Hypothetical Accident Thermal Evaluation

This section presents the thermal analyses of the NuPac 140-B Cask and Fuel Package for the hypothetical thermal test condition specified in 10 CFR 71.73 (c) (3). The initial temperature distribution in the cask prior to the hypothetical accident thermal event is taken as that corresponding to the 100°F steady state condition without insolation with 11 KW decay heat as discussed in Section 3.4. This is in accordance with 10 CFR 71.73 (b). To determine the effect of the hypothetical thermal test condition, the cask is exposed to 1475°F for a half hour at which time the thermal boundary is returned to a 100°F ambient air condition as specified in 10 CFR 71 (c) (3). The transient is then continued for a time sufficient to determine maximum values for all temperatures within the cask. A time limit of twenty hours was used in these analyses as the time period permitted most cask components to reach their maximum temperatures and allowed upper bound temperatures to be determined for the rest. This time limit will be extended as required during final design thermal analyses.

3.5.1 Thermal Models

The analytical models used to evaluate the hypothetical accident conditions are identical to those described in Sections 3.4.1.1, 3.4.1.2, 3.4.1.3, 3.4.1.4, and 3.4.1.5. Damage conditions, such as pin puncture of the thermal and neutron shields, side drop indentations of the impact limiters, and general drop damage to the copper fins will be considered during the final design. At this time, it is felt that such damage conditions will not be found to adversely affect any thermal/structural aspects of the cask because initial equilibrium temperatures before the fire are significantly lower than those found for normal conditions due to the absence of insolation.

The polyurethane foam that is being considered for the NuPac 140-B Cask is the same formulation, FR-3700, as used for the impact limiters of the similar cask of Reference (5). The material decomposes at about 400°F,

but Nuclear Packaging experience and test results, such as those presented in Reference (5), show that such decomposition terminates after the hypothetical accident thermal event, will occur only locally at external surfaces on the impact limiters, and will penetrate into the foam four to six inches which is insignificant. The same testing showed that minor tears or ruptures of the impact limiter shells were of no thermal consequence.

The char layer is not taken into account in the thermal models. Instead temperatures in the foam during the hypothetical accident thermal event are calculated as if the foam could be hotter than 400°F. This is conservative because charring blocks heat flow to the cask via heat absorbed in change of phase.

The borated-silicone was tested under 1475°F conditions by the Los Alamos Scientific Laboratory as reported in Reference (17). The pretest neutron attenuation factor for a 7.9 inch slab of as received material was about 17 and the post test factor ranged from 11 to 12.5. Weight loss during the test was 15%. The material is self extinguishing, and no deleterious gaseous char products have been reported. As will be shown subsequently from the thermal analysis, the neutron shield material only reaches 553 F during the hypothetical thermal test. A damaged region with exposed material locally would get hotter. However, like the polyurethane foam, this material experiences change of phase near 650°F, and as previously mentioned, the material is self extinguishing.

The half axisymmetric finite element model is depicted in Figure 3.5.1-1 below. The model is identical to that shown for normal conditions in Figure 3.4.1-1 except the personnel barrier is absent and no solar heat input is present.

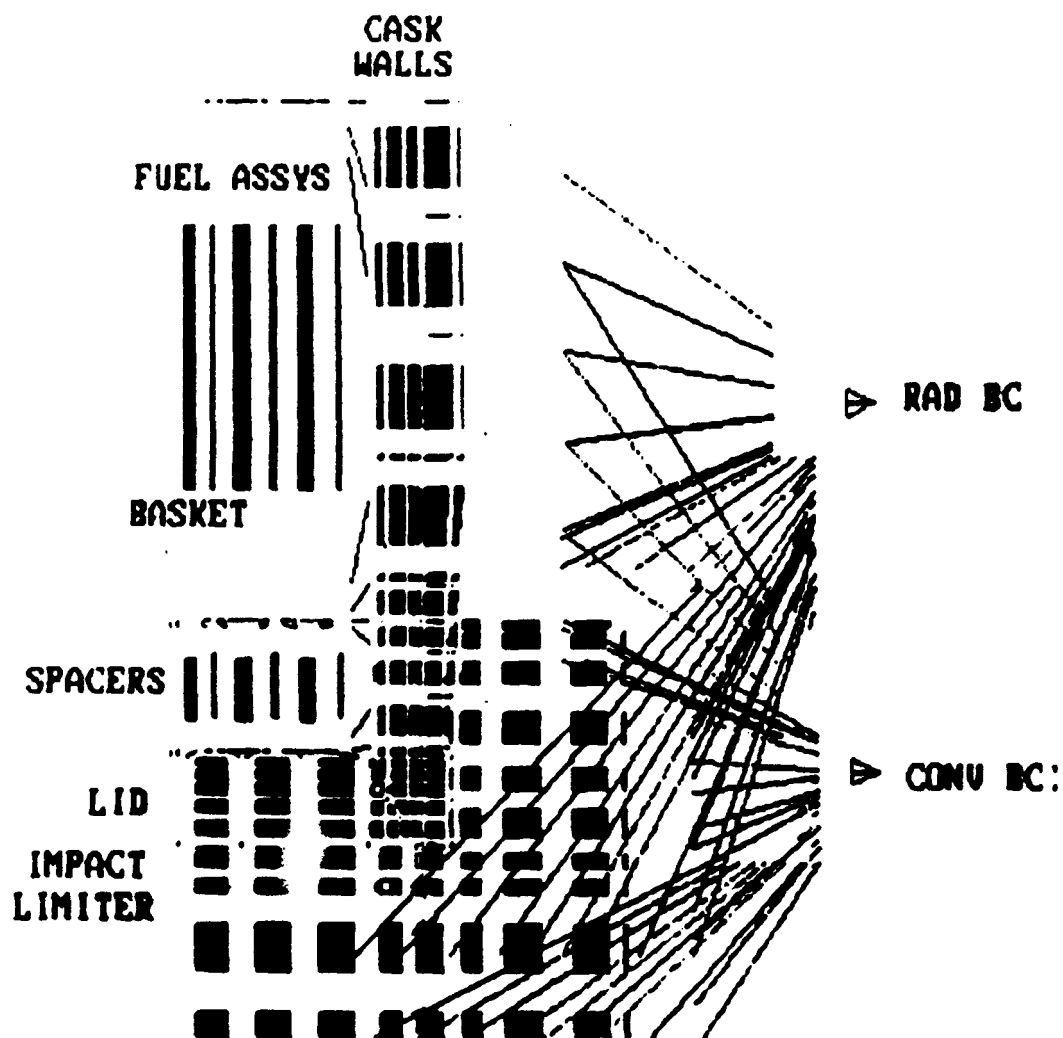


FIGURE 3.5.1-1
Exploded View of Half Axisymmetric Model of Cask with
Impact Limiter (Accident Conditions)

3.5.2 Package Conditions and Environment

Damage conditions, such as pin puncture of the thermal shield and side drop indentations of the impact limiters were not found to present any problems for a similar cask, as reported in Reference (5), and are not expected to present any problems for the NuPac 140-B Cask. The primary concern for the NuPac 140-B Cask are the O-ring seals, which are at 300°F for normal conditions as mentioned in Section 3.4.2. The above two damage conditions will definitely result in increased seal temperatures, however, initial equilibrium temperatures before the hypothetical accident thermal event are significantly lower, 248 °F, than those found for normal conditions due to the absence of insolation.

General drop damage to the copper fins will be considered during the final design. At this time, it is felt that such damage conditions will not be found to adversely affect any thermal/structural aspects of the cask.

3.5.3 Package Temperatures

The maximum temperatures noted for hypothetical accident conditions of transportation (i.e. 100 °F ambient, 30 min 1475 °F thermal test condition, and 11 KW payload decay heat) are presented in Table 3.5.3-1 below for the critical cask components listed in Table 3.1-2. It will be noted that critical cask component temperatures are not exceeded. It is also interesting to compare Table 3.5.3-1 with Table 3.4.2-1 and note that the O-ring seals are 28 °F hotter for normal conditions.

As previously mentioned, a steady state analysis without solar input is run prior to the 30 minute hypothetical thermal test condition. These temperature results are used for structural design for accident condition drop analyses and are presented in Table 3.5.3-2 below for other cask components of significance. A complete nodal listing of the cask temperature distribution is presented in Appendix 3.6.6. It should be mentioned that the results in Table 3.5.3-2 are comparable to the results previously presented in Table 3.4.2-2.

TABLE 3.5.3-1
Accident Condition Thermal Analysis Results for Critical
Cask Components

	O-RING	LIMITER	NEUTRON	FUEL	PERSONNEL
CASK DESIGN BASIC	SEAL	FOAM	SHIELD	CLAD	BARRIER
CONFIGURATION	TEMP	TEMP	TEMP	TEMP	TEMP
(TEMPERATURE LIMITS FOR CASK APPLICATION)	350°F	300°F	600°F	716°F	N/A
****10 CFR 71 ACCIDENT CONDITIONS*****					
CONFIGURATION EMPLOYS	272°F	280°F	553°F	682°F	N/A
PASSIVE THERMAL MANAGEMENT	(NODE)	(NODE)	(SEE)	(SEE)	
PER TABLE 3.1	(138)	(200)	(NOTE 1)	(NOTE 2)	

NOTE 1: This value was obtained from the thermal model of Section 3.4.1.2. An average value of 518°F is obtained from the thermal model of Section 3.4.1.1.

NOTE 2: This value was obtained from the thermal model of Section 3.4.1.5. All other values in this table were obtained from the thermal model of Section 3.4.1.1.

TABLE 3.5.3-2
 Accident Condition Thermal Analysis Results for Other
 Cask Components (Prior To 30 Min, 1,475 °F
 hypothetical thermal test condition)

THERMAL MODEL UTILIZED		TEMPERATURE RANGE (DEG F)		
o Inner steel shell	Section 3.4.1.1	244	to	222
o Lead gamma shield	"	242	to	217
o Outer steel shell	"	240	to	214
o Copper fins	"	239	to	205
o Boro-silicone neutron shield	"	239	to	205
o Thermal shield	"	174	to	166
o Load collar	"	236	to	173
o End closure forgings	"	241	to	220
o Cask lid	"	281	to	243
o O-ring seals	"	248	to	242
o Impact limiter	"	225	to	101
o Fuel assy spacers	"	335	to	260
o PWR basket	Section 3.4.1.3	530	to	304
o BWR basket	Section 3.4.1.4	548	to	355
o Fuel rods	Section 3.4.1.5	609	to	383

Figures 3.5.3-1 through 3.5.3-6 present transient temperature plots for significant cask components showing the time at which peak values occur. Basket temperatures in Figures 3.5.3-5 and 3.5.3-6 are based on homogenized assumptions for fuel assemblies and basket structure and are not to be used for design. They are presented for information and to illustrate basket and fuel assembly results obtained from the axisymmetric model of Section 3.4.1.1. The detailed models of Sections 3.4.1.3 for PWR basket, 3.4.1.4 for BWR basket, and Section 3.4.1.5 for the central PWR fuel assembly are to be used for basket design and fuel clad temperature verification.

Figure 3.5.3-2 showed a maximum neutron shield temperature of 518° F based upon the axisymmetric model of Section 3.4.1.1. This temperature is really an average because of the assumption that the copper fins and neutron shield conduct heat in parallel. To check local two dimensional effects, which show the true temperature distribution in the neutron shield, the thermal model of Section 3.4.1.2 was formulated. Figures 3.5.3-7 through 3.5.3-9 give local transient temperature results for the two dimensional more detailed thermal model of the cask wall/neutron shield region. Figure 3.5.3-7 shows node locations associated with the transient temperatures of Figures 3.5.3-8 and 3.5.3-9. Nodal temperatures 18 and 20 in Figure 3.5.3-8 show that the 518° F previously mentioned neutron shield temperature really ranges from 475° F to 553° F.

Basket temperature transients are given in Figures 3.5.3-10 for PWR, and 3.5.3-11 for BWR. For the basket nodal locations associated with these transients, refer to Figures 3.4.1-4 and 3.4.2-3 respectively. It is interesting to note that the maximum basket temperatures in the center are not very different for normal and accident conditions; namely, 580° F and 602 F for the PWR, 594° F and 595° F for the BWR respectively. It is also important to note that structural analysis of the baskets for accident drop loads is to be based on initial temperatures taken at time zero from Figures 3.5.3-10 and 3.5.3-11.

Finally, Figure 3.5.3-12 is presented to verify that the fuel cladding temperature limit of 716° F specified in Reference (1) is not exceeded. As previously mentioned in Section 3.4.2, the temperature distribution

in the fuel assembly thermal model discussed in Section 3.4.1.5 was not shown for normal conditions because the only contractual requirement for the spent fuel assemblies concerns the maximum fuel rod temperature. This value is shown in Figure 3.5.3-12 to be 682°F. It was conservatively calculated assuming equilibrium conditions with the maximum PWR basket temperature of 602°F from Figure 3.5.3-10 used as the central tube wall boundary temperature. Furthermore, only PWR fuel assemblies need be considered because the overall thermal resistance between the hotter central assembly fuel rods and the basket wall is significantly greater than that of the BWR assemblies. A typical 15 X 15 PWR fuel assembly has nearly twice as many rows of rods in between versus a typical 8 X 8 BWR fuel assembly.

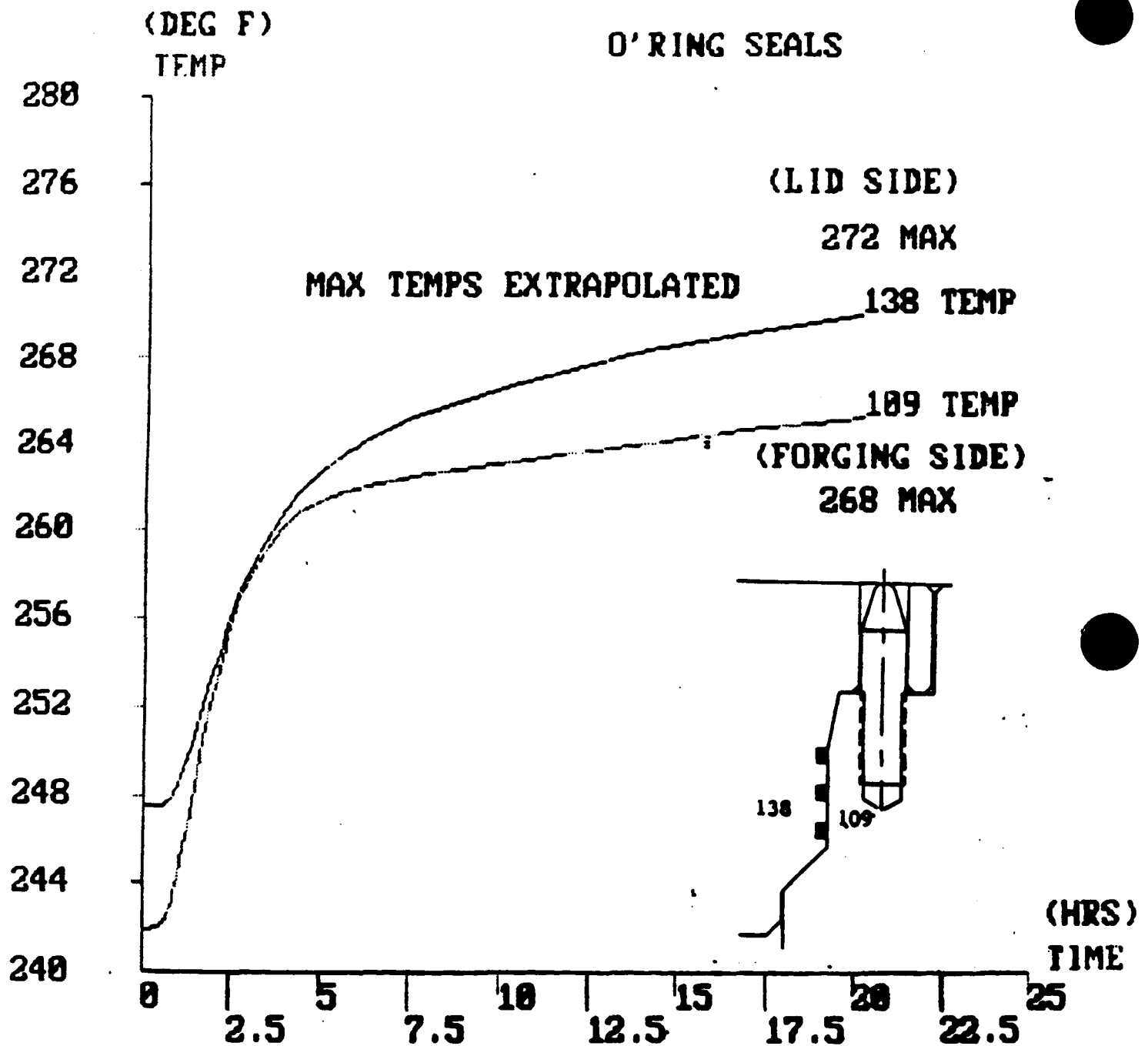


FIGURE 3.5.3-1

O-Ring Seal Temperature Transients for Accident Conditions

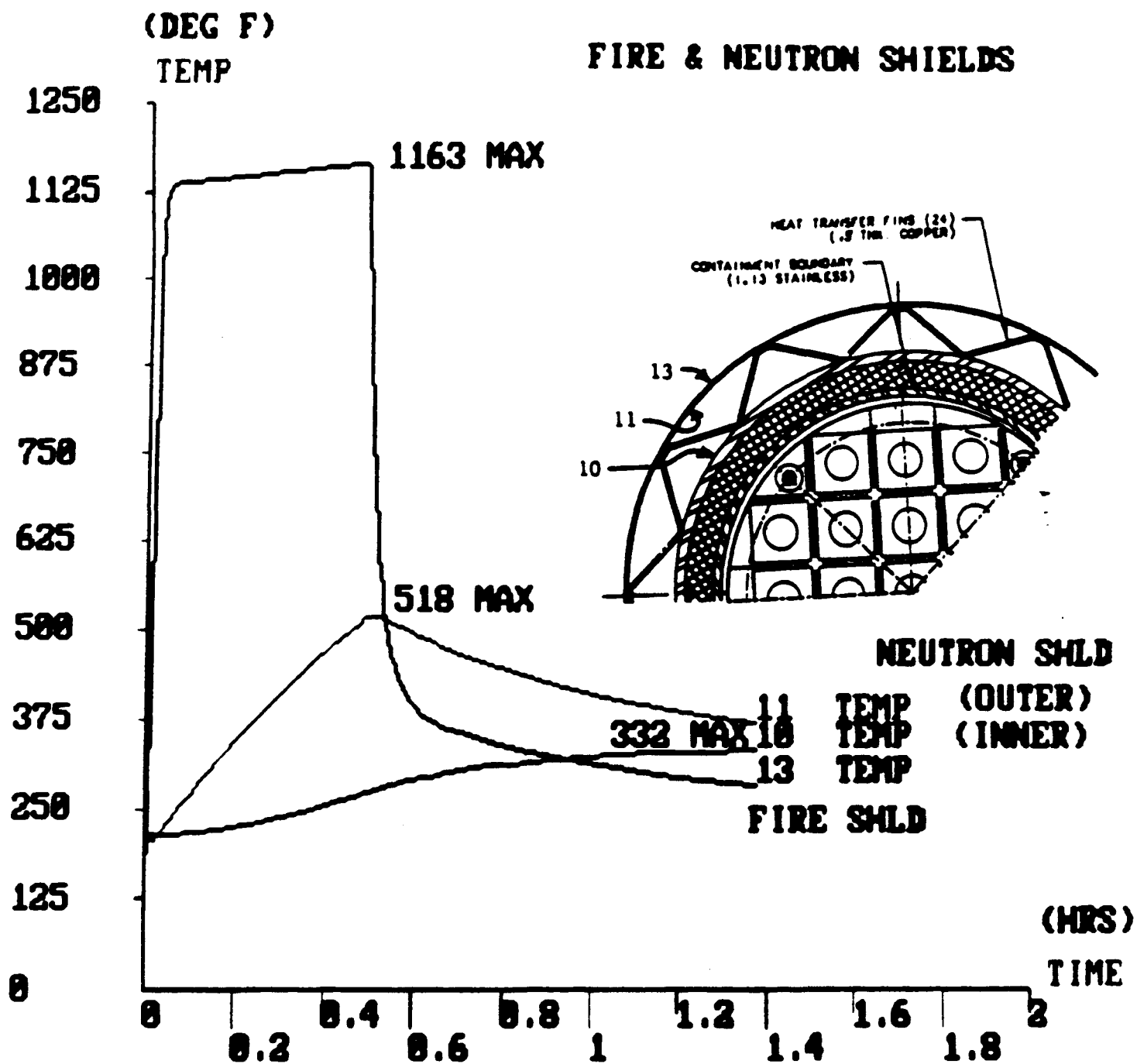


FIGURE 3.5.3-2
Fire, Neutron Shield & Outer Cask Wall Temperature
Transients for Accident Conditions

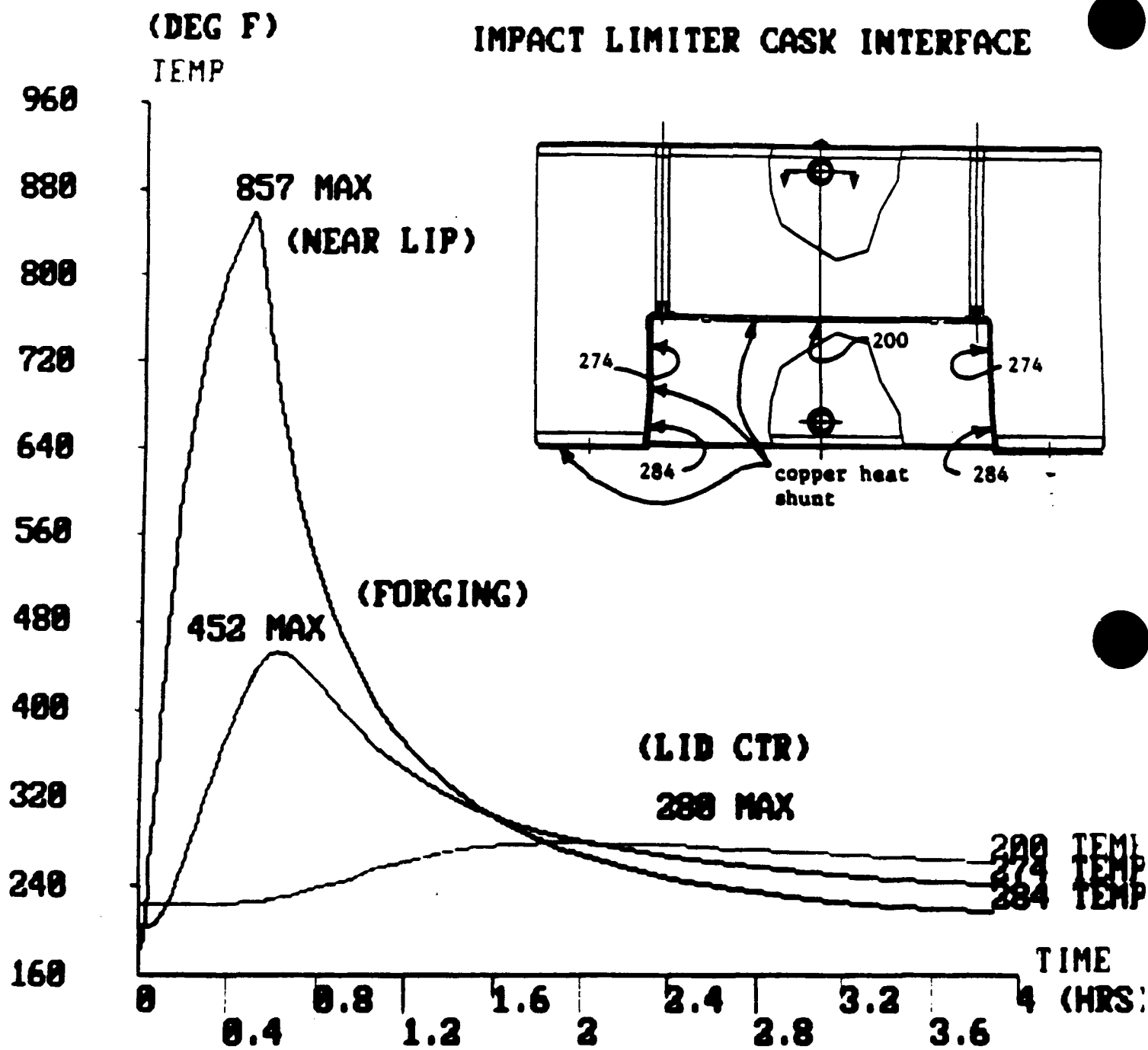


FIGURE 3.5.3-3
Impact Limiter Copper Liner Heat Shunt Temperature
Transients for Accident Conditions

LEAD GAMMA SHIELD

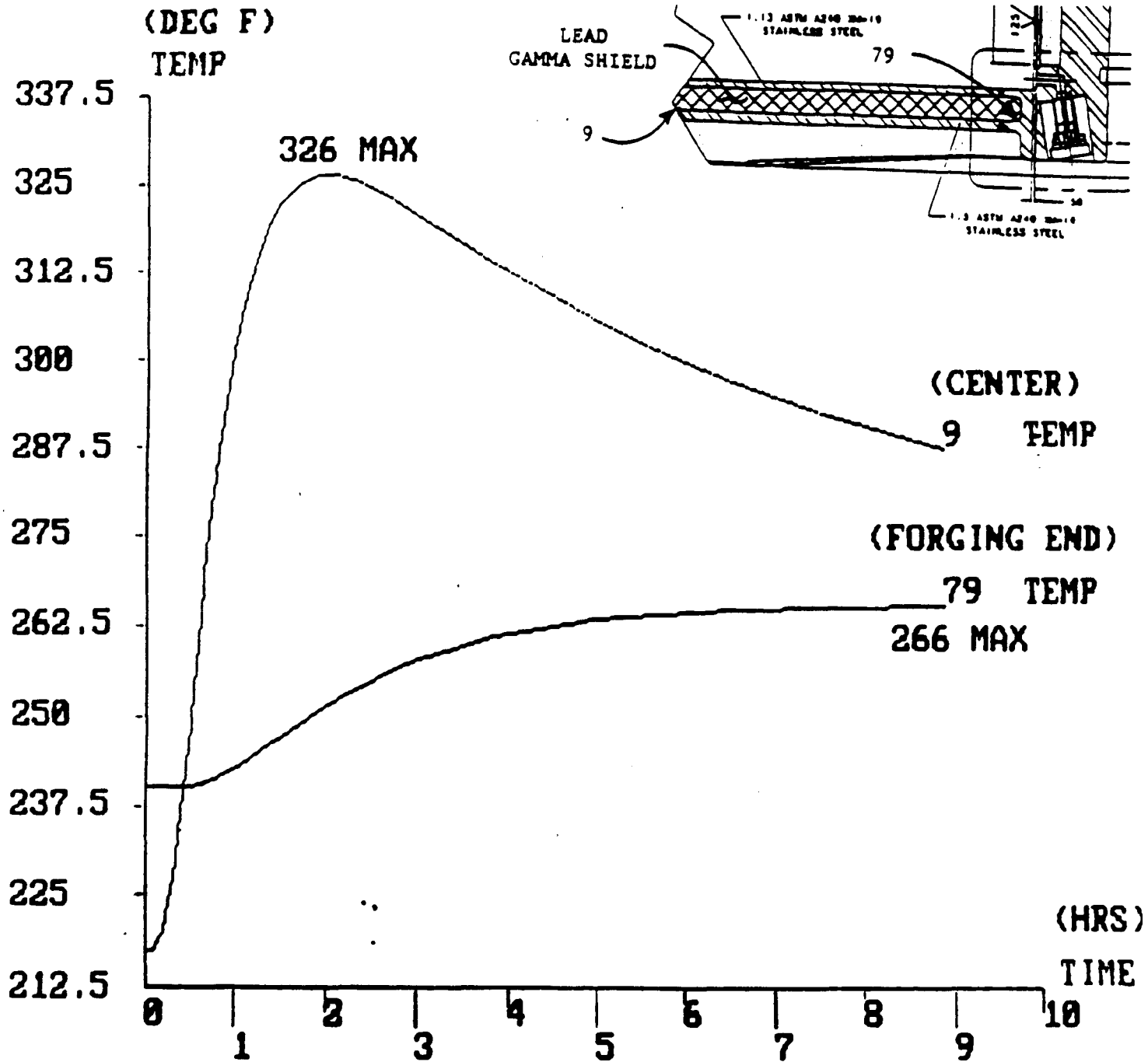


FIGURE 3.5.3-4

Lead Gamma Shield Temperature Transients for Accident Conditions

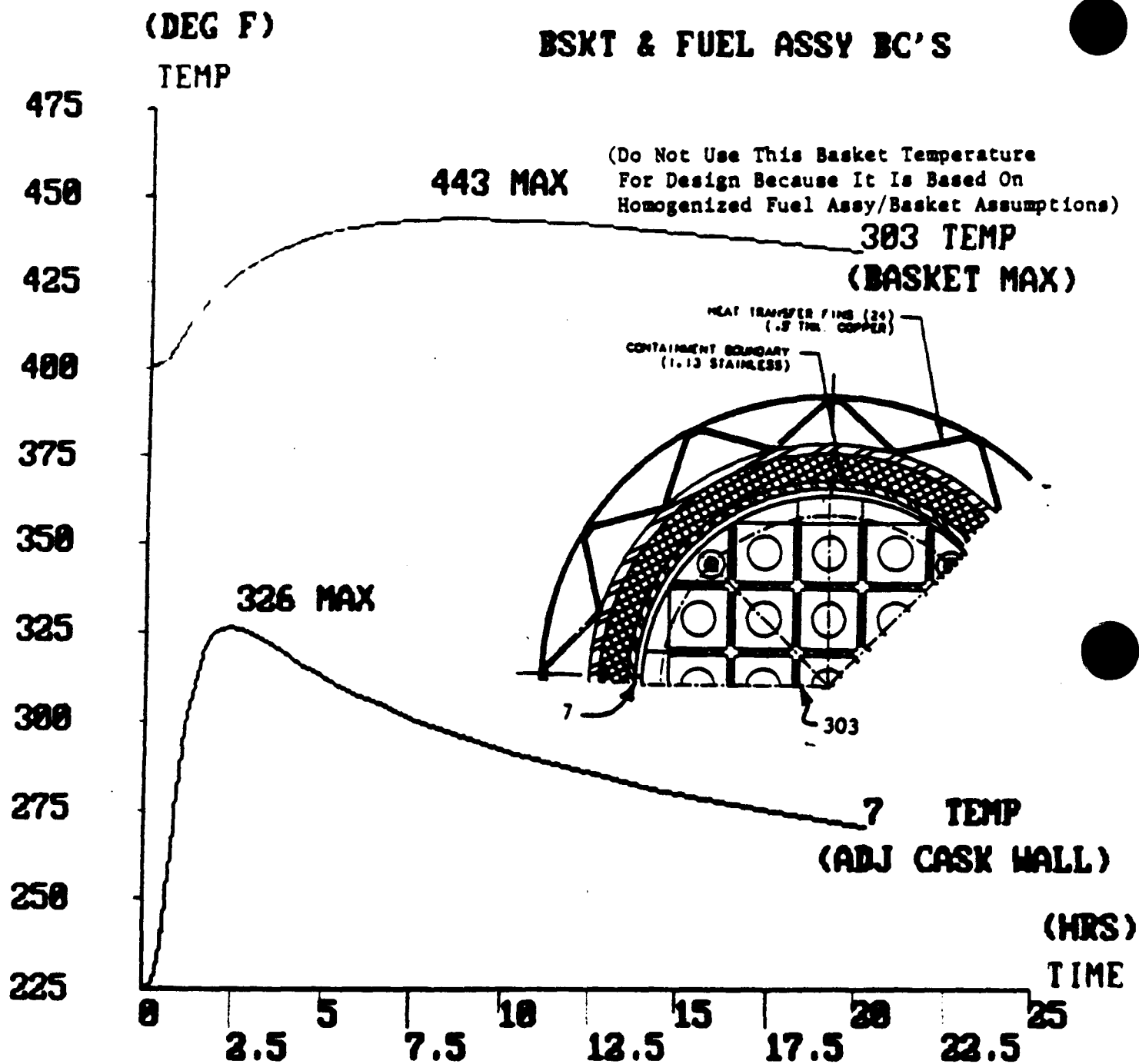


FIGURE 3.5.3-5

Inner Stainless Steel Cask Wall and Max Homogenous Basket
Wall Temperature Transients for Accident Conditions

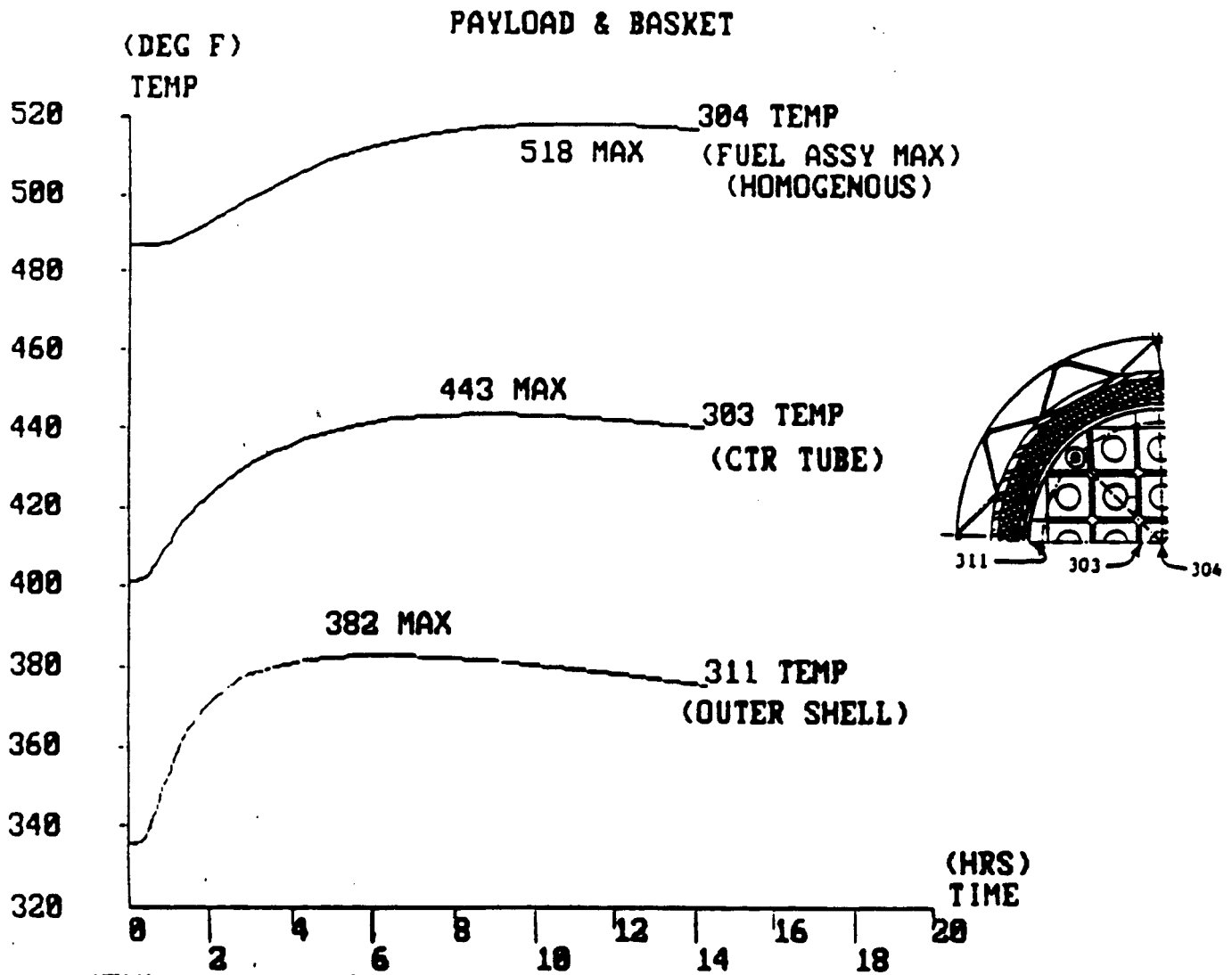


FIGURE 3.5.3-6

Max Homogenous Fuel Assembly, and Max & Min Homogenous
Basket Wall Temperature Transients for Accident Conditions

NODE LOCATIONS FOR TEMP VS TIME DATA

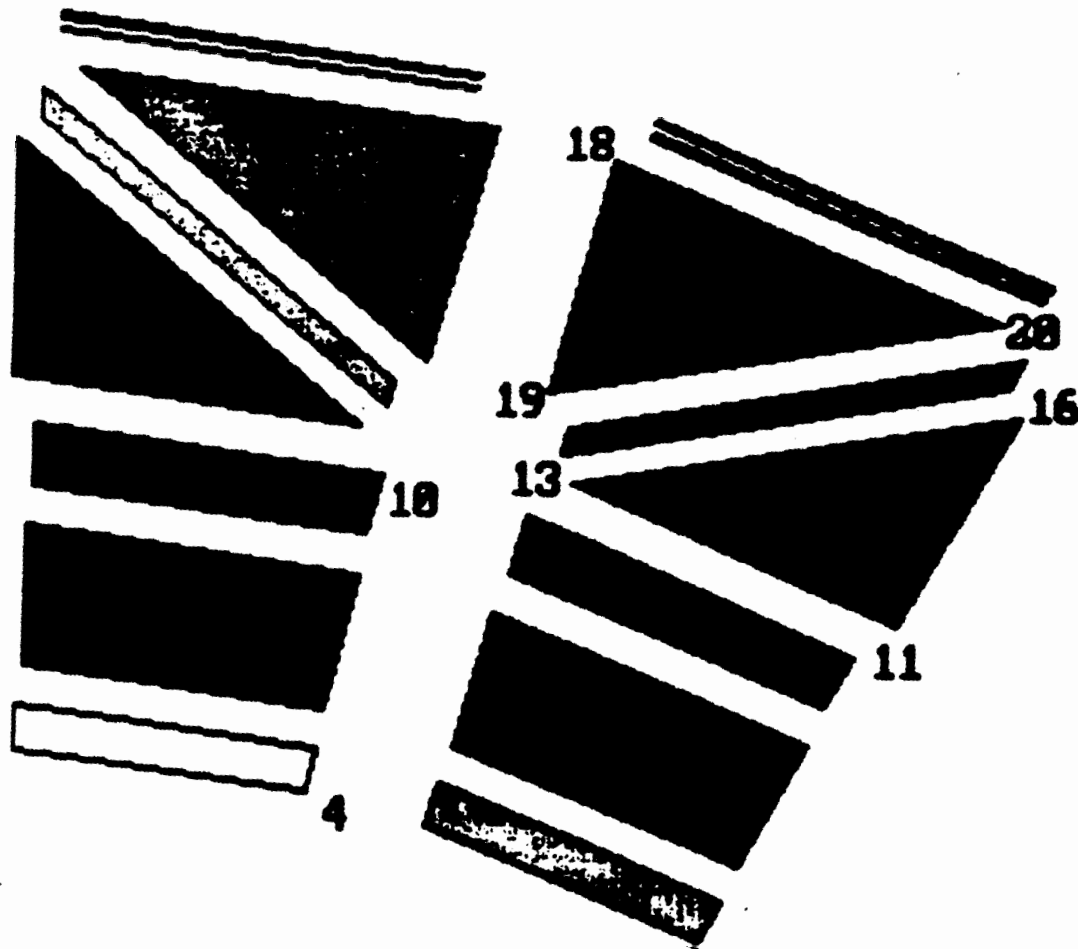


FIGURE 3.5.3-7
Cask Wall Region Nodal Definition for Temperature
Transients for Accident Conditions

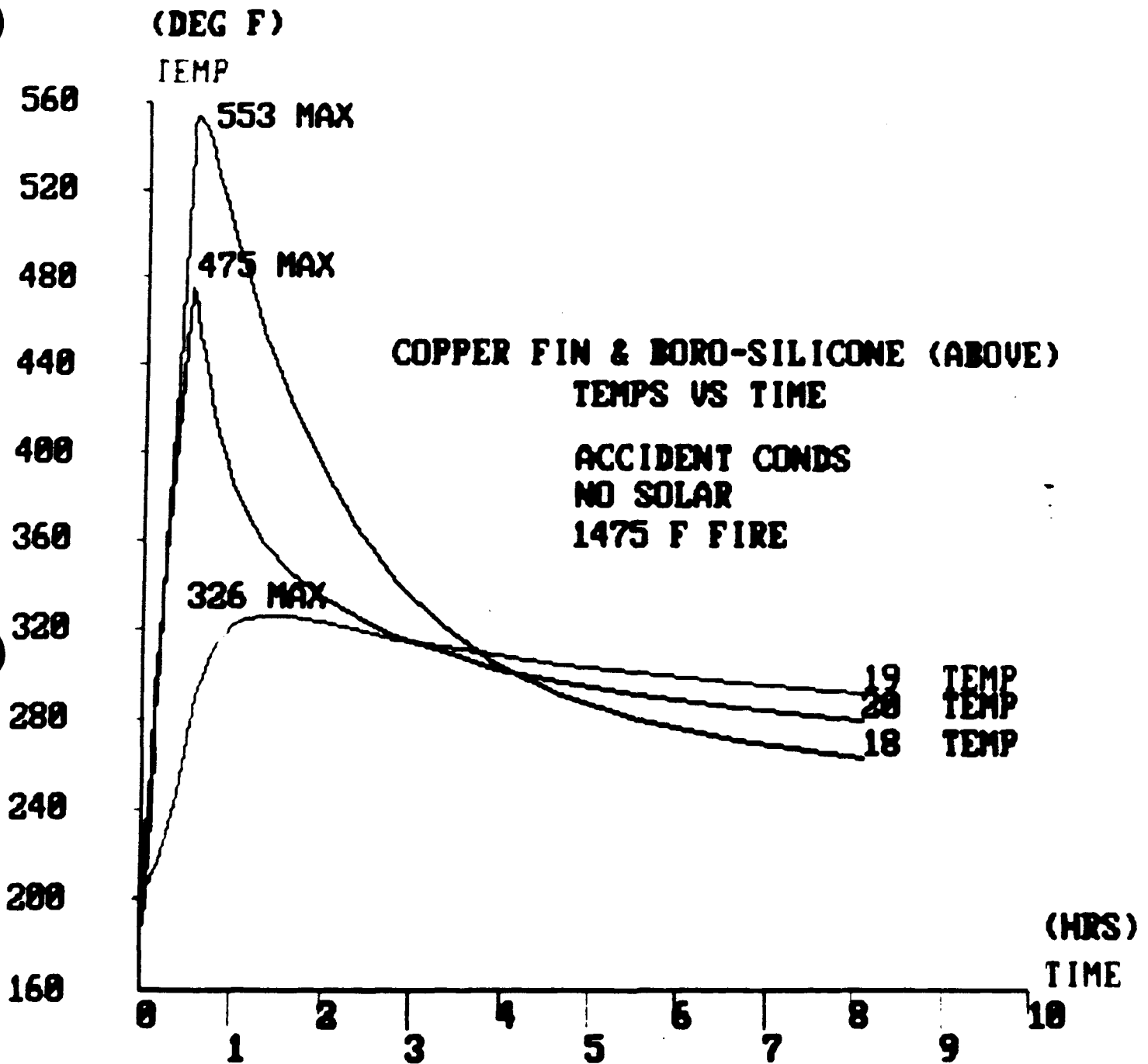


FIGURE 3.5.3-8
Neutron Shield & Copper Fin Temperature Transients for
Accident Conditions

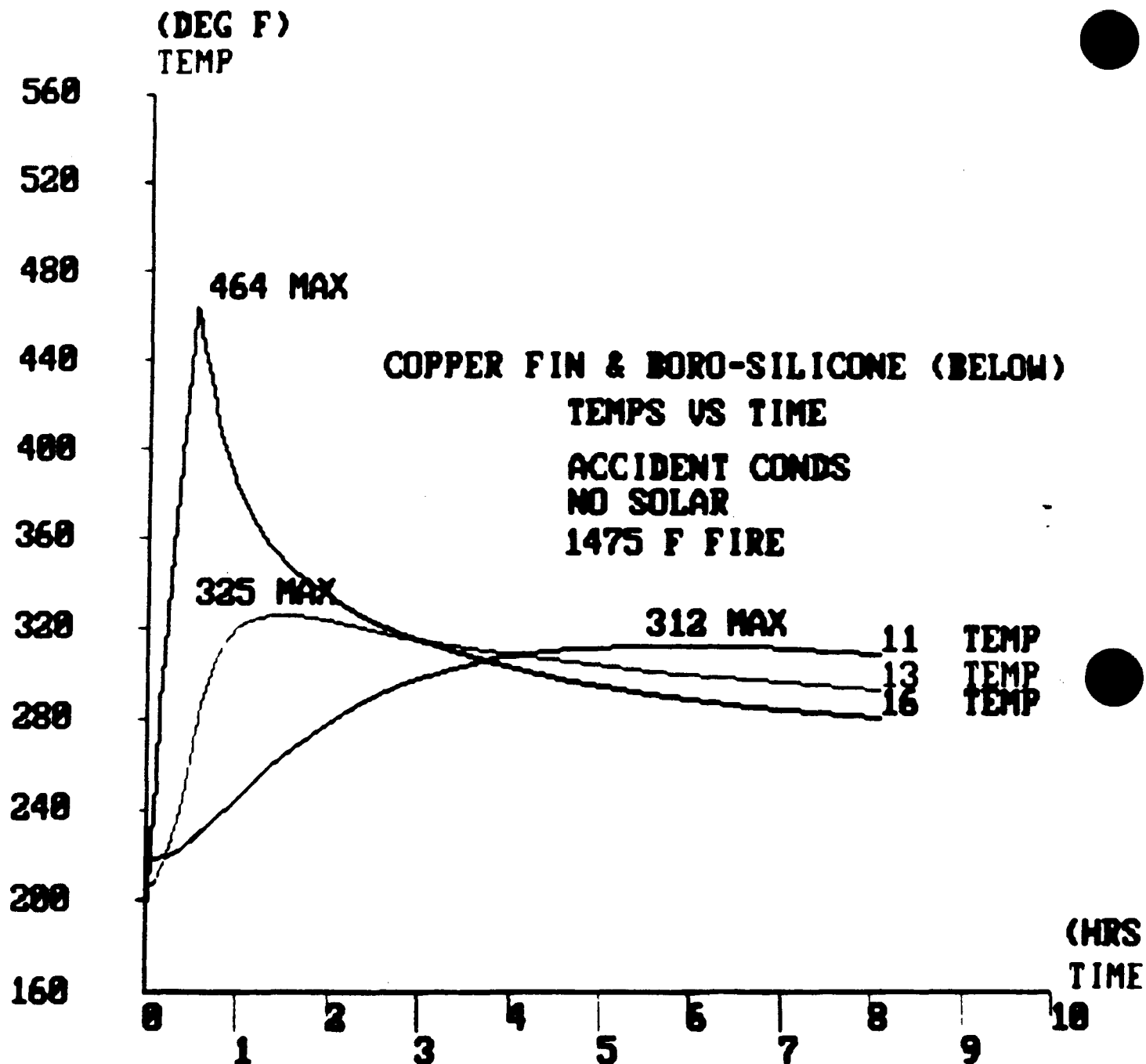


FIGURE 3.5.3-9
Neutron Shield & Copper Fin Temperature Transients for
Accident Conditions

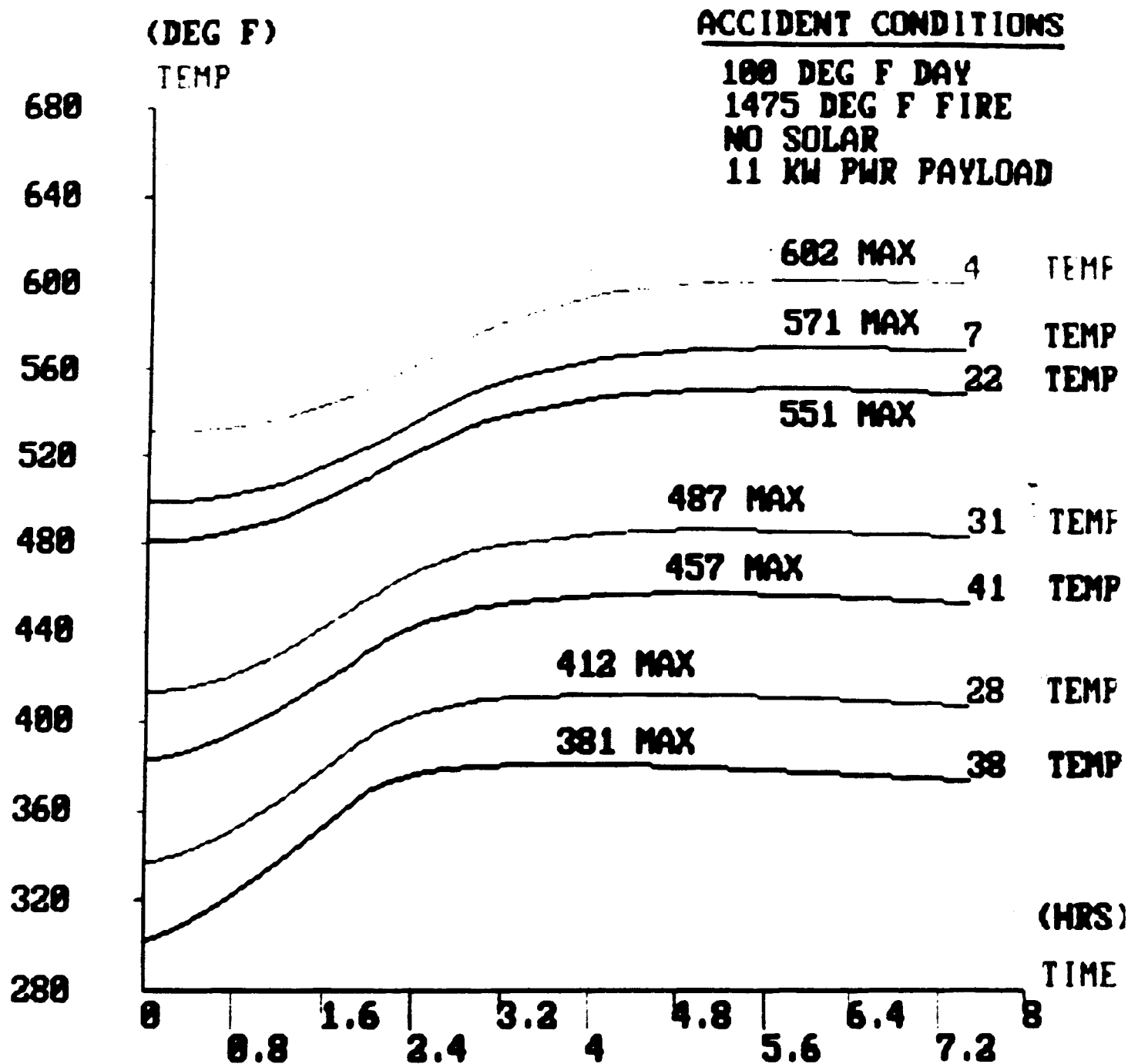


FIGURE 3.5.3-10
PWR Basket Temperature Transients for Accident Conditions

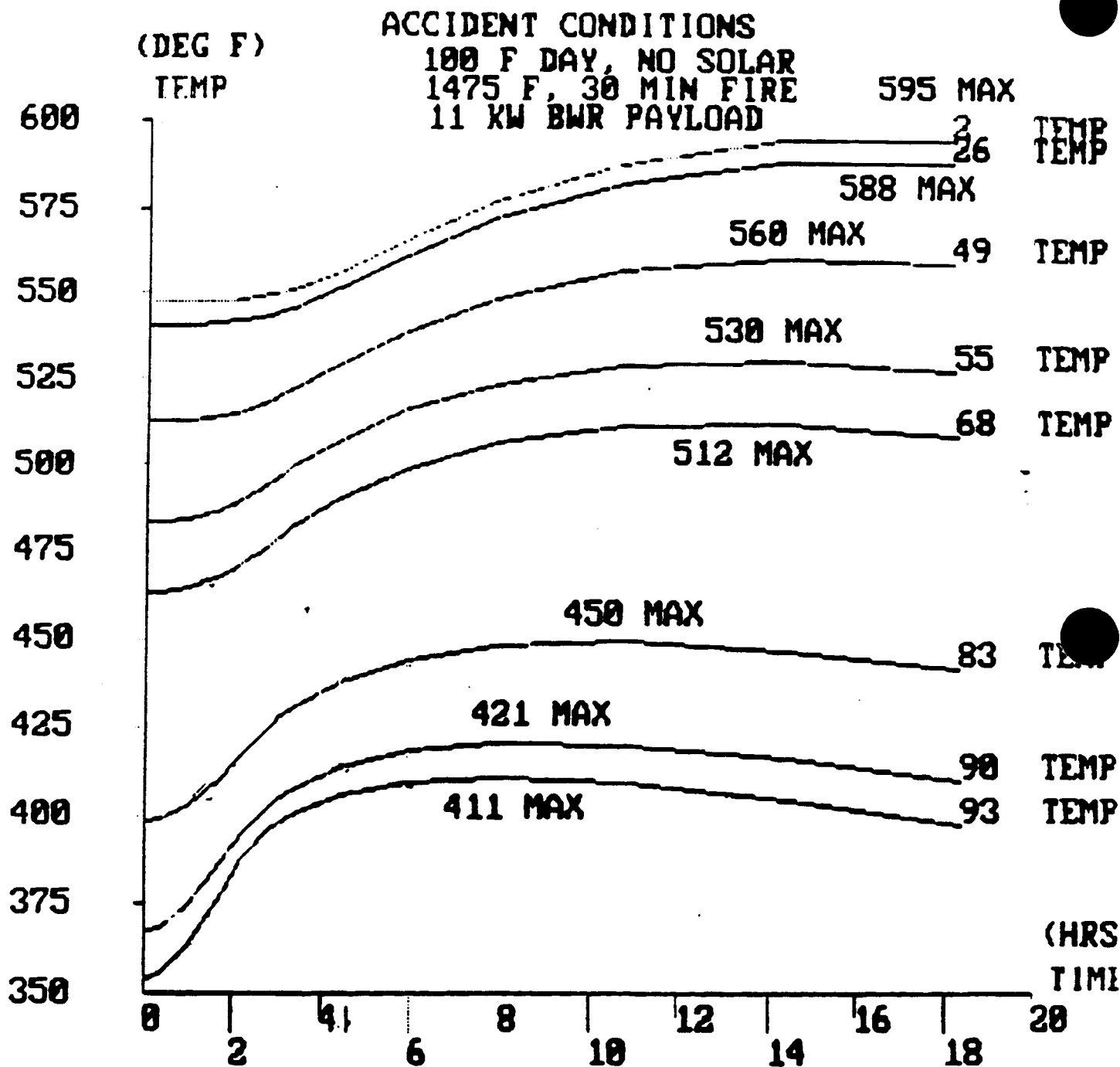


FIGURE 3.5.3-11
 BWR Basket Temperature Transients for Accident Conditions

FUEL ASSY TML MODEL ACCIDENT CONDITIONS

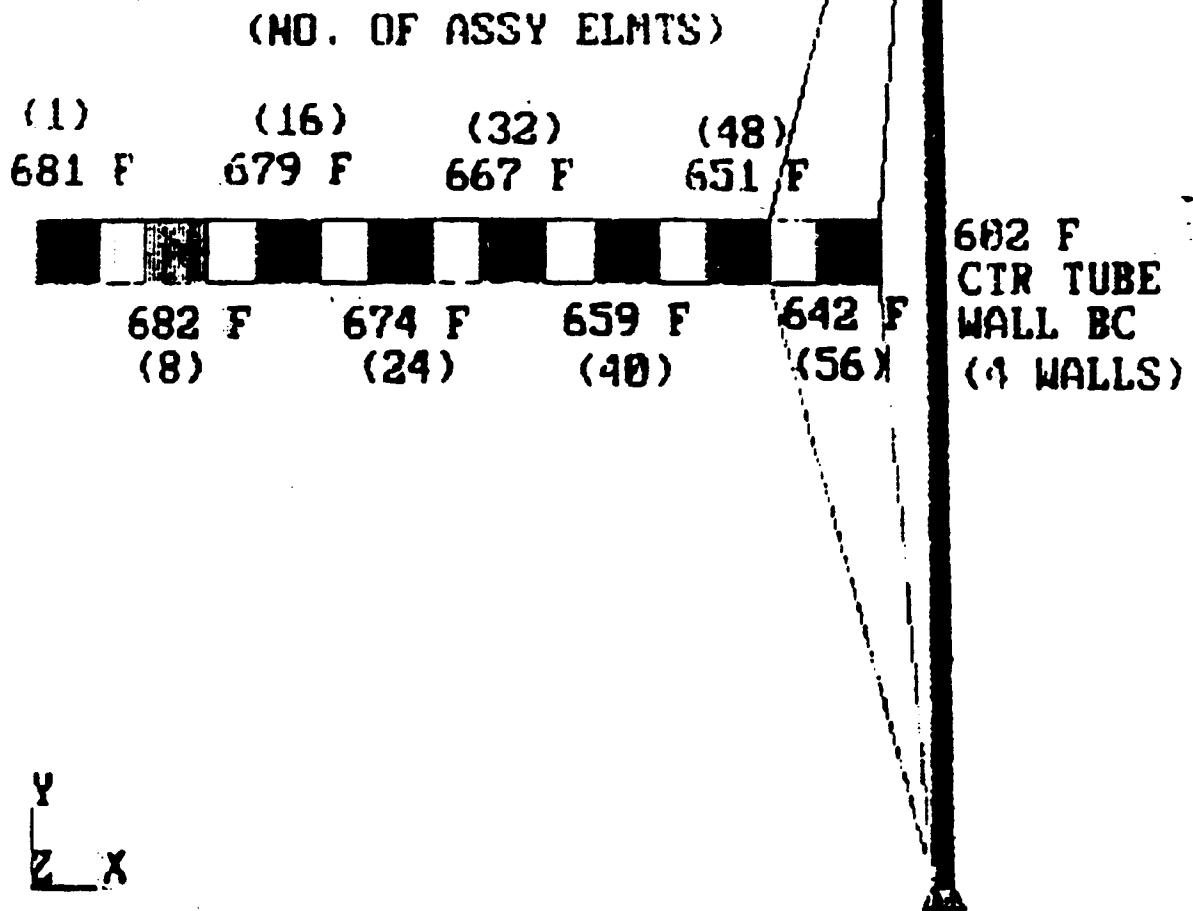


FIGURE 3.5.3-12

PWR Central Fuel Assembly Equilibrium Temperature Distribution
at Peak Basket Soakback for Transients for Accident Conditions

3.5.4 Maximum Internal Pressures

Maximum cask internal pressure for accident conditions is the sum of the fill gas pressure, residual fuel pool water that may remain after drain operations, and 100% fuel rod internal gas leakage. All pressures shall be calculated at peak soakback temperatures. The fill gas will be assumed to be nitrogen initially at 70°F and 14.7 psia. From Nuclear Packaging's experience with the IF-300 cask, 1.0 cubic feet of residual water will be assumed.

Assuming ideal gas mixtures, Dalton's law of partial pressures says the total pressure of the mixture of steam, nitrogen, and fuel rod gas leakage inside the cask is the sum of the individual gas pressures evaluated at the temperature and volume of the mixture. Thus, the temperature and volume of the mixture are calculated as follows:

- o Mixture volume = 135.5 cubic feet, which assumes a 50% fill factor approximation for the payload.
- o Mixture temperature = 552°F (1012 R), which is based on average internal surface temperatures of cask, basket and payload. The payload was assumed to be 21 PWR 17 X 17 fuel assemblies. The average surface temperatures of these three internal components were obtained from Figures 3.5.3-5, 3.5.3-10, & 3.5.3-12. Table 3.5.4-1 below summarizes the calculations for mixture temperature.

TABLE 3.5.4-1
Cask Internal Gas Mixture Temperature Calculation
(Accident Conditions)

	Internal Component	Area(sq-ft)	Temp(Deg F)	Product
o	Cask Walls	263	326	8,600
o	Basket Walls	1,139	492	560,000
o	Fuel Rod Walls	6,514	572	3,720,000
	SUMS	7,916	N/A	4,372,000

$$\text{MIXTURE TEMP} = 4,372,000/7916 = 552^{\circ}\text{F}$$

The steam, nitrogen, and fuel rod leakage gas pressures are calculated from ideal gas laws as follows:

$$P_{\text{steam}} = (62.4 \text{ lbm}) (85.8 \text{ ft-lbf/lbm-R}) (1012 \text{ R}) / [(144 \text{ sqin/sqft}) (135.5 \text{ cuft})] = 277.7 \text{ psia}$$

$$P_{\text{nitrogen}} = (14.7 \text{ psia}) (1012 \text{ R}) / (530 \text{ R}) = 28.0 \text{ psia}$$

$$P_{\text{fuel rod leakage}} = (28.5 \text{ lbm}) (40.84 \text{ ft-lbf/lbf-R}) (1012 \text{ R}) / [(144 \text{ sqin/sqft}) (135.5 \text{ cuft})] = 60.4 \text{ psia}$$

where fuel rod gas pressure design data were obtained from References (18) and (19) and are summarized in Table 3.5.4-2.

TABLE 3.5.4-2
Design Parameters for PWR Fuel Rods

- o 17 X 17 type assembly with 264 active fuel rods (21 fuel rod assemblies)
- o rod initial fill gas pressure = 565 psia
- o rod initial fill gas temperature = 530 R
- o rod fill gas constant = 40.84 ft-lb/lb-R
- o rod fill gas volume = 2.36 cu-in
- o rod avg fill gas temperature in cask = 1032 R
- o rod avg fill gas pressure in cask = $565(1032/530)$
= 1100 psia
- o rod avg fill gas weight in cask
 $w = \frac{(1100 \text{ psia})(13084 \text{ cu-in})}{(40.84 \text{ ft-lb/lb-R})(1032 \text{ R})(12 \text{ in/ft})} = 28.5 \text{ lbm}$

Applying Dalton's law of partial pressures, the gas mixture pressure inside of the cask is:

$$P_{\text{internal}} = P_{\text{steam}} + P_{\text{nitrogen}} + P_{\text{fuel rod leakage}} = 366 \text{ psia}$$

3.5.5 Maximum Thermal Stresses

Thermal stresses associated with the hypothetical thermal test condition can be classified as secondary, displacement limited stresses. The magnitude of these thermal stresses will be determined below and mentioned in Section 2.7.3.

As these thermal stresses are classified as secondary, they are only important regarding satisfaction of the requirement which limits the extreme total stress intensity range as given in paragraph C.7 of Regulatory Guide 7.6. The extreme total stress intensity range (including stress concentrations) between the initial state, the fabrication state, the normal operating conditions, and the accident conditions should be less than twice the adjusted value (adjusted to account for modulus of elasticity at the highest temperature) of S_e at 10 cycles given by the appropriate design fatigue curves. Evaluation of the stresses at the cask shell to end closure interface is addressed below. Evaluation per the criteria in paragraph C.7 of Regulatory Guide 7.6 is discussed in Section 2.7.3.

Thermal stresses for accident conditions will also be evaluated for cask lid bolts and the copper fins, primarily to verify that their heat dissipation function will not be upset. Finally, differential thermal expansion of the borated-silicone neutron shield will be mentioned below. Differential thermal expansion of the cask and basket under accident conditions will be quantified during the final design phase of the program. Cask pressure stresses due to thermally induced internal pressures will be considered in Section 2.7.3.

3.5.5.1 Cask Shell and Load Collar to End Closures

Transient analyses for the hypothetical thermal test condition show that peak thermal stresses occur in the load collar 12 minutes after the 30 minute, 1,475°F transient and in the two cask inner and outer shells 9.5 hours afterwards. Shell and forging temperatures are listed in Table 3.5.5-1 below, and are compared with the normal condition values listed in Figure 3.4.5-1.

TABLE 3.5.5-1
Cask Forging and Shell Temperatures for Thermal Stress Analyses

	NORMAL CONDITIONS (°F)		ACCIDENT CONDITIONS (°F)	
	<u>(Shell)</u>	<u>(Forging)</u>	<u>(Shell)</u>	<u>(Forging)</u>
o Load Collar	276	288	378	243
o Outer Shell	296	291	269	262
o Inner Shell	297	294	271	265

Corresponding thermal stresses are calculated for accident conditions utilizing the methods of Section 3.4.5.1 and the thermal gradients shown in Table 3.5.5-1 above. The results of the analysis are presented below in Table 3.5.5-2, which may be compared with normal condition calculations in Table 3.4.5-2.

TABLE 3.5.5-2
Thermal Stress Summary for Cask Shells and Load Collar

Cask Component	Wall Thickness (in.)	MEMBRANE STRESS (psi)		BENDING STRESS (psi)	
		Meridional	Hoop	Meridional	Hoop
o Inner Shell	1.13	-468	-1,408	-2,890	-866
o Outer Shell	1.50	468	-1,629	-2,953	-886
o Load Collar	0.25	-25,166	-31,716	-57,456	-17,237

3.5.5.2 Copper Fin to Outer Cask Shell Interface

Figures 3.5.3-8 and 3.5.3-9 show the maximum outer shell (node 13), and maximum fin (node 19) temperatures at the bolted joints to be 325°F and 326°F respectively; and to occur at identical times during the 30 minute, 1,475 F transient. Ignoring the 1°F difference, the thermal stress imposed on the half inch copper fins at the "stud welded bolt" attachment point may be conservatively calculated by increasing the normal condition value of 2362 psi, calculated in Section 3.4.5.2, by the ratio of the corresponding temperature differences; namely:

$$\sigma = (2,362) (326-70^{\circ}\text{F}) / [(300-70^{\circ}\text{F})] = 2,629 \text{ psi}$$

Figure 3.5.3-8 shows the maximum temperature difference in the fins to occur when (node 20) the sections adjacent to the thermal shield peak during the hypothetical thermal test transient at 475°F. At this time the fin roots (node 19) are shown to be at 280°F. Again, the thermal stress imposed on the fin may be conservatively calculated by increasing the normal condition values of 2216 psi compression and 621 psi bending, calculated in Section 3.4.5.2, by the ratio of the corresponding temperature differences; namely:

$$\sigma = (2,216) (378 - 70^{\circ}\text{F}) / [(300 - 70^{\circ}\text{F})] = 2,968 \text{ psi}$$

$$\sigma = (621) (378 - 70^{\circ}\text{F}) / [(300 - 70^{\circ}\text{F})] = 832 \text{ psi}$$

where the fin temperature of 378°F in the above calculations represents an average value along the fin of the two extremes, 475 and 280°F.

In sum, the copper fins will stay within their elastic limit for accident conditions because Reference (8) gives an allowable stress of 5000 psi, which is 33% lower than actual yield stress. Even though the allowable stress of 5000 psi of Reference (8) applies at 300°F. The fin bolted junction temperature above, at 326°F is not significantly above, 300°F. Thus, for accident conditions the heat dissipation function of the copper fins will not be degraded due to loosening at the "stud welded bolt" attachment point from one hypothetical thermal test cycle.

3.5.5.3 Cask Lid Bolted Interface

As found for normal conditions, transient thermal analysis also shows that the bolt and cask at the interface are at the same temperature, which is 265°F at peak soakback just beyond 20 hours after the 30 minute, 1475 F hypothetical thermal test transient. The relative longitudinal deflection between the cask lid flange and bolts imposes additional tension in the bolts and the resulting stress is calculated below.

$$\begin{aligned}\sigma &= (\alpha_{\text{cask}} - \alpha_{\text{bolt}}) (T_{\text{intf}} - T_{\text{ref}}) (E_{\text{bolt @265 F}}) \\ &= (8.59 (10)^{-6} - 6.69 (10)^{-6} \text{ in/in-F}) (265^{\circ} \text{F} - 70^{\circ} \text{F}) (26.8 (10)^6) \\ &= 9,929 \text{ psi}\end{aligned}$$

3.5.5.4 Relative Expansion of the Neutron Shield

Figure 3.5.3-8 shows the maximum temperature difference in the neutron shield borated-silicone segments to occur when (node 18) the sections adjacent to the thermal shield peak during the hypothetical thermal test transient at 553 °F. At this time the other side of the segments (node 19) are shown to be at 280°F. Thus, the borated-silicone segments adjacent to the thermal shield would have an average temperature of nearly 440°F for a short period, whereas design tolerances calculated in Section 3.4.5.3 were based on 300°F. The resulting increase in segment thermal expansion could be as high as 61% as calculated below:

$$(440 - 70 \text{ F}) / (300 - 70^{\circ} \text{F}) = 1.61$$

Borated-silicone neutron shield segments and end discs located elsewhere in the cask will not see the large thermal gradients above and will not be significantly affected. The above situation will be studied via analysis and/or test during final design. At this time it is believed that fire accident condition will not cause any problems because of the low modulus, viscoelastic and resilient nature of the borated-silicone; and the short period involved (see Figures 3.5.3-8 and 3.5.3-9). Thus, the tolerances calculated in Section 3.4.5.3 will be recommended for preliminary design.

3.5.6 Evaluation of Package Performance for the Hypothetical Accident Thermal Conditions

None of the temperatures noted from the transient analyses exceed temperature limitations of the respective materials as defined in Table 3.1-2 and Section 3.3. The highest lead temperature at 326 °F is well below the melting point of 620 °F. The highest O-ring seal temperature at 272 °F is well below the 350 °F compression set limit, and comfortably below the 300 °F limit determined by leak testing. Further leak testing may show that the 300 °F limit could be raised. The customer specification that the fuel cladding temperature not exceed 716 °F has been verified by conservative analysis for both PWR and BWR payloads. The maximum temperature calculated was 682 °F. Comparisons of 140-B thermal model analytical results with very similar Westinghouse MC-10 cask thermal test results, taken from Reference (13), show that the 682 °F peak cladding temperature could be 43 °F higher or 39 °F lower. Considering the conservative assumptions that went into the calculation of the 682 °F, the bias will be on the lower side. The details of the comparison are presented in Appendix 3.6.7.

Damaged conditions still have to be assessed by analyses, however, based upon the findings for the similar cask discussed in Reference (5) it is felt that such damage conditions will not be found to adversely affect the component temperature limits of concern, nor the shielding and containment aspects of the NuPac 140-B Cask design.

Regarding thermal expansion and stress, the heat dissipation function of the copper fins will not be degraded due to loosening at their 'stud welded bolt' attachment points due to thermal cycling, and thermal stress in the cask lid bolts is not excessive.

Borated-silicone neutron shield segments will be toleranced for proper fit. Additional expansion of the neutron shield segments adjacent to the thermal shield will be investigated during final design.

As input to Section 2.0, a 366 psia internal pressure is considered maximum for any cask component.

APPENDIX 3.6

- 3.6.1 Contract Requirements Regarding Thermal Design Aspects of NuPac 140-B Cask and Fuel Package per Contract Statement of Work, Reference (1)
- 3.6.2 Axisymmetric Cask Thermal Model Input, Element and Nodal Definitions, And The Normal Condition Nodal Temperatures
- 3.6.3 Thirty Degree 2-D Model of Cask Wall Input
- 3.6.4 Quarter 2-D Model of PWR Basket Input
- 3.6.5 Quarter 2-D Model of BWR Basket Input
- 3.6.6 Full 2-D Model of Center PWR Fuel Assembly Input
- 3.6.7 Accident Condition Equilibrium Prior To 30 minute, 1475°F Hypothetical Thermal Test Condition. Nodal Temperatures From Axisymmetric Cask Thermal Model
- 3.6.8 Comparisons of NuPac 140-B Cask Analytical Thermal Model Results With The Westinghouse MC-10 Storage Cask Test Data Given In Reference (13)
- 3.6.9 ANSYS-PC/THERMAL Element Descriptions Utilized for NuPac 140-B Cask Thermal Models

APPENDIX 3.6.1

Contract Requirements Regarding Thermal Design
Aspects of NuPac 140-B Cask and Fuel Package per
Contract Statement of Work, Reference (1).

The contractual requirements driving the thermal design aspects of the NuPac 140-B Cask and Fuel Package are taken from the contract statement of work and summarized in Table 3.6.1-1 below.

TABLE 3.6.1-1
CONTRACT REQUIREMENTS PER STATEMENT OF WORK

- O PAYLOAD HEAT DISSIPATION VIA PASSIVE HEAT TRANSFER TECHNIQUES
- O FUEL CLADDING SHALL NOT EXCEED 380 Deg C (716 Deg F)
- O INERT COVER GAS (Analysis Assumes Nitrogen)
- O DIFFERENTIAL THERMAL EXPANSION BASKET TO CASK BASIS 380 Deg C
FUEL CLADDING TEMPERATURE
- O PAYLOAD CHARACTERISTICS:
 - o 21 PWR ASSYS, 35000 MWD/MTU BURNUP, 10 YR DECAY
 - o 48 BWR ASSYS, 30000 MWD/MTU BURNUP, 10 YR DECAY
(Analysis Assumes 52 BWR Assys @ 35000 MWD/MTU)
- O CERT OF COMPLIANCE FROM NRC, THUS THERMAL REQUIREMENTS OF 10
CFR 71 APPLY

APPENDIX 3.6.2

Axisymmetric Cask Thermal Model Input, Element and
Nodal Definitions, And The Normal Condition Nodal
Temperatures

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140BTMLS\STEADYPO
30 Aug 1989

/TITLE.140-B THERMAL ANALYSIS FOR NORMAL CONDITIONS

C*** DEFINE ELEMENT TYPES

ET.1.55...1	*2-D AXISYMMETRIC ISOPAR THM SLD
ET.2.32	*2-D AXISYMMETRIC COND LINK
ET.3.31	*RADIATION LINK
ET.4.34	*CONVECTION LINK

C*** MATERIAL PROPERTIES

KXX.1..650.3.94E-4	*XM-19 SS(B/H-IN-F)
KXX.2.1.708,-4.83E-4	*LEAD
KXX.3..0242	*BORO-SILICONE
KXX.4.18.33	*COPPER
KXX.6..778.2.52E-4	*FUEL ASSY SPACER/TIE PLATES(304 SS)
KXX.7..389	*DEPLETED URANIUM (EFFECTIVE, 1.167/3)
KXX.9.6.14	*BASKET COMPOSITE MTL
KXX.10.2.67	*ALUMINUM (1100 SERIES, EFFECTIVE 10.69/4)
KXX.12.2.40E-4.7.30E-6	*POLYURETHANE(15 PCF, 175 TO 275 DEGF)
KXX.13.1.08E-3.1.90E-6	*AIR OR NITROGEN
KXX.15.7.93E-4.2.71E-6	*ARGON

C.1..120	*B/LBM-F
C.2..030	
C.3..460	
C.4..092	
C.6..120	*FUEL ASSY SPACER/TIE PLATES(304 SS)
C.7..028	
C.9..112	*BASKET COMP MTL
C.10..230	
C.12..300	

DENS.1..290	*LBM/CU-IN
DENS.2..310	
DENS.3..057	
DENS.4..161	*HALF COPPER (.322) LONG - RADIAL FINS
DENS.6..097	*FUEL ASSY SPACER/TIE PLATES(.33X304 SS)
DENS.7..303	*FOR EQUIVALENT VOLUMES (.683 LB/CUIN PURE)
DENS.9..299	*BASKET COMP MTL
DENS.10..0245	*FOR EQUIVALENT VOLUMES (.098/4)
DENS.12..00868	*15 PCF FOAM

HF.5.1.86E-3.8.60E-6	*NAT CONVECTION TO LIMITER(B/SQIN-ER-DEGF)
HF.11.2.08E-3.8.60E-6	*NAT CONV EXT TO TML SHIELD
HF.14.6.944	*CONTACT RES COEF LNTR/CASK & LID/CASK BTE

C*** REAL CONSTANTS

R.1.17.06	*COPPER FIN CONDUCTION LINKS
R.2..312	
R.3.15.86	
R.4.1.82	
R.5.5.46	
R.6.5.20	
R.7.3.51	
R.8.3.76	
R.9.5.63	

FIGURE 3.6.2-1
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

R.10.664	*INNER THERMAL SHIELD CONDUCTION LINKS
R.11.12.2	
R.12.604	
R.13.273	
R.14.385	
R.15.258	
R.16.664..67.1.	*INNER THERMAL SHIELD RADIATION LINKS
R.17.12.2..67.1.	
R.18.604..67.1.	
R.19.273..67.1.	
R.20.385..67.1.	
R.21.258..67.1.	
R.22.673.1...25	*RADIATION LINKS EXT FROM TML SHIELD
R.23.12.3.1...25	
R.24.617.1...25	
R.25.69.7.1...25	
R.26.208..5..25	
R.27.208..5..25	
R.28.673	*CONVECTION LINKS EXT FROM TML SHIELD
R.29.12.3	
R.30.617	
R.31.69.7	
R.32.208	
R.33.157	*CONDUCTION BETWEEN CASK AND LIMITER
R.34.233	
R.35.329	
R.36.209	
R.37.97	
R.38.139	
R.39.292	
R.40.162	
R.41.81	
R.42.10.1	
R.43.97.2	*CONTACT RESISTANCE AT LIMITER ATMT PLT
R.44.97.2	
R.45.157..24.1.	*RADIATION BETWEEN CASK AND LIMITER
R.46.233..24.1.	
R.47.329..24.1.	
R.48.209..24.1.	
R.49.97..24.1.	
R.50.139..24.1.	
R.51.292..24.1.	
R.52.162..24.1.	
R.53.81..24.1.	
R.54.10.1..24.1.	
R.55.83.7	*COND LID/CASK @ O-RINGS
R.56.150	
R.57.83.7..33.1.	*RAD LID/CASK @ O-RINGS
R.58.150..33.1.	

FIGURE 3.6.2-2
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140BTMLS\STEADYPD
30 Aug 1989

R.59.107	*CONTACT RES & LID/CASK BOLTS
R.60.10.1	*CONVECTION TO IMPACT LMTR EXT
R.61.81	
R.62.162	
R.63.243	
R.64.200	
R.65.209	
R.66.329	
R.67.511	
R.68.544	
R.69.640	
R.70.768	
R.71.512	
R.72.256	
R.73.320	
R.74.384	
R.75.832	
R.76.576	
R.77.320	
R.78.448	
R.79.511	
R.80.329	
R.81.116	
R.82.10.1.1...8	*RADIATION TO IMPACT LMTR EXT
R.83.81.1...8	
R.84.162.1...8	
R.85.243.1...8	
R.86.200.1...8	
R.87.209.1...8	
R.88.329.1...8	
R.89.511.1...8	
R.90.544.1...8	
R.91.640.1...8	
R.92.768.1...8	
R.93.512.1...8	
R.94.256.1...8	
R.95.320.1...8	
R.96.384.1...8	
R.97.832.1...8	
R.98.576.1...8	
R.99.320.1...8	
R.100.448.1...8	
R.101.511.1...8	
R.102.329.1...8	
R.103.116.1...8	
R.104.1049	*PERS BAR/SUN SHLD CONVECTION
R.105.1049.1...9	*PERS BAR/SUN SHLD RADIATION
R.106.2.79..66.1.	*RAD FUEL ASSY TUBES TO TIE PLATES/SPCR
R.107.22.3..66.1.	
R.108.33.3..66.1.	
R.109.6.83..33.1.	*RAD/COND BASKET TO LID/CLOSURE
R.110.39.2..33.1.	
R.111.151..33.1.	

FIGURE 3.6.2-3
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140BTMLS\STEADYPD
30 Aug 1989

R,112,6.83
R,113,39.2
R,114,151

R,115,120,.28,1.
R,116,29.2,.28,1.
R,117,189,.28,1.
R,118,45.9,.28,1.
R,119,420,.28,1.
R,120,102,.28,1.
R,121,520,.28,1.
R,122,126,.28,1.
R,123,731,.28,1.
R,124,178,.28,1.

*RADIATION INTERNAL IN BASKET (RADIAL)

R,125,13.9
R,126,1.51
R,127,26.1
R,128,2.83

*CONDUCTION INTERNAL IN BASKET (RADIAL)

R,129,234,.33,1.
R,130,467,.33,1.
R,131,353,.33,1.
R,132,171,.33,1.
R,133,143,.33,1.
R,134,185,.33,1.
R,135,157,.33,1.

*RAD BSKT EXT/CASK (RADIAL)

R,136,8174
R,137,16312
R,138,12330
R,139,5973
R,140,4995
R,141,6462
R,142,5484

*GAS COND BSKT EXT/CASK (RADIAL)

R,143,14.38

*TML SHLD COLLAR TO FORGING WELD

R,144,2.26
R,145,6.76
R,146,11.3
R,147,150
R,148,17.8
R,149,19.7
R,150,19.7
R,151,19.7
R,152,10.6
R,153,12.3
R,154,14.8

*LIMITER SHUNT AREAS (.5 IN COPPER)

*.25 IN

C*** OPTIONS

KAN,-1
ITER,-20,20,20
KBC,1
TOFFST,460
TIME,0
KTEMP,-1

*THERMAL ANALYSES
*OPT CONVERGENCE PO @ LAST ITER NO LIST
*STP BCS FOR NEXT LOAD STEP
*RAD LINKS PRES, SIG DEG R
*STEADY STATE ANALYSIS(DFT)
*HEAT GENERATION LOADS PRESENT

FIGURE 3.6.2-4
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:

LS\STEADYPO
30 Aug 1989

C*** NODE DESCRIPTIONS

N,1,100,100	*RADIATION BC
N,2,100,50	*CONVECTION BC
N,3,58,70.5	*PERSONNEL BARRIER
N,4,59,70.5	
N,5,59,141.75	
N,6,58,141.75	
N,7,28.5,141.75	*CONTAINMENT/SHIELD CYL WALLS
N,8,29.625,141.75	
N,9,33.125,141.75	
N,10,34.625,141.75	
N,11,40.5,141.75	
N,12,40.625,141.75	
N,13,40.75,141.75	
NGEN,4,7,7,13,1.,-16.4	
N,35,28.5,92.25	
N,36,29.625,92.25	
N,37,33.125,92.25	
N,38,34.625,92.25	
N,39,39.75,92.25	
N,40,39.875,92.25	
N,41,41.01,92.25	
N,42,28.5,77	
N,43,29.625,77	
N,44,33.125,77	
N,45,34.625,77	
N,46,39.19,77	
N,47,39.34,77	
N,48,39.91,77	
N,49,28.5,75.25	
N,50,29.625,75.25	
N,51,33.125,75.25	
N,52,34.625,75.25	
N,53,39.08,75.25	
N,54,39.21,75.25	
N,55,39.80,75.25	
N,56,28.5,70	
N,57,29.625,70	
N,58,33.125,70	
N,59,34.625,70	
N,60,38.75,70	
N,61,39.00,70	
N,62,39.42,70	
N,63,28.5,66	
N,64,29.625,66	
N,65,33.125,66	
N,66,34.625,66	
N,67,38.53,66	
N,68,38.65,66	
N,69,39.14,66	
N,70,28.5,60	
N,71,29.625,60	
N,72,33.125,60	
N,73,34.625,60	
N,74,38.20,60	
N,75,38.35,60	

FIGURE 3.6.2-5
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140BTHLS\STEADYPD
30 Aug 1989

N,76,38.70.60
NGEN,2,7,70,76,1,-6.75

N,84,28.5,51.5
N,85,29.625,51.5
N,86,33.125,51.5
N,87,34.625,51.5
N,88,38.70,51.5
NGEN,2,16,84,88,1,-1.43
NGEN,2,4,102,104,1,-1.47
N,105,29.9,48.6
NGEN,3,4,105,108,1,-2.5
N,117,31.6,43.6
NGEN,2,4,114,116,1,-3.6

*END FORGING

N,132,33.27,40
N,133,33.27,43.6
N,134,31.6,40
N,135,31.6,43.6
N,136,29.888,40
N,137,29.888,43.6
N,138,29.888,46.1
N,139,29.888,48.6
N,140,28.44,50.07
N,141,28.44,51.5
N,142,27,40
N,143,27,43.6
N,144,27,46.1
N,145,27,48.6
N,146,27,50.07
N,147,27,51.5
NGEN,4,6,142,147,1,-9

*CASK LID

N,200,39.25
N,201,36
N,202,32
N,203,20
N,204,8
N,205,1
N,206,0
NGEN,4,7,200,206,1,9
NGEN,2,7,221,227,1,6.3
NGEN,2,7,228,234,1,5.68
NGEN,2,7,235,241,1,5.09
NGEN,2,7,242,248,1,10
NGEN,2,7,249,254,1,9
NGEN,2,7,256,260,1,1
N,268,64
N,269,38.95,46.1
N,270,44.0,46.1
N,271,54.0,46.1
N,272,63.0,46.1
N,273,64.0,46.1
NGEN,2,5,269,273,1,5.4
NGEN,2,5,274,278,1,8.5
N,284,39.4,66
NGEN,2,5,280,283,1,6.0
N,289,39.6,69

*LIMITER

FIGURE 3.6.2-6

AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CIRC\1408TMLS\STEADYPD
30 Aug 1989

NGEN,2.5,285,287,1.,3.0
N,288,64.66
N,293,39.7,70
N,294,44.70
N,295,54.70
N,296,64.70

N,300,0,141.75
N,301,3.34,141.75
N,302,4.923,141.75
N,303,5.257,141.75
N,304,6.846,141.75
N,305,11.655,141.75
N,306,13.308,141.75
N,307,14.442,141.75
N,308,16.696,141.75
N,309,20.304,141.75
N,310,22.615,141.75
N,311,23.635,141.75
NGEN,2,12,300,311,1.,-71.75
N,324,0.69
N,325,3.34,69
N,326,6.846,69
N,327,11.655,69
N,328,16.696,69
N,329,20.304,69
NGEN,2,6,324,329,1.,-16.5
N,336,4.923,52.5
N,337,5.257,52.5
N,338,6.846,52.5
N,339,14.442,52.5
N,340,22.615,52.5
N,341,23.635,52.5

*PAYLOAD BASKET

C*** ELEMENT DESCRIPTIONS

TYPE,1
MAT,10
E,3,4,5,6

*ALUMINUM PERSONNEL BARRIER

TYPE,1
MAT,1
E,14,15,8,7
RP10,7,7,7,7
E,16,17,10,9
RP10,7,7,7,7
E,19,20,13,12
RP10,7,7,7,7

*304 SS CONTAINMENT WALLS & TML SHLD
*INCLUDING LOAD COLLAR

MAT,2
E,15,16,9,8
RP10,7,7,7,7

*LEAD GAMMA SHIELD

MAT,3
E,17,18,11,10
RP10,7,7,7,7

*BORO-SILICONE NEUTRON SHIELD

TYPE,2
MAT,4

*CONDUCTING PINS (COPPER)

FIGURE 3.6.2-7
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140BTMLS\STEADYPD
30 Aug 1989

REAL.1
E.10.11
E.17.18
E.24.25
REAL.2
E.31.32
REAL.3
E.38.39
REAL.4
E.45.46
REAL.5
E.52.53
REAL.6
E.59.60
E.66.67
REAL.7
E.73.74
E.80.81
REAL.8
E.80.73
E.81.74
E.73.66
E.74.67
E.66.59
E.67.60
E.60.53
E.59.52
E.53.46
E.52.45
E.46.39
E.45.38
REAL.9
E.39.32
E.38.31
E.32.25
E.31.24
E.25.18
E.24.17
E.18.11
E.17.10

*LONGITUDINAL CONDUCTION

REAL.10
MAT.13
E.11.12
E.18.19
E.25.26
REAL.11
E.32.33
REAL.12
E.39.40
REAL.13
E.53.54
REAL.14
E.67.68
REAL.15
E.74.75

*COND THRU AIR BETWEEN SHIELDS

TYPE.3
REAL.16

*RADIATION BETWEEN SHIELDS

FIGURE 3.6.2-8
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

E.11.12
E.18.19
E.25.26
REAL.17
E.32.33
REAL.18
E.39.40
REAL.19
E.53.54
REAL.20
E.67.68
REAL.21
E.74.75

REAL.22
E.13.6
E.20.6
E.27.6
REAL.23
E.34.3
REAL.24
E.41.3
REAL.25
E.48.3
REAL.26
E.55.3
REAL.27
E.55.293

*RADIATION EXT FROM TML SHIELD

TYPE.4
MAT.11
REAL.28
E.13.2
E.20.2
E.27.2
REAL.29
E.34.2
REAL.30
E.41.2
REAL.31
E.48.2
REAL.32
E.55.2

*CONVECTION EXT FROM TML SHIELD

TYPE.2
MAT.13
REAL.33
E.62.289
REAL.34
E.69.284
REAL.35
E.76.279

*CONDUCTION BETWEEN CASK AND LIMITER

FIGURE 3.6.2-9
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\1408THLS\STEADYPD
30 Aug 1989

REAL.36
E.88.274
REAL.37
E.112.269
REAL.38
E.120.235
REAL.39
E.142.221
REAL.40
E.148.214
REAL.41
E.154.207
REAL.42
E.160.200

TYPE.4
MAT.14
REAL.43
E.119.235
E.119.228

*CONTACT RESISTANCE CASK AND LIMITER BOLTS

TYPE.3
REAL.45
E.62.289
REAL.46
E.69.284
REAL.47
E.76.279
REAL.48
E.88.274
REAL.49
E.112.269
REAL.50
E.120.235
REAL.51
E.142.221
REAL.52
E.148.214
REAL.53
E.154.207
REAL.54
E.160.200

*RADIATION BETWEEN CASK AND LIMITER

TYPE.2
MAT.13
REAL.55
E.140.100
REAL.56
E.138.109

*AIR COND SET LID & CASK O-RINGS

TYPE.3
REAL.57
E.140.100
REAL.58
E.138.109

*RAD SET LID & CASK O-RINGS

TYPE.4
MAT.14
REAL.59

*CONTACT RES SET LID & CASK BOLTS

FIGURE 3.6.2-10
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140STMLS\STEADYPD
30 Aug 1989

E.135.117

TYPE.1

*END FORGING

MAT.1

E.84.85.78.77

RP4.1.1.1.1

E.100.101.85.84

RP4.1.1.1.1

E.105.101.100.100

E.105.106.102.101

RP3.1.1.1.1

E.109.110.106.105

RP3.1.1.1.1

E.113.117.109.109

E.117.110.109.109

E.117.114.110.110

E.114.115.111.110

RP2.1.1.1.1

E.118.119.115.114

RP2.1.1.1.1

E.160.154.155.161

*CASK LID

RP5.1.1.1.1

E.154.148.149.155

RP5.1.1.1.1

E.148.142.143.149

RP5.1.1.1.1

E.142.136.137.143

RP5.1.1.1.1

E.136.134.135.137

E.134.132.133.135

MAT.12

*LIMITER

E.201.208.207.200

RP6.1.1.1.1

E.208.215.214.207

RP6.1.1.1.1

E.215.222.221.214

RP6.1.1.1.1

E.222.229.228.221

RP6.1.1.1.1

E.229.236.235.228

RP6.1.1.1.1

E.236.243.242.239

RP6.1.1.1.1

E.243.250.249.242

RP6.1.1.1.1

E.250.257.256.249

RP5.1.1.1.1

E.255.268.261.254

E.257.264.263.256

RP5.1.1.1.1

E.235.242.270.269

RP4.7.7.1.1

E.269.270.275.274

RP4.1.1.1.1

E.274.275.280.279

RP4.1.1.1.1

FIGURE 3.6.2-11

Axisymmetric Thermal Model ANSYS Input

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140BTMLS\STEADYPD
30 Aug 1989

E.279.280.285.284
RP4.1.1.1.1
E.284.285.290.289
RP3.1.1.1.1
E.287.288.296.292
E.289.290.294.293
RP3.1.1.1.1

TYPE.4
MAT.5

*CONVECTION EXT TO LIMITER

REAL.60
E.206.2
REAL.61
E.213.2
REAL.62
E.220.2
REAL.63
E.227.2
REAL.64
E.234.2
REAL.65
E.241.2
REAL.66
E.248.2
REAL.67
E.255.2
REAL.68
E.268.2
REAL.69
E.267.2
REAL.70
E.266.2
REAL.71
E.265.2
REAL.72
E.264.2
REAL.73
E.263.2
REAL.74
E.273.2
REAL.75
E.278.2
REAL.76
E.283.2
REAL.77
E.288.2
REAL.78
E.296.2
REAL.79
E.295.2
REAL.80
E.294.2
REAL.81
E.293.2

TYPE.3
REAL.82
E.206.1
REAL.83

*RADIATION EXT TO LIMITER

FIGURE 3.6.2-12
Axisymmetric Thermal Model ANSYS Input

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140BTMLS\STEADYPPD
30 Aug 1989

E.213.1
REAL.84
E.220.1
REAL.85
E.227.1
REAL.86
E.234.1
REAL.87
E.241.1
REAL.88
E.248.1
REAL.89
E.255.1
REAL.90
E.268.1
REAL.91
E.267.1
REAL.92
E.266.1
REAL.93
E.265.1
REAL.94
E.264.1
REAL.95
E.263.1
REAL.96
E.273.1
REAL.97
E.278.1
REAL.98
E.283.1
REAL.99
E.288.1
REAL.100
E.296.1
REAL.101
E.295.3
REAL.102
E.294.3
REAL.103
E.293.3

TYPE.2
MAT.4
REAL.144
E.200.207
REAL.145
E.207.214
REAL.146
E.214.221
REAL.147
E.221.228
REAL.148
E.228.235
REAL.149
E.235.269
E.269.274
E.274.279
REAL.150

*LIMITER SHUNT CONDUCTION LINKS

FIGURE 3.6.2-13
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140BTMLS\STEADYPD
30 Aug 1989

E.279.284
REAL.151
E.284.289
E.289.293
REAL.152
E.293.294
REAL.153
E.294.295
REAL.154
E.295.296

MAT.10
TYPE.1
E.3.4.5.6

*PERSONNEL BARRIER/SUN SHIELD

TYPE.4
MAT.5
REAL.104
E.3.2
E.4.2
E.5.2
E.6.2

*PERS BAR/SUN SHLD CONVECTION

TYPE.3
REAL.105
E.4.1
E.5.1

*PERS BAR/SUN SHLD RADIATION

MAT.7
TYPE.1
E.312.313.301.300
RP3.4.4.4.4

*FUEL ROD ASSEMBLIES

MAT.6
E.330.331.325.324
RP3.2.2.2.2

*FUEL ASSY SPACERS/TIE PLATES

MAT.9
E.314.315.303.302
RP3.4.4.4.4
E.336.337.315.314
RP3.2.2.4.4

*BASKET WALLS BETWEEN FUEL RODS

TYPE.3
REAL.106
E.312.324
E.313.325
REAL.107
E.316.326
E.317.327
REAL.108
E.320.328
E.321.329

*RAD FUEL ASSY TO SPCR/TIE PLTS

REAL.109
E.330.165
E.331.165
REAL.110
E.332.159

*RAD END OF BASKET TO LID/CLOSURE

FIGURE 3.6.2-14
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\1408TMLS\STEADYPD
30 Aug 1989

E.333.159
REAL.111
E.334.153
E.335.147

TYPE.2
REAL.112
E.330.165
E.331.165
REAL.113
E.332.159
E.333.159
REAL.114
E.334.153
E.335.147

*COND END OF BASKET TO LID/CLOSURE

TYPE.3
REAL.115
E.301.302
E.313.314
REAL.116
E.325.314
E.331.336
REAL.117
E.303.304
E.315.316
REAL.118
E.315.326
E.337.332
REAL.119
E.305.306
E.317.318
REAL.120
E.327.318
E.333.338
REAL.121
E.307.308
E.319.320
REAL.122
E.319.328
E.339.334
REAL.123
E.309.310
E.321.322
REAL.124
E.329.322
E.335.340

*RADIATION INTERNAL IN BASKET (RADIAL)

TYPE.2
MAT.9
REAL.125
E.303.306
E.315.318
REAL.126
E.337.338
REAL.127
E.307.310
E.319.322
REAL.128

*CONDUCTION INTERNAL IN BASKET (RADIAL)

FIGURE 3.6.2-15
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

Date: 31 Aug 1989
Time: 3:14p

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\140BTHLS\STEADYPD
30 Aug 1989

E.339.340

TYPE.3
REAL.129
E.311.7
REAL.130
E.311.14
E.311.21
E.323.35
REAL.131
E.323.42
REAL.132
E.323.56
REAL.133
E.323.63
REAL.134
E.341.70
REAL.135
E.341.77

*RADIATION EXTERNAL FROM BSKT (RADIAL)

TYPE.2
MAT.13
REAL.136
E.311.7
REAL.137
E.311.14
E.311.21
E.323.35
REAL.138
E.323.42
REAL.139
E.323.56
REAL.140
E.323.63
REAL.141
E.341.70
REAL.142
E.341.77

*CONDUCTION EXTERNAL FROM BSKT (RADIAL)

TYPE.1
MAT.3
E.80.81.74.73
RP10,-7,-7,-7,-7

*BORO-SILICONE NEUTRON SHIELD

TYPE.2
MAT.1
REAL.143
E.81.82

*THERM SHLD TO FORGING WELD

C*** SPECIFYING MODEL LOADS

NT.1.TEMP.100
NT.2.TEMP.100

*RADIATION HEAT SINK/SOURCE
*CONVECTION HEAT SINK/SOURCE

HFLOW.4.HEAT.448
HFLOW.5.HEAT.448
HFLOW.206.HEAT,.864
HFLOW.213.HEAT,6.92
HFLOW.220.HEAT,13.8

*SOLAR INSOLATION(B/HR-SM)

FIGURE 3.6.2-16
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

HFLOW,227,HEAT,20.8
HFLOW,234,HEAT,17.1
HFLOW,241,HEAT,17.8
HFLOW,248,HEAT,28
HFLOW,255,HEAT,43.6
HFLOW,268,HEAT,69.6
HFLOW,267,HEAT,109

HFLOW,266,HEAT,131
HFLOW,265,HEAT,87.6
HFLOW,264,HEAT,43.6
HFLOW,263,HEAT,54.8
HFLOW,273,HEAT,65.6
HFLOW,278,HEAT,142
HFLOW,283,HEAT,98.4
HFLOW,288,HEAT,54.8
HFLOW,296,HEAT,57.2

QE,326,.072
QE,327,.072
QE,328,.072

*FUEL ROD ASSY DECAY HEAT (B/HR-CUIN)

C***LOAD STEP 1 (NEEDED FOR STABILITY OF LOAD STEP 2)

PRDISP,-1
PRNF,-1
PRSTR,-1

LWRITE

C***LOAD STEP 2 (11KW)

/TITLE,DSGN WITH CU FINS,BORO-SIL NS, STL/CU CP BAS

PRDISP,20
PRNF,-1
PRSTR,-1

QE,326,.362
QE,327,.362
QE,328,.362

LWRITE

AFWRITE

FINISH

FIGURE 3.6.2-17
AXISYMMETRIC THERMAL MODEL ANSYS INPUT

HALF AXISYMMETRIC MODEL OF CASK WITH IMPACT LIMITER

(ELEMENT BREAKDOWN)

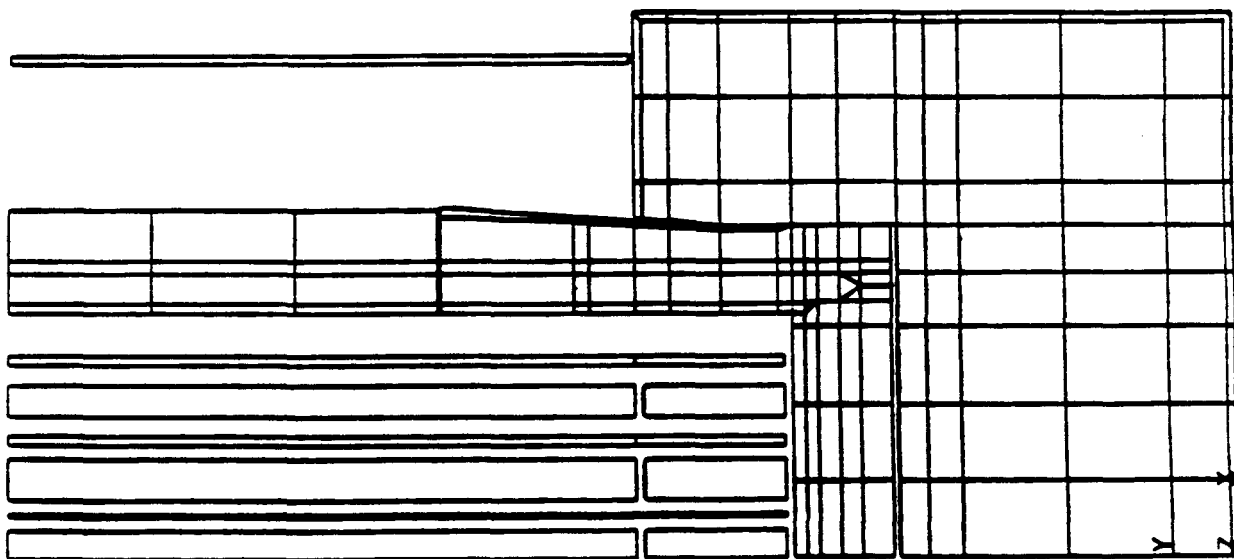


FIGURE 3.6.2-18
FINITE ELEMENT BREAKDOWN FOR AXISYMMETRIC THERMAL
MODEL OF CASK AND IMPACT LIMITER

HALF AXISYMMETRIC MODEL OF CASK WITH IMPACT LIMITER

(ELEMENT BREAKDOWN EXPLOADED VIEW)

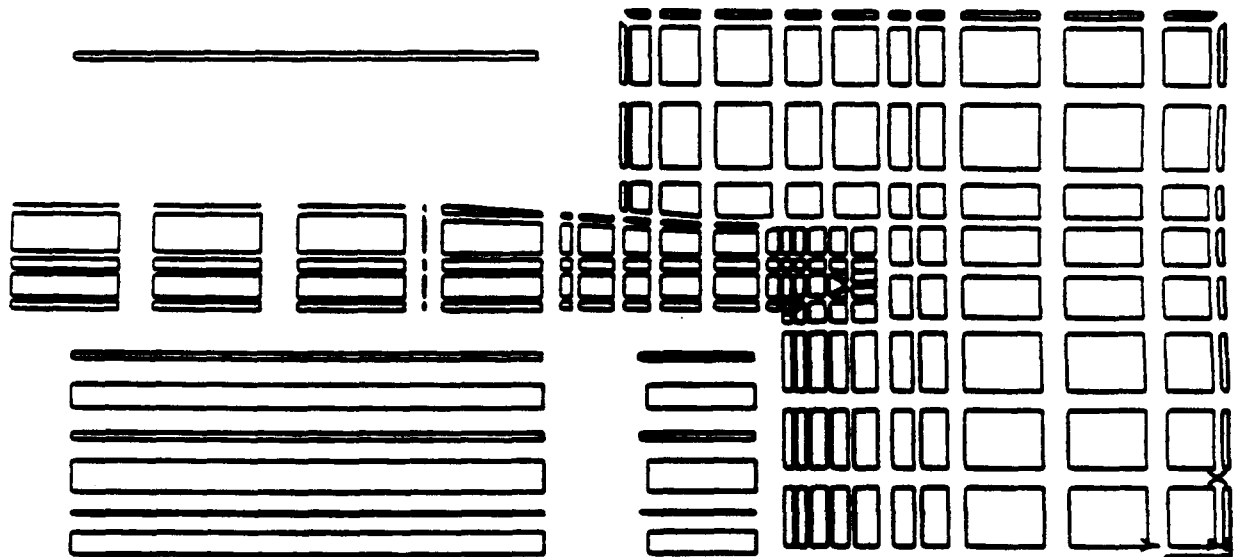


FIGURE 3.6.2-19
FINITE ELEMENT BREAKDOWN FOR AXISYMMETRIC THERMAL
MODEL OF CASK AND IMPACT LIMITER (EXPLODED VIEW)

312	313	314	315	316	317	318	319	320	321	322	323	324	325	326	327	328	329	330	331	332	333	334	335	336	337	338	339	340	341	342	343	344	345	346	347	348	349	350	351	352	353	354	355	356	357	358	359	360	361	362	363	364	365	366	367	368	369	370	371	372	373	374	375	376	377	378	379	380	381	382	383	384	385	386	387	388	389	390	391	392	393	394	395	396	397	398	399	400	401	402	403	404	405	406	407	408	409	410	411	412	413	414	415	416	417	418	419	420	421	422	423	424	425	426	427	428	429	430	431	432	433	434	435	436	437	438	439	440	441	442	443	444	445	446	447	448	449	450	451	452	453	454	455	456	457	458	459	460	461	462	463	464	465	466	467	468	469	470	471	472	473	474	475	476	477	478	479	480	481	482	483	484	485	486	487	488	489	490	491	492	493	494	495	496	497	498	499	500	501	502	503	504	505	506	507	508	509	510	511	512	513	514	515	516	517	518	519	520	521	522	523	524	525	526	527	528	529	530	531	532	533	534	535	536	537	538	539	540	541	542	543	544	545	546	547	548	549	550	551	552	553	554	555	556	557	558	559	560	561	562	563	564	565	566	567	568	569	570	571	572	573	574	575	576	577	578	579	580	581	582	583	584	585	586	587	588	589	590	591	592	593	594	595	596	597	598	599	600	601	602	603	604	605	606	607	608	609	610	611	612	613	614	615	616	617	618	619	620	621	622	623	624	625	626	627	628	629	630	631	632	633	634	635	636	637	638	639	640	641	642	643	644	645	646	647	648	649	650	651	652	653	654	655	656	657	658	659	660	661	662	663	664	665	666	667	668	669	670	671	672	673	674	675	676	677	678	679	680	681	682	683	684	685	686	687	688	689	690	691	692	693	694	695	696	697	698	699	700	701	702	703	704	705	706	707	708	709	710	711	712	713	714	715	716	717	718	719	720	721	722	723	724	725	726	727	728	729	730	731	732	733	734	735	736	737	738	739	740	741	742	743	744	745	746	747	748	749	750	751	752	753	754	755	756	757	758	759	760	761	762	763	764	765	766	767	768	769	770	771	772	773	774	775	776	777	778	779	780	781	782	783	784	785	786	787	788	789	790	791	792	793	794	795	796	797	798	799	800	801	802	803	804	805	806	807	808	809	810	811	812	813	814	815	816	817	818	819	820	821	822	823	824	825	826	827	828	829	830	831	832	833	834	835	836	837	838	839	840	841	842	843	844	845	846	847	848	849	850	851	852	853	854	855	856	857	858	859	860	861	862	863	864	865	866	867	868	869	870	871	872	873	874	875	876	877	878	879	880	881	882	883	884	885	886	887	888	889	890	891	892	893	894	895	896	897	898	899	900	901	902	903	904	905	906	907	908	909	910	911	912	913	914	915	916	917	918	919	920	921	922	923	924	925	926	927	928	929	930	931	932	933	934	935	936	937	938	939	940	941	942	943	944	945	946	947	948	949	950	951	952	953	954	955	956	957	958	959	960	961	962	963	964	965	966	967	968	969	970	971	972	973	974	975	976	977	978	979	980	981	982	983	984	985	986	987	988	989	990	991	992	993	994	995	996	997	998	999	1000	1001	1002	1003	1004	1005	1006	1007	1008	1009	1010	1011	1012	1013	1014	1015	1016	1017	1018	1019	1020	1021	1022	1023	1024	1025	1026	1027	1028	1029	1030	1031	1032	1033	1034	1035	1036	1037	1038	1039	1040	1041	1042	1043	1044	1045	1046	1047	1048	1049	1050	1051	1052	1053	1054	1055	1056	1057	1058	1059	1060	1061	1062	1063	1064	1065	1066	1067	1068	1069	1070	1071	1072	1073	1074	1075	1076	1077	1078	1079	1080	1081	1082	1083	1084	1085	1086	1087	1088	1089	1090	1091	1092	1093	1094	1095	1096	1097	1098	1099	1100	1101	1102	1103	1104	1105	1106	1107	1108	1109	1110	1111	1112	1113	1114	1115	1116	1117	1118	1119	1120	1121	1122	1123	1124	1125	1126	1127	1128	1129	1130	1131	1132	1133	1134	1135	1136	1137	1138	1139	1140	1141	1142	1143	1144	1145	1146	1147	1148	1149	1150	1151	1152	1153	1154	1155	1156	1157	1158	1159	1160	1161	1162	1163	1164	1165	1166	1167	1168	1169	1170	1171	1172	1173	1174	1175	1176	1177	1178	1179	1180	1181	1182	1183	1184	1185	1186	1187	1188	1189	1190	1191	1192	1193	1194	1195	1196	1197	1198	1199	1200	1201	1202	1203	1204	1205	1206	1207	1208	1209	1210	1211	1212	1213	1214	1215	1216	1217	1218	1219	1220	1221	1222	1223	1224	1225	1226	1227	1228	1229	1230	1231	1232	1233	1234	1235	1236	1237	1238	1239	1240	1241	1242	1243	1244	1245	1246	1247	1248	1249	1250	1251	1252	1253	1254	1255	1256	1257	1258	1259	1260	1261	1262	1263	1264	1265	1266	1267	1268	1269	1270	1271	1272	1273	1274	1275	1276	1277	1278	1279	1280	1281	1282	1283	1284	1285	1286	1287	1288	1289	1290	1291	1292	1293	1294	1295	1296	1297	1298	1299	1300	1301	1302	1303	1304	1305	1306	1307	1308	1309	1310	1311	1312	1313	1314	1315	1316	1317	1318	1319	1320	1321	1322	1323	1324	1325	1326	1327	1328	1329	1330	1331	1332	1333	1334	1335	1336	1337	1338	1339	1340	1341	1342	1343	1344	1345	1346	1347	1348	1349	1350	1351	1352	1353	1354	1355	1356	1357	1358	1359	1360	1361	1362	1363	1364	1365	1366	1367	1368	1369	1370	1371	1372	1373	1374	1375	1376	1377	1378	1379	1380	1381	1382	1383	1384	1385	1386	1387	1388	1389	1390	1391	1392	1393	1394	1395	1396	1397	1398	1399	1400	1401	1402	1403	1404	1405	1406	1407	1408	1409	1410	1411	1412	1413	1414	1415	1416	1417	1418	1419	1420	1421	1422	1423	1424	1425	1426	1427	1428	1429	1430	1431	1432	1433	1434	1435	1436	1437	1438	1439	1440	1441	1442	1443	1444	1445	1446	1447	1448	1449	1450	1451	1452	1453	1454	1455	1456	1457	1458	1459	1460	1461	1462	1463	1464	1465	1466	1467	1468	1469	1470	1471	1472	1473	1474	1475	1476	1477	1478	1479	1480	1481	1482	1483	1484	1485	1486	1487	1488	1489	1490	1491	1492	1493	1494	1495	1496	1497	1498	1499	1500	1501	1502	1503	1504	1505	1506	1507	1508	1509	1510	1511	1512	1513	1514	1515	1516	1517	1518	1519	1520	1521	1522	1523	1524	1525	1526	1527	1528	1529	1530	1531	1532	1533	1534	1535	1536	1537	1538	1539	1540	1541	1542	1543	1544	1545	1546	1547	1548	1549	1550	1551	1552	1553	1554	1555	1556	1557	1558	1559	1560	1561	1562	1563	1564	1565	1566	1567	1568	1569	1570	1571	1572	1573	1574	1575	1576	1577	1578	1579	1580	1581	1582	1583	1584	1585	1586	1587	1588	1589	1590	1591	1592	1593	1594	1595	1596	1597	1598	1599	1600	1601	1602	1603	1604	1605	1606	1607	1608	1609	1610	1611	1612	1613	1614	1615	1616	1617	1618	1619	1620	1621	1622	1623	1624	1625	1626	1627	1628	1629	1630	1631	1632	1633	1634	1635	1636	1637	1638	1639	1640	1641	1642	1643	1644	1645	1646	1647	1648	1649	1650	1651	1652	1653	1654	1655	1656	1657	1658	1659	1660	1661	1662	1663	1664	1665	1666	1667	1668	1669	1670	1671	1672	1673	1674	1675	1676	1677	1678	1679	1680	1681	1682	1683	1684	1685	1686	1687	1688	1689	1690	1691	1692	1693	1694	1695	1696	1697	1698	1699	1700	1701	1702	1703	1704	1705	1706	1707	1708	1709	1710	1711	1712	1713	1714	1715	1716	1717	1718	1719	1720	1721	1722	1723	1724	1725	1726	1727	1728	1729	1730	1731	1732	1733	1734	1735	1736	1737	1738	1739	1740	1741	1742	1743	1744	1745	1746	1747	1748	1749	1750	175

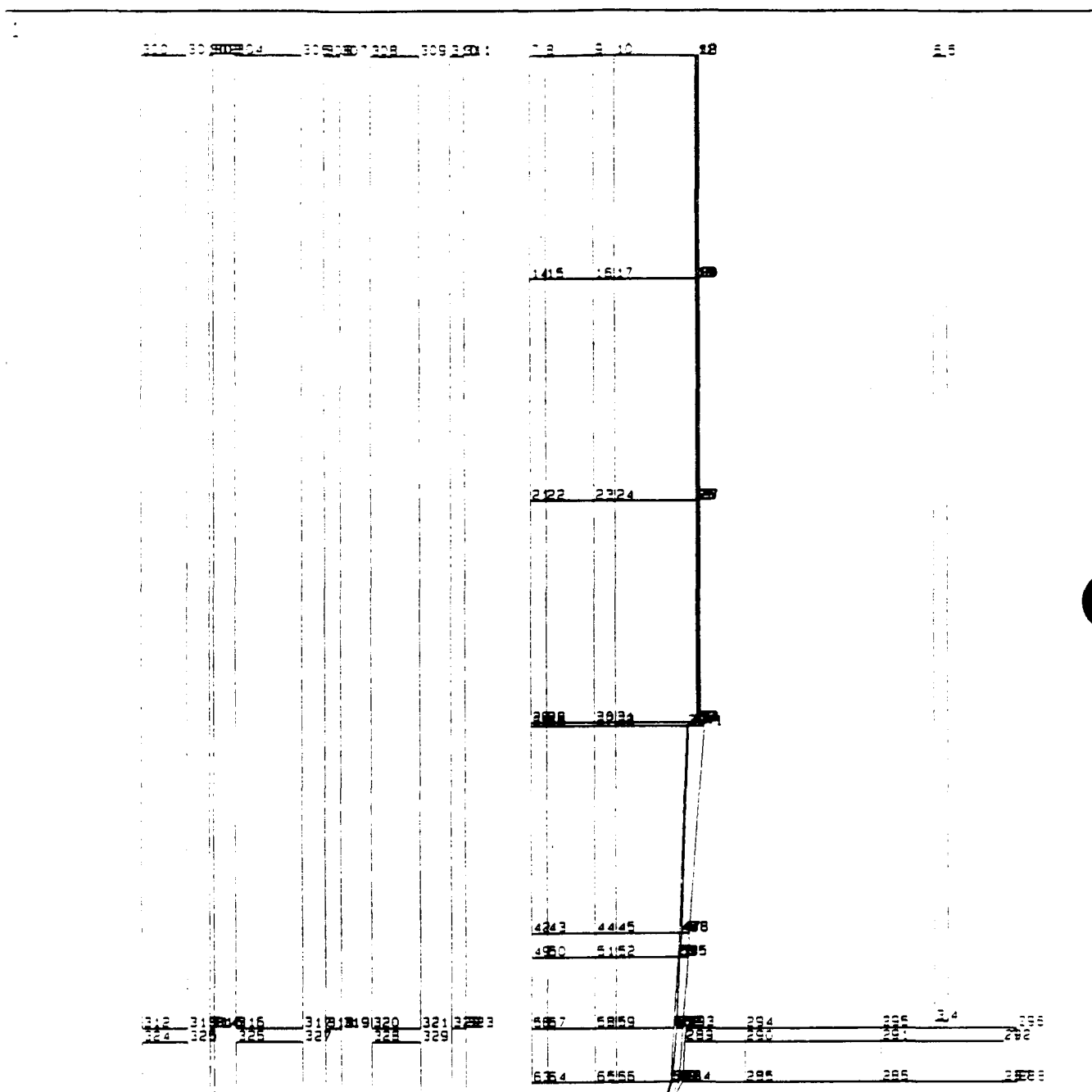


FIGURE 3.6.2-21
FINITE ELEMENT BREAKDOWN FOR AXISYMMETRIC THERMAL
MODEL OF CASK (LEFT SIDE)

NODE	TEMP	NODE	TEMP	NODE	TEMP
1	100.000	52	295.789	114	287.202
2	100.000	53	289.558	115	285.155
3	175.401	54	259.568	116	282.796
4	175.386	55	257.949	117	293.476
5	177.315	56	302.908	118	274.857
6	177.337	57	300.288	119	272.678
7	291.126	58	297.813	120	281.971
8	287.939	59	296.679	132	296.326
9	283.151	60	293.233	133	295.043
10	279.590	61	263.221	134	297.792
11	271.126	62	263.601	135	293.577
12	240.890	63	300.931	136	297.891
13	240.588	64	299.571	137	298.902
14	287.202	65	297.527	138	300.248
15	286.225	66	296.126	139	304.861
16	284.470	67	291.319	140	307.010
17	283.190	68	276.386	141	308.836
18	274.018	69	275.749	142	300.352
19	243.083	70	297.851	143	302.343
20	243.035	71	297.411	144	304.284
21	288.980	72	296.392	145	306.694
22	287.356	73	295.611	146	308.652
23	285.239	74	290.951	147	311.166
24	283.680	75	276.181	148	310.037
		76	276.037	149	311.299
		77	297.790	150	312.552
		78	296.228	151	313.784
25	275.037			152	314.174
26	243.966	79	294.256	153	313.990
27	243.790	80	293.165	154	317.092
28	289.646	81	290.649		
29	289.870	82	290.353	155	319.213
30	288.517	83	290.367	156	321.733
31	287.262	84	295.340	157	325.528
32	279.198	85	295.340	158	328.707
33	246.037	86	293.117	159	332.879
34	246.116	87	292.269	160	321.557
35	291.815	88	288.765	161	323.518
36	290.291	100	295.875	162	326.300
37	288.598	101	294.651	163	329.878
38	286.829	102	292.279	164	331.951
39	279.398	103	291.191	165	333.340
40	248.342	104	289.095	200	278.134
41	245.061	105	293.543	201	265.979
42	300.693	106	291.624	202	250.741
43	298.999	107	290.280	203	203.268
44	297.105	108	288.486	204	151.275
45	296.566	109	295.216	205	115.473
46	289.771	110	290.023	206	109.728
47	252.027	111	288.352	207	277.486
48	253.175	112	285.917	208	265.126
49	297.852	113	294.026	209	249.696
50	298.528				
51	297.367				

FIGURE 3.6.2-22
 NORMAL CONDITION STEADY STATE NODAL TEMPERATURES
 FROM CASK AXISYMMETRIC THERMAL MODEL

NODE	TEMP	NODE	TEMP	NODE	TEMP
210	201.961	261	113.483	316	482.434
211	150.403	263	118.045	317	480.600
212	115.177	264	117.853	318	393.673
213	109.497	265	117.872	319	393.589
214	275.689	266	117.665	320	445.117
215	262.948	267	117.447	321	444.037
216	247.070	268	113.237	322	361.428
217	198.527	269	263.318	323	361.333
218	148.185	270	230.509	324	386.839
219	114.761	271	172.205	325	386.391
220	109.434	272	123.727	326	383.980
221	273.006	273	118.057	327	383.157
222	259.243	274	255.912	328	357.797
223	241.885	275	227.504	329	357.946
224	191.505	276	173.720	330	333.514
225	144.223	277	123.945	331	334.642
226	114.088	278	117.768	332	333.526
227	109.330	279	242.608	333	333.922
228	272.745	280	220.428	334	314.130
229	255.604	281	176.370	335	312.282
230	234.267	282	124.025	336	371.384
231	183.521	283	118.090	337	371.381
232	140.345	284	231.346	338	369.073
233	113.501	285	213.266	339	369.014
234	109.340	286	183.221	340	338.289
235	272.338	287	135.128	341	338.176
236	242.726	288	119.026		
237	220.550	289	224.874		
238	174.228	290	208.827		
239	136.114	291	184.888		
240	112.825	292	163.928		
241	109.245	293	222.621		
242	223.626	294	206.994		
		295	185.237		
		296	173.872		
		300	506.337		
243	216.227				
244	202.222				
245	164.464				
246	131.845	301	503.181		
247	112.117	302	448.916		
248	109.171	303	448.889		
249	167.913	304	522.363		
250	164.522	305	518.185		
251	159.872	306	427.895		
252	141.979	307	427.596		
253	123.384	308	475.544		
254	111.132	309	472.798		
255	109.099	310	384.709		
256	123.179	311	384.329		
257	122.751	312	466.243		
258	122.106	313	464.858		
259	120.159	314	409.559		
260	117.764	315	409.552		

MAXIMUMS
NODE 304
VALUE 522.363

FIGURE 3.6.2-23
NORMAL CONDITION STEADY STATE NODAL TEMPERATURES
FROM CASK AXISYMMETRIC THERMAL MODEL

APPENDIX 3.6.3
THIRTY DEGREE 2-D MODEL OF CASK WALL INPUT

Date: 1 Sep 1989
Time: 9:20a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\FININTFS\FININTF
1 Sep 1989

/TITLE,140-B COPPER FIN THERMAL SHIELD INTERFACE

C*** DEFINE ELEMENT TYPES

ET,1,55	*2-D ISOPAR THM SLD
ET,2,32	*2-D COND LINK
ET,3,31	*RADIATION LINK
ET,4,34	*CONVECTION LINK

C*** MATERIAL PROPERTIES

KXX,1,.650,3.94E-4	*XM-19 SS(B/H-IN-F)
KXX,2,1.708,-4.83E-4	*LEAD
KXX,3,.0242	*SILICONE
KXX,4,18.75	*COPPER
KXX,5,1.08E-3,1.90E-6	*AIR OR NITROGEN

C,1,.120	*B/LBM-F
C,2,.030	
C,3,.460	
C,4,.092	

DENS,1,.290	*LBM/CU-IN
DENS,2,.410	
DENS,3,.057	
DENS,4,.322	

HF,6,2.08E-3,8.60E-6	*NAT CONV EXT TO TML SHIELD(B/SQIN-HR-DEGF)

C*** OPTIONS

KAN,-1	*THERMAL ANALYSES
ITER,-20,20,20	*OPT CONVERGENCE PO @ LAST ITER NO LIST
KBC,1	*STP BCS FOR NEXT LOAD STEP
TOFFST,460	*RAD LINKS PRES. SIG DEG R
TIME,0	*STEADY STATE ANALYSIS(DFT)
CSYS,1	*R VS THETA CYL COORD SYS

C*** REAL CONSTANTS

R,1,7.18	* SQ IN (41.13 IN)X(10 DEG)/
	* (57.3 RM PER DEG)
R,2,7.18,.67,1	* A.F.E.(SIGMA DFT .119E-10
	* B/HR-SQIN-DEGR4)
R,3,7.22	* SQ IN (41.38 IN)X(10 DEG)/
	* (57.3 RM PER DEG)
R,4,7.22,1,.25	* A.F.E.(SIGMA DFT .119E-10
	* B/HR-SQIN-DEGR4)
R,5,2.0	* WELDLEG AREA
R,6,.125	* WELDLEG AREA(TWO QTR IN SECTIONS)

C*** SPECIFYING NODES AND LOCATIONS (R,THETA,Z)

N,1,50,80
N,2,50,70
N,3,28,90
N,4,28,75

FIGURE 3.6.3-1

THIRTY DEGREE 2-D MODEL OF CASK WALL INPUT

Date: 1 Sep 1989
Time: 9:20a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CIRC\FININTFS\FININTF
1 Sep 1989

N.5.28.60
N.6.29.13.90
N.7.29.13.75
N.8.29.13.60
N.9.32.63.90
N.10.32.63.75
N.11.32.63.60
N.12.34.13.90
N.13.34.13.75
N.14.34.13.60
N.15.40.17.90
N.16.40.17.60
N.17.41.01.90
N.18.41.01.75
N.19.34.84.75
N.20.41.01.60
N.23.41.13.90
N.24.41.13.75
N.25.41.13.60
N.26.41.26.90
N.27.41.26.75
N.28.41.26.60
N.29.41.38.90
N.30.41.38.75
N.31.41.38.60

C*** SPECIFYING MODEL ELEMENTS

TYPE,1 *INNER CONTAINMENT SHELL(304 SS)

MAT,1

E.3,4,7,6

E.4,5,8,7

MAT,2

*LEAD GAMMA SHIELD

E.6,7,10,9

E.7,8,11,10

MAT,1

*OUTER CASK SHELL(304 SS)

E.9,10,13,12

E.10,11,14,13

MAT,3

*NEUTRON SHIELD(BORO-SILICONE)

E.12,13,15,15

E.13,14,16,16

E.19,18,17,17

E.19,20,18,18

MAT,4

*COPPER FINS

E.13,19,17,15

E.13,16,20,19

E.17,18,24,23

E.18,20,25,24

MAT,1

*OUTER THERMAL SHIELD

E.26,27,30,29

E.27,28,31,30

TYPE,2

*AIR CONDUCTION FINS TO TML SHLD

REAL,1

FIGURE 3.6.3-2

THIRTY DEGREE 2-D MODEL OF CASK WALL INPUT

Date: 1 Sep 1989
Time: 9:20a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\FININTFS\FININTF
1 Sep 1989

MAT.5
E.23.26
E.24.27
E.25.28

TYPE.3
REAL.2
E.23.26
E.24.27
E.25.28

*RADIATION FINS TO TML SHLD

TYPE.4
MAT.6
REAL.3
E.29.2
E.30.2
E.31.2

*NAT CONV EXT TML SHLD

TYPE.3
REAL.4
E.29.1
E.30.1
E.31.1

*RADIATION EXT TML SHLD

TYPE.2
MAT.4
REAL.5
E.13.19
REAL.6
E.17.23
E.20.25

*CONDUCTION AT FIN ENDS

C*** SPECIFYING MODEL LOADS

NT.1,TEMP.177
NT.2,TEMP.100

*RADIATION HEAT SINK/SOURCE
*CONVECTION HEAT SINK/SOURCE

HFLOW.3,HEAT,5.02

*FUEL ROD ASSY DECAY HEAT (B/HR) (4.89
* SQIN X 11 KW X 3.413 B/HR-W)/ (PI X 56
* IN X 180 IN + 2 X PI*(28)**2)

HFLOW.4,HEAT,5.02
HFLOW.5,HEAT,5.02

C***LOAD STEP 1 (11KW)

PRDISP.20
PRNP,-1
PRSTR,-1

AFWRITE

FINISH

FIGURE 3.6.3-3

THIRTY DEGREE 2-D MODEL OF CASK WALL INPUT

APPENDIX 3.6.4
QUARTER 2-D MODEL OF PWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:46a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTPWRS\BASKETSS
1 Sep 1989

/TITLE,140-B BASKET STEEL/COPPER COMP (NORMAL)
*** DEFINE ELEMENT TYPES

ET,1.55	*2-D ISOPAR THM SLD
ET,2.32	*2-D COND LINK
ET,3.31	*RADIATION LINK
ET,4.34	*CONVECTION LINK

C*** MATERIAL PROPERTIES

KXX,1..778,2.52E-4	*304 SS(B/H-IN-F)
KXX,2..650,3.94E-4	*XM-19 STEEL
KXX,3.12.5	*BORATED COPPER
KXX,4.18.75	*COPPER
KXX,5..124	*EQUIVALENT @ STEEL CORNER KEYS
	* (1.0 X .125, HC = 30 BTU/FT2-HR-F)
KXX,6.1.08E-3,1.9E-6	*AIR OR NITROGEN
KXX,7.7.92E-4,1.1E-6	*ARGON
KXX,8.1.63E-2,1.71E-5	*HELIUM

C,1..120	*B/LBM-F
C,2..120	
C,3..092	
C,4..092	
C,5..120	

DENS,1..290	*LBM/CU-IN
DENS,2..290	
DENS,3..322	
DENS,4..322	
DENS,5..290	

C*** OPTIONS

KAN,-1	*THERMAL ANALYSES
ITER,-20,20,20	*OPT CONVERGENCE PO @ LAST ITER NO LIST
KBC,1	*STP BCS FOR NEXT LOAD STEP
TOFFST,460	*RAD LINKS PRES, SIG DEG R
TIME,0	*STEADY STATE ANALYSIS(DFT)

C*** REAL CONSTANTS

R,1..165	* SQ IN
R,2..04	
R,3..09	
R,4.4.38,1..33	* A.F.E. (SIGMA DFT .119E-10
	* B/HR-SQIN-DEGR4)
R,5.8.76,1..33	
R,6.8.36,1..33	
R,7.4.00,1..33	
R,8.4.55,1..33	
R,9.9.10,1..33	
R,10.4.38,.5..33	
R,11.8.76,.5..33	
R,12.4.00	* GAS CONDUCTION AREAS
R,13.4.55	

FIGURE 3.6.4-1
QUARTER 2-D MODEL OF PWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:46a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTPWRS\BASKETSS
1 Sep 1989

R.14.9.10
R.15.4.38
R.16.8.76
R.17.8.36
R.18.4.38
R.19.8.76
R.20.4.38

R.21.125

*CORNER AREAS

R.22.062

*OUTER SHELL & EXTENSIONS

R.23.4.38

*COND AREAS FOR BASKET CAVITIES

R.24.8.76

R.25.8.36

C*** SPECIFYING NODES AND LOCATIONS

LOCAL.11.1.0.0

N.1.38.45

N.38.28.25.45

N.55.28.25.0

N.58.28.25.20

N.59.28.25.70

N.62.28.25.90

N.70.28.5.0

N.71.28.5.9

N.72.28.5.11

N.73.28.5.20

N.74.28.5.30.6

N.75.28.5.32.8

N.76.28.5.45

N.77.28.5.57.2

N.78.28.5.59.4

N.79.28.5.70

N.80.28.5.79

N.81.28.5.81

N.82.28.5.90

CSYS

N.2.4.38.0

N.3.4.38.4.38

N.4.0.4.38

N.5.5.38.0

N.6.5.38.4.38

N.7.5.38.5.38

N.8.4.38.5.38

N.9.0.5.38

N.10.9.76.4.38

N.11.9.76.5.38

N.12.5.38.9.76

N.13.4.38.9.76

N.14.14.14.0

N.15.14.14.4.38

N.16.14.14.5.38

N.17.14.14.9.76

N.18.14.14.14.14

FIGURE 3.6.4-2
QUARTER 2-D MODEL OF PWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:46a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CIRC\BSKTPWRS\BASKETSS
1 Sep 1989

N.19.9.76.14.14
N.20.5.38.14.14
N.21.4.38.14.14
N.22.0.14.14
N.23.15.14.0
N.24.15.14.4.38
N.25.15.14.5.38
N.26.15.14.9.76
N.27.15.14.14.14
N.28.15.14.15.14
N.29.14.14.15.14
N.30.9.76.15.14
N.31.5.38.15.14
N.32.4.38.15.14
N.33.0.15.14
N.34.19.52.4.38
N.35.19.52.5.38
N.36.19.52.14.14
N.37.19.52.15.14
N.39.15.14.19.52
N.40.14.14.19.52
N.41.5.38.19.52
N.42.4.38.19.52
N.43.23.9.0
N.44.23.9.4.38
N.45.23.9.5.38
N.46.23.9.9.76
N.47.23.9.14.14
N.48.23.5.15.14
N.49.15.14.23.5
N.50.14.14.23.9
N.51.9.76.23.9
N.52.5.38.23.9
N.53.4.38.23.9
N.54.0.23.9
N.56.27.65.4.38
N.57.27.6.5.38
N.60.5.38.27.60
N.61.4.38.27.65

C*** SPECIFYING MODEL ELEMENTS

TYPE.2
REAL.1
MAT.3
E.2.3
E.3.4
E.5.6
E.8.9
E.14.15
E.15.10
E.10.6
E.16.17
E.16.11
E.11.7
E.7.12
E.12.20

*BORATED COPPER POISON PANELS

FIGURE 3.6.4-3
QUARTER 2-D MODEL OF PWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:46a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTPWRS\BASKETSS
1 Sep 1989

E.17.18
E.18.19
E.19.20
E.21.22
E.21.13
E.13.8
E.23.24
E.24.34
E.34.44
E.25.26
E.25.35
E.35.45
E.26.27
E.27.36
E.36.47
E.29.30
E.29.40
E.40.50
E.30.31
E.31.41
E.41.52
E.32.33
E.32.42
E.42.53
E.43.44
E.45.46
E.46.47
E.50.51
E.51.52
E.53.54

MAT.4
REAL.22
E.44.56
E.45.57
E.28.37
E.37.48
E.28.39
E.39.49
E.52.60
E.53.61

*STEEL EXTENSIONS

REAL.22
E.55.56
E.56.57
E.57.58
E.58.47
E.47.48
E.48.38
E.38.49
E.49.50
E.50.59
E.59.60
E.60.61
E.61.62

*STEEL OUTER SHELL

REAL.21
MAT.5
E.3.6

*COPPER/STEEL CORNER INTERFACES

FIGURE 3.6.4-4
QUARTER 2-D MODEL OF PWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:46a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTPWRS\BASKETSS
1 Sep 1989

E.6.7
E.7.8
E.8.3
E.15.24
E.24.25
E.25.16
E.16.15
E.18.27
E.27.28
E.28.29
E.29.18
E.21.20
E.20.31
E.31.32
E.32.21

TYPE.3

*RADIATION OUTER INTERNAL
*BASKET CAVITIES

REAL.4
E.43.55
E.54.62
REAL.5
E.46.58
E.51.59
REAL.6
E.37.38
E.39.38

TYPE.2

*GAS CONDUCTION OUTER INTERNAL
*BASKET CAVITIES

MAT.6
REAL.23
E.43.55
E.54.62
REAL.24
E.46.58
E.51.59
REAL.25
E.37.38
E.39.38

TYPE.3

*RADIATION FROM BASKET TO CASK

REAL.7
E.56.1
E.57.1
E.47.1
E.48.1
E.49.1
E.50.1
E.60.1
E.61.1
REAL.8
E.55.1
E.62.1
REAL.9
E.58.1
E.38.1
E.59.1

FIGURE 3.6.4-5
QUARTER 2-D MODEL OF PWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:46a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTPWRS\BASKETSS
1 Sep 1989

TYPE.2
MAT.6
REAL.12
E.56.71
E.57.72
E.47.74
E.48.75
E.49.77
E.50.78
E.60.80
E.61.81
REAL.13
E.55.70
E.62.82
REAL.14
E.58.73
E.38.76
E.59.79

*GAS CONDUCTION FROM BASKET TO CASK

TYPE.3

*RADIATION INNER INTERNAL BASKET
*CAVITIES

REAL.10
E.2.5
E.14.23
E.4.9
E.22.33
REAL.11
E.10.11
E.34.35
E.19.30
E.36.37
E.13.12
E.42.41
E.17.26
E.40.39

TYPE.2

*GAS CONDUCTION INNER INTERNAL BASKET
*CAVITIES

MAT.6
REAL.18
E.2.5
E.14.23
E.4.9
E.22.33
REAL.19
E.10.11
E.34.35
E.19.30
E.36.37
E.13.12
E.42.41
E.17.26
E.40.39

C*** SPECIFYING MODEL LOADS

NT.1.TEMP.291

*RADIATION HEAT SINK/SOURCE

NT.70.TEMP.291..82.1

*CONDUCTION HEAT SINK/SOURCE

FIGURE 3.6.4-6
QUARTER 2-D MODEL OF PWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:46a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTPWRS\BASKETSS
1 Sep 1989

HFLOW,2,HEAT,.030

*FUEL ROD ASSY DECAY HEAT (B/HR) (535
* WATTS X 3.413 B/HR-W)/ (8.76 X 4 X 144
* SQIN) TIMES SQIN PER TUBE DIVIDED BY
* NUMBER OF NODES

HFLOW,3,HEAT,1.057
HFLOW,4,HEAT,1.057
HFLOW,5,HEAT,1.268
HFLOW,6,HEAT,1.268
HFLOW,10,HEAT,1.268
HFLOW,15,HEAT,1.268
HFLOW,14,HEAT,1.268
HFLOW,23,HEAT,1.268
HFLOW,24,HEAT,1.268
HFLOW,34,HEAT,1.268
HFLOW,44,HEAT,1.268
HFLOW,43,HEAT,1.268
HFLOW,9,HEAT,1.268
HFLOW,8,HEAT,1.268
HFLOW,13,HEAT,1.268
HFLOW,21,HEAT,1.268
HFLOW,22,HEAT,1.268
HFLOW,33,HEAT,1.268
HFLOW,32,HEAT,1.268
HFLOW,42,HEAT,1.268
HFLOW,53,HEAT,1.268
HFLOW,54,HEAT,1.268
HFLOW,7,HEAT,1.585
HFLOW,11,HEAT,1.585
HFLOW,16,HEAT,1.585
HFLOW,17,HEAT,1.585
HFLOW,18,HEAT,1.585
HFLOW,19,HEAT,1.585
HFLOW,20,HEAT,1.585
HFLOW,12,HEAT,1.585
HFLOW,25,HEAT,1.585
HFLOW,35,HEAT,1.585
HFLOW,45,HEAT,1.585
HFLOW,46,HEAT,1.585
HFLOW,47,HEAT,1.585
HFLOW,36,HEAT,1.585
HFLOW,27,HEAT,1.585
HFLOW,26,HEAT,1.585
HFLOW,31,HEAT,1.585
HFLOW,30,HEAT,1.585
HFLOW,29,HEAT,1.585
HFLOW,40,HEAT,1.585
HFLOW,50,HEAT,1.585
HFLOW,51,HEAT,1.585
HFLOW,52,HEAT,1.585
HFLOW,41,HEAT,1.585

C***LOAD STEP 1 (11KW)

PRDISP,20
PRNF,-1
PRSTR,-1

FIGURE 3.6.4-7
QUARTER 2-D MODEL OF PWR BASKET INPUT

APPENDIX 3.6.5

QUARTER 2-D MODEL OF BWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:51a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTBWRS\BASKETSS
1 Sep 1989

/TITLE,140-B BASKET STEEL/COPPER COMP (NORMAL)
*** DEFINE ELEMENT TYPES

ET,1,55	*2-D ISOPAR THM SLD
ET,2,32	*2-D COND LINK
ET,3,31	*RADIATION LINK
ET,4,34	*CONVECTION LINK

C*** MATERIAL PROPERTIES

KXX,1,.778,2.52E-4	*304 SS(B/H-IN-F)
KXX,2,.650,3.94E-4	*XM-19 STEEL
KXX,3,12.5	*BORATED COPPER
KXX,4,18.75	*COPPER
KXX,5,2.79	*EQUIVALENT @ STEEL/COPPER CORNER KEYS
	* (.38 X.06, HC = 300 BTU/FT2-HR-F)
KXX,6,1.08E-3,1.90E-6	*AIR OR NITROGEN
KXX,7,7.92E-4,1.1E-6	*ARGON
KXX,8,1.63E-2,1.71E-5	*HELIUM

C,1,.120	*B/LBM-F
C,2,.120	
C,3,.092	
C,4,.092	
C,5,.120	

DENS,1,.290	*LBM/CU-IN
DENS,2,.290	
DENS,3,.322	
DENS,4,.322	
DENS,5,.290	

C*** OPTIONS

KAN,-1	*THERMAL ANALYSES
ITER,-20,20,20	*OPT CONVERGENCE PO @ LAST ITER NO LIST
KBC,1	*STP BCS FOR NEXT LOAD STEP
TOFFST,460	*RAD LINKS PRES, SIG DEG R
TIME,0	*STEADY STATE ANALYSIS(DFT)

C*** REAL CONSTANTS

R,1,.06	* SQ IN
R,2,3,13	
R,3,6,26	
R,4,3,13,.5,.3	* A.F.E.(SIGMA DFT .119E-10
	* B/HR-SQIN-DEGR4)
R,5,6,26,.5,.3	
R,6,3,21,.23,1.	*RAD AREA/SCRIPT? EXTERNAL
R,7,6,41,.23,1.	
R,8,7,90,.23,1.	
R,9,9,37,.23,1.	
R,10,3,21	* GAS CONDUCTION AREAS
R,11,6,41	
R,12,7,90	

FIGURE 3.6.5-1
QUARTER 2-D MODEL OF BWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:51a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTBWRS\BASKETSS
1 Sep 1989

R.13.9.37

C*** SPECIFYING NODES AND LOCATIONS

LOCAL.11.1.0.0
N.1.38.45
N.90.28.25.90
N.91.28.25.77.5

N.92.28.25.63.5
N.93.28.25.45
N.94.28.25.26.5
N.95.28.25.12.5
N.96.28.25.0

N.100.28.5.90
N.101.28.5.77.5
N.102.28.5.63.5
N.103.28.5.45
N.104.28.5.26.5
N.105.28.5.12.5
N.106.28.5.0

CSYS
N.2.0.0
RP5.3.6.26.0
N.26.0.6.26
RP5.3.6.26.0
N.49.0.12.52
RP4.3.6.26.0
N.68.0.18.78
RP4.3.6.26.0
N.83.0.25.04
RP2.3.6.26.0

N.3.38.0
RP5.3.6.26.0
N.27.38.6.26
RP5.3.6.26.0
N.50.38.12.52
RP4.3.6.26.0
N.69.38.18.78
RP4.3.6.26.0
N.84.38.25.04
RP2.3.6.26.0

N.4.5.88.0
RP4.3.6.26.0
N.28.5.88.6.26
RP4.3.6.26.0
N.51.5.88.12.52
RP3.3.6.26.0
N.70.5.88.18.78
RP3.3.6.26.0
N.85.5.88.25.04

N.16.0.38
RP5.1.6.26.0
N.21.0.5.88
RP5.1.6.26.0

FIGURE 3.6.5-2
QUARTER 2-D MODEL OF BWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:51a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTBWRS\BASKETSS
1 Sep 1989

N,40,0,6.64
RP5,1,6.26,0
N,45,0,12.14
RP4,1,6.26,0
N,60,0,12.9
RP4,1,6.26,0
N,64,0,18.4
RP4,1,6.26,0
N,78,0,19.16
RP3,1,6.26,0
N,81,0,24.66
RP2,1,6.26,0
N,88,0,25.42
RP2,1,6.26,0

C*** SPECIFYING MODEL ELEMENTS

TYPE,2 *PANELS (BASICALLY COPPER)
REAL,1

MAT,4
E,3,4
RP4,3,3
E,27,28
RP4,3,3
E,50,51
RP3,3,3
E,69,70
RP3,3,3
E,84,85
E,87,92
E,16,21
RP5,1,1
E,40,45
RP4,1,1
E,44,94
E,60,64
RP4,1,1
E,78,81
RP2,1,1
E,80,92

E,15,96 *STEEL EXTENSIONS
E,39,95
E,77,93
E,89,91
E,88,90

E,96,95 *STEEL OUTER SHELL
E,95,94
E,94,93
E,93,92
E,92,91
E,91,90

MAT,5 *PANEL JUNCTION INTERFACES
E,2,3
RP5,3,3
E,4,5
RP4,3,3

FIGURE 3.6.5-3
QUARTER 2-D MODEL OF BWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:51a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTBWRS\BASKETSS
1 Sep 1989

E.2.16
RP5.3.1
E.21.26
RP5.1.3
E.26.27
RP5.3.3
E.28.29
RP4.3.3
E.26.40
RP5.3.1
E.45.49
RP4.1.3
E.49.50
RP4.3.3
E.51.52
RP3.3.3
E.49.60
RP4.3.1
E.64.68
RP3.1.3
E.68.69
RP3.3.3
E.70.71
RP3.3.3
E.68.78
RP3.3.1
E.81.83
RP2.1.3
E.83.84
RP2.3.3
E.85.86
E.83.88
RP2.3.1

TYPE.2

*CONDUCTION OUTER INTERNAL
*BASKET CAVITIES

MAT.6
REAL.2
E.88.90
E.89.91
E.39.95
E.15.96
REAL.3
E.89.91
E.87.92
E.44.94
E.80.92
E.77.93
E.59.94

TYPE.3

*RADIATION OUTER INTERNAL
*BASKET CAVITIES

REAL.4
E.84.90
E.85.91
E.25.95
E.20.96
REAL.5
E.87.91

FIGURE 3.6.5-4
QUARTER 2-D MODEL OF BWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:51a

Nuclear Packaging, Inc.
Federal Way, WA

D:\NCRC\BSKTBWRS\BASKETSS
1 Sep 1989

E.87.92
E.44.94
E.44.95
E.80.92
E.80.93
E.75.92
E.75.93
E.63.93
E.63.94
E.59.94
E.59.93

TYPE.3
REAL.6
E.90.1
E.96.1
REAL.7
E.91.1
E.95.1
REAL.8
E.92.1
E.94.1
REAL.9
E.93.1

*RADIATION FROM BASKET TO CASK

TYPE.2
MAT.6
REAL.10
E.90.100
E.96.106
REAL.11
E.91.101
E.95.105
REAL.12
E.92.102
E.94.104
REAL.13
E.93.103

*GAS CONDUCTION FROM BASKET TO CASK

C*** SPECIFYING MODEL LOADS

NT.1,TEMP.291

*RADIATION HEAT SINK/SOURCE

NT.100,TEMP.291..106.1

*CONDUCTION HEAT SINK/SOURCE

HFLOW.3,HEAT..633..12.3

*FUEL ROD ASSY DECAY HEAT (B/HR) (213
* WATTS X 3.413 B/HR-W)/ (5.93 X 4 X 144
* SQIN) TIMES SQIN PER TUBE DIVIDED BY
* NUMBER OF NODES

HFLOW.4,HEAT..633..13.3
HFLOW.16,HEAT..633
HFLOW.17,HEAT.1.27..19.1
HFLOW.20,HEAT..633
HFLOW.21,HEAT..633
HFLOW.22,HEAT.1.27..24.1
HFLOW.25,HEAT..633
HFLOW.27,HEAT.1.27..36.3
HFLOW.28,HEAT.1.27..37.3
HFLOW.40,HEAT..633

FIGURE 3.6.5-5
QUARTER 2-D MODEL OF BWR BASKET INPUT

Date: 1 Sep 1989
Time: 9:51a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\BSKTSWRS\BASKETSS
1 Sep 1989

HFLOW,41,HEAT,1.27,,43.1
HFLOW,44,HEAT,.633
HFLOW,45,HEAT,.633
HFLOW,46,HEAT,1.27,,48.1
HFLOW,50,HEAT,1.27,,56.3
HFLOW,59,HEAT,.633
HFLOW,51,HEAT,1.27,,57.3
HFLOW,94,HEAT,1.27
HFLOW,60,HEAT,.633
HFLOW,61,HEAT,1.27,,62.1
HFLOW,63,HEAT,.633
HFLOW,64,HEAT,.633
HFLOW,65,HEAT,1.27,,66.1
HFLOW,67,HEAT,.633
HFLOW,69,HEAT,1.27,,72.3
HFLOW,75,HEAT,.633
HFLOW,70,HEAT,1.27,,73.3
HFLOW,76,HEAT,.633
HFLOW,78,HEAT,.633
HFLOW,79,HEAT,1.27
HFLOW,80,HEAT,.633
HFLOW,81,HEAT,.633
HFLOW,82,HEAT,1.27
HFLOW,92,HEAT,1.27
HFLOW,84,HEAT,.633,,87.3
HFLOW,85,HEAT,.633

C***LOAD STEP 1 (11KW)

PRDISP,20
PRSTR,-1

AFWRITE

FINISH

FIGURE 3.6.5-6
QUARTER 2-D MODEL OF BWR BASKET INPUT

APPENDIX 3.6.6

FULL 2-D MODEL OF CENTER PWR FUEL ASSEMBLY INPUT

Date: 1 Sep 1989
Time: 11:22a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRO\FUELASSY\TEMP
1 Sep 1989

/TITLE. FUEL ASSY THERMAL ANALYSIS (ACCIDENT CONDITIONS)

C***DEFINE ELEMENTS

ET.1.55	*2-D ISOPARAMETRIC SOLID
ET.2.32	*2-D CONDUCTION LINK
ET.3.31	*RADIATION LINK

C***DEFINE MATERIAL PROPERTIES

KXX.2.3850	*URANIUM DIOXIDE(X 10000)

C***REAL CONSTANTS

R.1.105
R.2.281
R.3.468
R.4.655
R.5.842
R.6.1028
R.7.1215
R.8.1915

R.9.187..676.1.
R.10.187..149.1.
R.11.243..360.1.
R.12.243..078.1.
R.13.486..355.1.
R.14.486..078.1.
R.15.729..352.1.
R.16.729..078.1.
R.17.972..351.1.
R.18.972..077.1.
R.19.1215..350.1.
R.20.1215..077.1.
R.21.1458..350.1.
R.22.1458..061.1.
R.23.1701..156.1.

C***OPTIONS

KAN,-1	*THERMAL ANALYSIS
ITER,-10.10.10	*TEN ITERATIONS MAX. LIST TENTH. SAVE TENTH DA
TA	
KBC.1	*STEP BOUNDARY CONDITIONS
TOFFST.460	*RAD LINKS WITH TEMPS IN DEG F
KTEMP,-1	*HEAT GENERATION PRESENT

C***NODE DEFINITIONS

N.1.0.3
N.2..33.3
N.3..33.3.33
N.4.0.3.33
NGEN.32.4.ALL....563
N.33.4.504.0
N.34.4.614.0
N.35.4.614.6.33

FIGURE 3.6.6-1

FULL 2-D MODEL OF CENTER PWR FUEL ASSEMBLY INPUT

Date: 1 Sep 1989
Time: 11:22a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CROC\FUELASSTEMP
1 Sep 1989

N.36.4.504.6.33

C***ELEMENT DESCRIPTIONS

TYPE.1
MAT.2
E.1.2.3.4
RP8.4.4.4.4
E.33.34.35.36

*FUEL ROD ISO PARA SOLIDS

TYPE.3
REAL.9
E.2.5
E.3.8
REAL.10
E.2.9
E.3.12

*RADIATION LINKS

REAL.11
E.6.9
E.7.12
REAL.12
E.6.13
E.7.16

REAL.13
E.10.13
E.11.16
REAL.14
E.10.17
E.11.20

REAL.15
E.14.17
E.15.20
REAL.16
E.14.21
E.15.24

REAL.17
E.18.21
E.19.24
REAL.18
E.18.25
E.19.28

REAL.19
E.22.29
E.23.28
REAL.20
E.22.29
E.23.32

REAL.21
E.26.29
E.27.32
REAL.22

~

FIGURE 3.6.6-2

FULL 2-D MODEL OF CENTER PWR FUEL ASSEMBLY INPUT

Date: 1 Sep 1989
Time: 11:22a

Nuclear Packaging, Inc.
Federal Way, WA

D:\CRC\FUELASSEMBLY\TEMP
1 Sep 1989

E.26.33
E.27.36

REAL.23
E.30.33
E.31.36

***SPECIFYING LOADS

NT.33,TEMP.602
NT.34,TEMP.602
NT.35,TEMP.602
NT.36,TEMP.602

PRDISP.10
PRNF,-1
PRSTR,-1

HFLOW.5,HEAT.1.755,.8
HFLOW.9,HEAT.3.5,.12
HFLOW.13,HEAT.4.375,.16
HFLOW.17,HEAT.6.15,.20
HFLOW.21,HEAT.6.15,.24
HFLOW.25,HEAT.10.525,.28
HFLOW.29,HEAT.12.275,.32

LWRITE

PRDISP.10
PRNF,-1
PRSTR,-1

HFLOW.5,HEAT.17.55,.8
HFLOW.9,HEAT.35,.12
HFLOW.13,HEAT.43.75,.16
HFLOW.17,HEAT.61.5,.20
HFLOW.21,HEAT.61.5,.24
HFLOW.25,HEAT.105.25,.28
HFLOW.29,HEAT.122.75,.32

LWRITE

AFWRITE

FINISH

FIGURE 3.6.6-3
FULL 2-D MODEL OF CENTER PWR FUEL ASSEMBLY INPUT

APPENDIX 3.6.7

**Accident Condition Equilibrium Prior To 30 minute,
1475°F Hypothetical Thermal Test Condition.
Nodal Temperatures From Axisymmetric Cask Thermal Model**

NODE	TEMP	NODE	TEMP	NODE	TEMP
1	100.000	56	245.836	118	222.169
2	100.000	57	243.120	119	220.101
7	225.817	58	240.685	120	229.071
8	222.389	59	239.560	132	243.488
9	217.471	60	236.262	133	242.137
10	213.691	61	205.600	134	245.033
11	204.933	62	205.977	135	240.592
12	167.135	63	244.569	136	245.158
13	166.816	64	243.212	137	246.173
14	221.344	65	241.266	138	247.487
		66	239.938	139	252.200
		67	235.638	140	254.358
		68	220.931	141	256.212
15	220.396			142	247.923
16	218.705			143	249.752
17	217.390			144	251.665
18	207.925	69	220.356		
19	168.964	70	242.611	145	254.065
20	168.912	71	242.158	146	256.032
21	224.022	72	241.195	147	258.561
22	222.411	73	240.458	148	257.827
23	220.364	74	236.386	149	258.994
24	218.761	75	222.632	150	260.175
25	209.784	76	222.526	151	261.339
26	170.287	77	243.509	152	261.683
27	170.096	78	241.892	153	261.444
28	227.209	79	239.982	154	265.095
29	227.420	80	238.963	155	267.090
30	226.025	81	236.693	156	269.561
31	224.659	82	236.406	157	273.336
32	216.087	83	236.403	158	276.526
33	174.178	84	241.250	159	280.735
34	174.263	85	241.261	160	269.591
35	229.444	86	239.110	161	271.475
36	227.897	87	238.296	162	274.205
37	226.166	88	235.229	163	277.734
38	224.243	100	241.943	164	279.766
39	216.390	101	240.756	165	281.087
40	176.720	102	238.477	200	224.882
41	173.117	103	237.453	201	215.432
42	241.957	104	235.541	202	203.657
43	240.196	105	239.844	203	167.582
44	238.325	106	238.025	204	129.460
45	237.803	107	236.722	205	104.429
46	230.917	108	235.021	206	100.541
47	185.751	109	241.980	207	224.312
48	186.969	110	236.713	208	214.702
49	239.437	111	235.050	209	202.782
50	240.146	112	232.723	210	166.538
51	239.013	113	240.960	211	128.796
52	237.413	114	234.157	212	104.230
53	231.162	115	232.094	213	100.395
54	195.585	116	229.770		
55	193.878	117	240.488		

FIGURE 3.6.7-1
ACCIDENT CONDITION EQUILIBRIUM PRIOR TO FIRE NODAL
TEMPERATURES FROM AXISYMMETRIC CASK THERMAL MODEL

NODE	TEMP	NODE	TEMP	NODE	TEMP
214	222.732	266	100.199	320	407.932
215	212.844	267	100.089	321	406.935
216	200.604	268	100.029	322	312.557
217	163.809	269	211.388	323	312.460
218	127.105	270	185.535	324	334.958
219	103.941	271	140.465	325	334.519
220	100.368	272	104.521	326	332.745
221	220.385	273	100.432	327	331.987
222	209.750	274	204.514	328	305.762
223	196.378	275	182.246	329	306.041
224	158.263	276	140.994	330	281.258
225	124.088	277	104.598	331	282.360
226	103.435	278	100.237	332	281.379
227	100.272	279	192.326	333	281.762
228	220.158	280	174.830	334	261.576
229	206.799	281	141.191	335	259.661
230	190.276	282	104.405	336	324.552
231	151.988	283	100.420	337	324.549
232	121.137	284	182.223	338	321.752
		285	167.557	339	321.693
		286	143.950	340	289.577
		287	111.348	341	289.459
233	103.010				
234	100.281				
235	219.784				
236	196.539	288	100.942		
237	179.402	289	176.536		
238	144.704	290	162.975		
239	117.902	291	143.792		
240	102.526	292	130.471		
241	100.236	293	174.579		
242	181.019	294	161.106		
243	175.568	295	143.527		
244	164.983	296	137.290		
245	137.054	300	467.038		
246	114.589	301	463.883		
247	102.028	302	401.254		
248	100.181	303	401.227		
249	137.757	304	486.839		
250	135.383	305	482.791		
251	132.102	306	379.959		
252	119.346	307	379.661		
253	107.417	308	438.217		
254	101.014	309	435.737		
255	100.098	310	336.028		
256	104.180	311	335.649		
257	103.924	312	427.436		
258	103.474	313	426.046		
259	102.121	314	361.866		
260	100.801	315	361.859		
261	100.127	316	447.041		
263	100.402	317	445.262		
264	100.333	318	345.675		
265	100.300	319	345.590		

MAXIMUMS	
NODE	304
VALUE	486.839

FIGURE 3.6.7-2
 ACCIDENT CONDITION EQUILIBRIUM PRIOR TO FIRE NODAL TEMPERATURES
 FROM AXISYMMETRIC CASK THERMAL MODEL

APPENDIX 3.6.8

COMPARISONS OF NuPac 140-B CASK ANALYTICAL THERMAL
MODEL RESULTS WITH THE WESTINGHOUSE MC-10 STORAGE
CASK TEST DATA GIVEN IN REFERENCE (m)

COMPARISON OF WESTINGHOUSE MC-10 CASK EXPERIMENTAL THERMAL
DATA WITH NuPac 140-B CASK ANALYTICAL THERMAL RESULTS

TEMPERATURE DIFFERENCES	MC-10	140-B
o Max Cladding to Cask Wall ID's	256 °F (2) (229 Avg)	369 °F (1)
o Correction for Wall Area & Heat Load Diff- erences	-16 °F (-14 Avg)	N/A
o Correction for Higher Conductivity of MC-10 .4 in Aluminum Basket Walls	172 °F (115 Avg)	N/A
CORRECTED COMPARISON	412 °F (330 Avg)	369 °F

CONCLUSION: 140-B maximum cladding temperatures could be
lower by 39 F or higher by 43 °F.

NOTE (1) Assumes normal conditions, 660 °F - 291 °F = 369 °F
per Table 3.7 and Figure 3.9 with nitrogen as the
cover gas.

NOTE (2) Based on nitrogen test/analysis results with wall
at 73 °C and cladding at 200 °C (Avg) and 215 °C (Max)
taken from Figure 5-28 of Reference (m).

The average MC-10 data above are based on an average of highest and lowest cladding temperatures across the MC-10 basket. The other temperature difference is based on the highest cladding temperatures. All MC-10 data are based on the horizontal orientation which was reported to be worst case.

The MC-10 cask has an effective inside wall area 34800 sqin and a test payload heat input of 12.7 KW. In comparison, the NuPac 140-B Cask has an effective inside wall area of 32200 sqin and an assumed PWR payload heat input of 11.2 KW. Thus, the MC-10 cask has 8% more wall area, which would increase heat dissipation and reduce interior temperature difference, but it also has 13% more heat to dissipate, which increases the interior temperature difference. The net result would be a 5% greater spread in temperature difference when compared to the NuPac 140-B Cask. Thus, -16°F is shown in the table above as an adjustment to the MC-10 temperature difference under the assumption that MC-10 had operated at the same 11.2 KW heating rate and dissipated the heat through the same 32200 sqin area as the NuPac 140-B Cask.

The MC-10 basket consisted of .400 inch aluminum plates put together to form the compartments for the 24 square PWR fuel assembly payload. The NuPac 140-B Cask has a similar design with .128 inch equivalent aluminum plates. The NuPac 140-B Cask .128 inch equivalent aluminum thickness includes basket diameter effects (with the MC-10 being 68 inches and the NuPac 140-B Cask being 57 inches) and includes effects of the copper heat transfer material. It also includes the effects of contact conductance in the panel joints.

Thus, the MC-10 basket has more heat transfer material in its structure to conduct heat away. This decreases the interior temperature difference in the basket the ratio of the wall thickness, or by a factor of $.400/.128 = 3.125$. From the test data of Reference (m), Figure 5-28, the maximum basket temperature difference is 150°C - 105°C and the average basket temperature difference is 135°C - 105°C, or 81°F and 54°F respectively. Multiplying these two values by 3.125 and subtracting 81°F and 54°F from the results gives 172°F and 115°F. These two values are shown in the table above as adjustments to the MC-10 temperature

APPENDIX 3.6.9

**ANSYS-PC\THERMAL ELEMENT DESCRIPTIONS
UTILIZED FOR 140-B CASK THERMAL MODELS**

SUMMARY

This appendix contains a description of the four specific finite elements utilized to formulate 140-B Cask thermal models. The descriptions are taken without change from the ANSYS-PC\Thermal manual of Reference (t).

STIF31

11.31 Radiation Link

The radiation link is a uniaxial element which defines the radiation heat flow rate between two points in space. The element has a single degree of freedom (temperature) at each node. The radiation element is applicable to both steady-state and transient thermal analyses in two (plane and axisymmetric) and three dimensions.

An empirical relationship in which the form factor and the area are independently multiplied by the temperatures is also available. The emissivity may be temperature dependent.

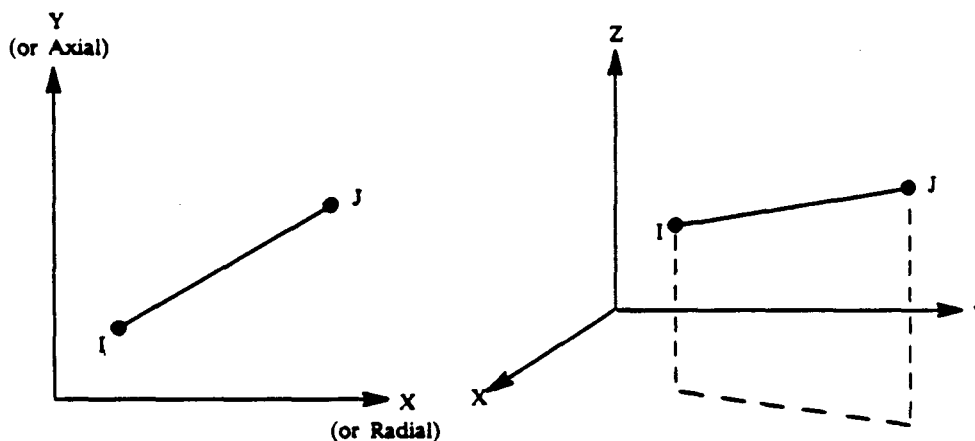


Figure 11.31.1 Radiation Link

■ Input Data

The geometry, node locations, and the coordinate systems for the radiation element are shown in Figure 11.31.1. The element is defined by two nodes, a radiating surface area (AREA), a geometric form factor, the emissivity, and the Stefan-Boltzmann constant (SBC). For axisymmetric problems, the radiation area should be input on a "per radian" basis.

The emissivity may be constant or temperature (absolute) dependent. If it is constant, the value is input as a real constant. If it is temperature dependent, the values are input for the material property EMIS and the real constant value is used only to identify the material property number. In this case, the MAT value associated with the element is not used.

A summary of the radiation element input data is given in Table 11.31.1. A general description of element input, including the special features, is given in Chapter 10.

STIF31

Table 11.31.1 Input Data Summary

Element Name		STIF31
No. of Nodes	2	I,J
DOF per Node	1	TEMP
Real Constants	4	AREA, FORM FACTOR, EMISSIVITY, SBC (If EMISSIVITY = -N, use material N for emissivity vs. temperature evaluation)
Material Properties	1	EMIS (Required only if EMISSIVITY = -N)
Convection Surfaces	0	
Heat Generation Rates	0	
Special Features	Nonlinear	
KEYOPT(3)	0 - Use standard radiation equation 1 - Use empirical radiation equation	

Note - The Stefan-Boltzmann constant (SBC) defaults to $0.1190\text{E-}10$ with units of BTU, Hr, in, °R (or °F if TOFFST is used)

■ Theory

The standard radiation function is defined as follows:

$$q = \sigma \epsilon F A (T(I)^4 - T(J)^4)$$

where: σ = Stefan-Boltzmann Constant (SBC)
(defaults to 0.1190×10^{-10} BTU/Hr-in²-°R⁴)

ϵ = emissivity

F = geometric form factor

A = area (Length²)

q = heat flow rate (Heat/Time)

T = temperature

The nonlinear temperature equation is solved by an iterative solution based on the form:

$$\left[(T(I)^2 + T(J)^2)(T(I) + T(J)) \right]_p (T(I) - T(J))$$

where the $[]_p$ term is evaluated at the average temperatures of the previous two iterations.

An empirical radiation function of the following form may also be selected with the KEYOPT(3) key:

$$q = \sigma \epsilon (F T(I)^4 - A T(J)^4)$$

where F and A are arbitrary input constants.

STIF31

■ Assumptions and Restrictions

If the default Stefan-Boltzmann constant is used, the units associated with this element are BTU's, inches, hours and °R (or °F + TOFFST). Other data input for this analysis must be consistent with this set of units or an appropriate conversion factor should be included in the radiation element's real constants. Nodes may or may not be coincident. An iterative solution is required with this element.

■ Output Data

Output data for this element are written to the postdata file and/or listed in the output listing, as requested on the output control commands (*Load Option* module). Data written to the postdata file, known as postdata items, can be retrieved during postprocessing.

The following is a list of all output labels for this element and their definitions. See Section 10.2 for a general description of element output quantities.

User-Specified Quantities:

EL	Element Number
NODES	Node Connectivity
MAT	Material Number
AREA	Radiating Surface Area
EMIS	Emissivity
FORM FACTOR	Geometric Form Factor

ANSYS-Calculated Quantities:

TEMP	Nodal Temperature
HEAT RATE	Heat Flow Rate (positive from node I to node J)

□ Postdata

Each postdata item is identified by a unique number on the postdata file. The following table shows the postdata associated with this element.

Table 11.31.2 STIF31 Postdata Item Numbers

	Centroid	Node	
		I	J
EMIS		1	2
TEMP		5	6
HEAT RATE	3		
AREA	4		
FORM FACTOR	7		

STIF31

□ Output Listing

The following table shows how the data are arranged on the element output listing.

Table 11.31.3 STIF31 Element Output Listing

Line 1	— EL, NODES, MAT, AREA
Line 2	— EMIS(I, J), TEMP (I, J), HEAT RATE

STIF32

11.32 2-D Heat Conducting Bar

The two-dimensional conducting bar is a uniaxial element with the ability to conduct heat between its nodes. The element has a single degree of freedom (temperature) at each node. The conducting bar is applicable to two-dimensional (plane and axisymmetric) thermal analyses, both steady-state and transient.

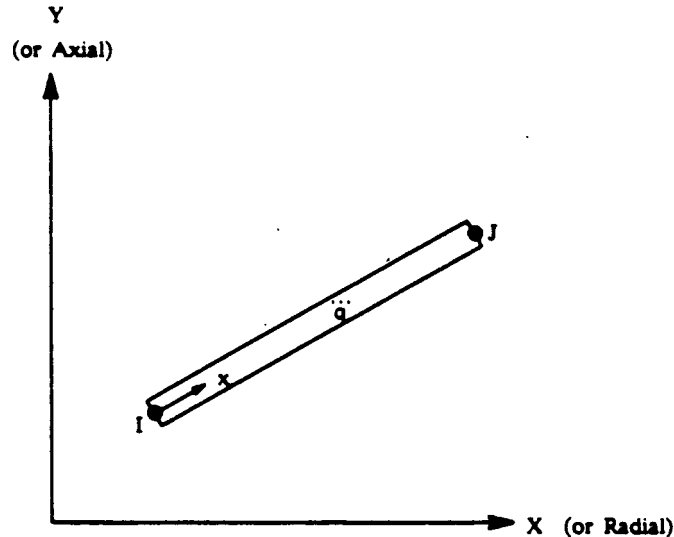


Figure 11.32.1 Two-Dimensional Heat Conducting Bar

■ Input Data

The geometry, node locations, loading, and coordinate system for the conducting bar element are shown in Figure 11.32.1. The element is defined by two nodes, a cross-sectional area (AREA), and material properties. For an axisymmetric analysis, the area must be defined on a "per radian" basis. The specific heat (C) and the density (DENS) may be assigned any values for steady-state solutions.

The thermal conductivity (KXX) is in the element longitudinal direction. An average internal heat generation rate may be applied to the element. A summary of the bar element input data is given in Table 11.32.1. A general description of element input is given in Chapter 10.

STIF32

Table 11.32.1 Input Data Summary

Element Name		STIF32
No. of Nodes	2	I,J
DOF per Node	1	TEMP
Real Constants	1	AREA
Material Properties	3	KXX, DENS, C
Convection Surfaces	0	
Heat Generation Rate	1	Average

■ Theory

The temperature distribution for this element is obtained from the numerical solution of the following equation:

$$\rho C_p \frac{\partial T}{\partial t} = \frac{\partial}{\partial x} \left(k_x \frac{\partial T}{\partial x} \right) + \ddot{q}$$

where:

- k_x = thermal conductivity (Heat/Length-Time-Deg)
- ρ = density (Weight/Volume or Mass/Volume)
- C_p = specific heat (Heat/Weight-Deg or Heat/Mass-Deg)
- $\ddot{q}$ = internal heat generation rate (Heat/Volume-Time)
- T = temperature

The temperature function is a linear polynomial of the form:

$$T(x) = C_1 + C_2 x$$

where the x-axis extends from node I to node J.

■ Assumptions and Restrictions

Heat is assumed to flow only in the longitudinal element direction. The element must be in an X-Y plane and the global X-axis must be the radial direction for axisymmetric problems. The element must not have a zero length, so nodes I and J must not be coincident. A free end of the element (i.e., not connected to another element and not subjected to a boundary constraint) is assumed to be adiabatic.

A conducting bar can be modeled with one element or many elements. The user must use more than one element if temperature results are required at intermediate points along the bar. The length of the element does affect the integration time step.

STIF32

■ Output Data

Output data for this element are written to the postdata file and/or listed in the output listing, as requested on the output control commands (*Load Option* module). Data written to the postdata file, known as postdata items, can be retrieved during postprocessing.

The following is a list of all output labels for this element and their definitions. See Section 10.2 for a general description of element output quantities.

User-Specified Quantities:

EL	Element Number
NODES	Node Connectivity
AREA	Cross-Sectional Area

ANSYS-Calculated Quantities:

LENGTH	Element Length
TEMP	Nodal Temperature
HEAT RATE	Heat Flow Rate (positive from node I to node J)

□ Postdata

Each postdata item is identified by a unique number on the postdata file. The following table shows the postdata associated with this element.

Table 11.32.2 STIF32 Postdata Item Numbers

	Centroid	Node	
		I	J
HEAT RATE	1		
LENGTH	2		
AREA	3		
TEMP		4	5

□ Output Listing

The following table shows how the data are arranged on the element output listing.

Table 11.32.3 STIF32 Element Output Listing

Line 1	— EL, NODES, LENGTH, AREA, TEMP (I, J), HEAT RATE

STIF34

11.34 Convection Link

The convection link is a uniaxial element with the ability to convect heat between its nodes. The element has a single degree of freedom (temperature) at each node. The convection element is applicable to both steady-state and transient thermal analyses in two (plane and axisymmetric) and three dimensions. A nonlinear convection coefficient can be defined which may also be a function of temperature or time.

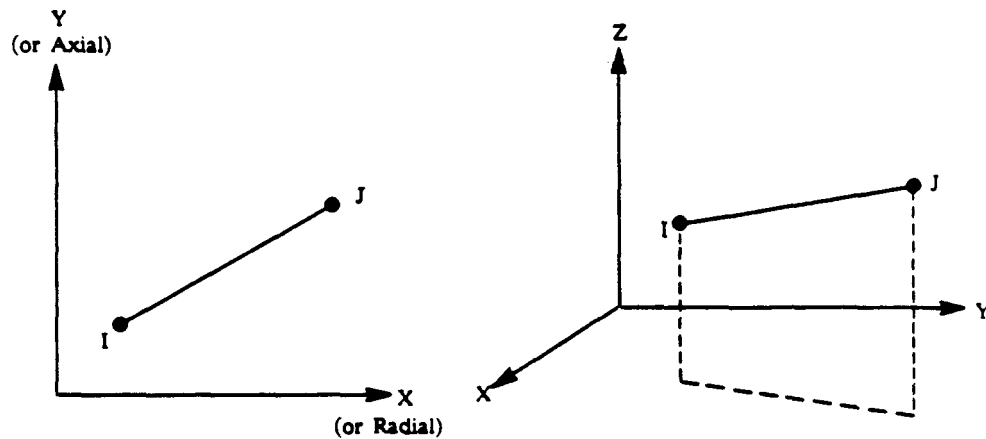


Figure 11.34.1 Convection Link

■ Input Data

The geometry, node locations, and the coordinate systems for the convection element are shown in Figure 11.34.1. The element is defined by two nodes, a convection surface area (AREA), two empirical terms, and a film coefficient (HF). In an axisymmetric analysis, the convection area must be expressed on a "per radian" basis. The empirical terms *n* (input as EN) and CC determine the form of the convection equation described later in this section.

A special option obtained with KEYOPT(3)=2 allows an alternate input for HF and a scale factor (F) input. The alternate input uses the *HCOEF* and *TBULK* fields on the CV (or EC) command for the HF and F values, respectively. For this option, if *HCOEF* is greater than zero, the HF material property values input in the material table are ignored. If no F value is input, a value of 1.0 is assumed. If only F is input on the CV command (that is, *HCOEF* is left blank), the HF material property from the material table is used. Note, values used in the *TBULK* field will ramp within a load step if KBC=0 and |NITITER|>1. A CV command must be included (even if the *HCOEF* and *TBULK* fields are left blank) for all STIF34 elements having KEYOPT(3)=2. KEYOPT(3) defines the procedure for evaluating a temperature dependent film coefficient.

A summary of the convection element input data is given in Table 11.34.1. A general description of element input, including the special features, is given in Chapter 10.

STIF34

Table 11.34.1 Input Data Summary

Element Name		STIF34
No. of Nodes	2	I,J
DOF per Node	1	TEMP
Real Constants	3	AREA, EN, CC
Material Properties	1	HF
Convection Faces	0	
Heat Generation Rates	0	
Special Features	Nonlinear if EN $\neq$ 0.0 or KEYOPT(3) = 3	
KEYOPT(2)*	0 - Use average of T(I) and T(J) to evaluate HF 1 - Use greater of T(I) and T(J) for evaluation 2 - Use lesser of T(I) and T(J) for evaluation 3 - Use differential (T(I)-T(J)) for evaluation	
KEYOPT(3)	0 - Standard element input and empirical term 2 - Use alternate input for HF and F (input at CV command (Load Module)) 3 - Use discontinuous empirical term	

* Only applicable if HF is a function of temperature (T).

■ Theory

The convection function is defined as follows:

$$q = h_f AE(T(I) - T(J))$$

where:

- q = heat flow rate (Heat/Time)
- h_f = film coefficient (Heat/Length²-Time-Deg)
- A = area (Length²)
- T = temperature (this iteration) (Deg)
- E = empirical convection term = $F|T_p(I) - T_p(J)|^n + CC/h_f$

(Note, E=F if n and CC=0.0. F=1.0 unless KEYOPT(3)=2)

- T_p = temperature (previous iteration) (Deg)
- n = empirical coefficient (EN)
- CC = input constant

If KEYOPT(3)=3, the value of E depends upon the value of ΔT_p , i.e., E equals the larger of $|\Delta T_p|^n$ or CC/h_f , where $\Delta T_p = T_p(I) - T_p(J)$.

STIF34

■ Assumptions and Restrictions

If $T_p(I) = T_p(J)$ and n is nonzero, the first term of E is defined to be zero. Since all unspecified nodal temperatures are initially set to the uniform temperature, a nonzero value of n may result in no heat flowing through the element in the first iteration of a thermal solution.

If n is nonzero or KEYOPT(3)=3, the element is nonlinear and requires an iterative solution.

Nodes may be coincident. There are no restrictions on the length of the element. A zero length element is allowed.

■ Output Data

Output data for this element are written to the postdata file and/or listed in the output listing, as requested on the output control commands (*Load Option* module). Data written to the postdata file, known as postdata items, can be retrieved during postprocessing.

The following is a list of all output labels for this element and their definitions. See Section 10.2 for a general description of element output quantities.

User-Specified Quantities:

EL	Element Number
NODES	Node Connectivity
HFILM	Film Coefficient (input as HF)
AREA	Convection Surface Area

ANSYS-Calculated Quantities:

TEMP	Nodal Temperature
HEAT RATE	Heat Flow Rate (positive from node I to node J)

STIF34

□ **Postdata**

Each postdata item is identified by a unique number on the postdata file. The following table shows the postdata associated with this element.

Table 11.34.2 STIF34 Postdata Item Numbers

	Centroid	Node	
		I	J
HEAT RATE	1		
HFILM	2		
AREA	3		
TEMP		4	5

□ **Output Listing**

The following table shows how the data are arranged on the element output listing.

Table 11.34.3 STIF34 Element Output Listing

Line 1 — EL, NODES, HFILM, AREA, TEMP (I, J), HEAT RATE

STIF55

11.55 2-D Isoparametric Thermal Solid

This isoparametric thermal element can be used as a biaxial plane element or as an axisymmetric ring element with a two-dimensional thermal conduction capability. The element has four nodes with a single degree of freedom (temperature) at each node.

The element is applicable to two-dimensional (plane and axisymmetric) thermal analyses, both steady-state and transient. A similar element (STIF77), with mid-side node capability, is described in Section 11.77.

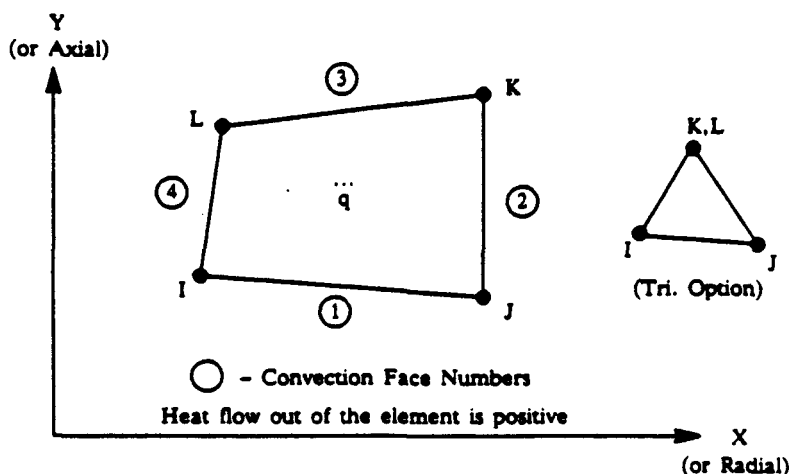


Figure 11.55.1 2-D Isoparametric Thermal Solid

■ Input Data

The geometry, node locations, face numbers, loading, and the coordinate system for the element are shown in Figure 11.55.1. The element is defined by four nodes and the material properties. The thermal conductivities (KXX, KYY) are defined in the global X and Y directions. The specific heat (C) and density (DENS) are used only in transient solutions. An average internal heat generation rate may be applied to the element. All of the element lateral faces have convection capability and are numbered as shown in Figure 11.55.1.

KEYOPT(1) defines the procedure for evaluating a temperature-dependent film coefficient. A summary of the element input data is given in Table 11.55.1. A general description of element input is given in Chapter 10.

STIF55

Table 11.55.1 Input Data Summary

Element Name		STIF55
No. of Nodes	4	I,J,K,L
DOF per Node	1	TEMP
Real Constants	0	
Material Properties	4	KXX, KYY, DENS, C (GLOBAL)
Convection Faces	4	IJ, JK, KL, LI
Heat Generation Rates	1	Average
KEYOPT(1)*	0	- Evaluate film coefficient (if any) at average film temperature, (TS + TB)/2
	1	- Evaluate at element surface temperature, TS
	2	- Evaluate at fluid bulk temperature, TB
	3	- Evaluate at differential temperature, TS - TB
KEYOPT(3)	0	- Plane
	1	- Axisymmetric

*only applicable if HF is a function of temperature.

■ Theory

The temperature distribution for this element is obtained from the numerical solution of the following equation (for the plane analysis):

$$\rho C_p \left(\frac{\partial T}{\partial t} \right) = \frac{\partial}{\partial x} \left(k_{xx} \frac{\partial T}{\partial x} \right) + \frac{\partial}{\partial y} \left(k_{yy} \frac{\partial T}{\partial y} \right) + \dots + \dot{q}$$

where:

- ρ = density (Weight/Volume or Mass/Volume)
- C_p = specific heat (Heat/Weight-Deg or Heat/Mass-Deg)
- k = thermal conductivity (Heat/Length-Time-Deg)
- $\dots$
- $\dot{q}$ = internal heat generation rate (Heat/Volume-Time)
- T = temperature

For the axisymmetric analysis the equation (for constant k values) is of the form:

$$\rho C_p \left(\frac{\partial T}{\partial t} \right) = k_{xx} \left(\frac{1}{x} \frac{\partial T}{\partial x} + \frac{\partial^2 T}{\partial x^2} \right) + k_{yy} \left(\frac{\partial^2 T}{\partial y^2} \right) + \dots + \dot{q}$$

where the X-axis is in the radial direction and the Y-axis is in the axial direction.

STIF55

The temperature functions used for the element are bilinear as described in Section 2.55 of the *ANSYS Theoretical Manual* (Ref. 5). A 3x3 lattice of integration points is used for the numerical (Gaussian) integration procedure.

■ Assumptions and Restrictions

The element must have a nonzero area. The element must lie in an X-Y plane as shown in Figure 11.55.1 and the X-axis must correspond to the radial direction for axisymmetric problems. Also, axisymmetric structures should be modeled in the +X quadrants. A triangular element may be formed by defining duplicate K and L node numbers as described in Chapter 10. The specific heat is evaluated at each integration point to allow for abrupt changes (such as melting) within a coarse grid of elements. A free surface of the element (i.e., not adjacent to another element and not subjected to a boundary constraint) is assumed to be adiabatic. Thermal transients having a fine integration time step and a severe thermal gradient at the surface also require a fine mesh at the surface.

■ Output Data

Output data for this element are written to the postdata file and/or listed in the output listing, as requested on the output control commands (*Load Option* module). Data written to the postdata file, known as postdata items, can be retrieved during postprocessing.

The following is a list of all output labels for this element and their definitions. For an axisymmetric analysis the face area and the heat flow rate are on a "per radian" basis. Heat flowing out of the element is considered to be positive. See Section 10.2 for a general description of element output quantities.

User-Specified Quantities:

EL	Element Number
NODES	Node Connectivity
MAT	Material Number
HFILM	Film Coefficient
TBULK	Fluid Bulk Temperature

STIF55

ANSYS-Calculated Quantities:

VOL	Element Volume
TGX,TGY	Thermal Gradient Components (at centroid)
TGSUM	Thermal Gradient Vector Sum (at centroid)
TFX,TFY	Thermal Flux (Heat Flow Rate/Cross-Sectional Area) Components (at centroid)
TFSUM	Thermal Flux Vector Sum (at centroid)
FACE	Convection Face Number
AREA	Convection Face Area
TAVG	Average Face Temperature
HEAT RATE	Heat Flow Rate (across face) (positive out of the element)

□ Postdata

Each postdata item is identified by a unique number on the postdata file. The following table shows the postdata associated with this element.

Table 11.55.2 STIF55 Postdata Item Numbers

	Centroid	Face			
		1 I-J	2 J-K	3 K-L	4 L-I
TGSUM	1				
TFSUM	2				
VOL	3				
TGX	43				
TGY	44				
TFX	45				
TFY	46				
FACE NO.		4	12	20	28
AREA *		5	13	21	29
FACE NODES*		6,7	14,15	22,23	30,31
HPFILM *		8	16	24	32
TAVG *		9	17	25	33
TBULK *		10	18	26	34
HEAT RATE*		11	19	27	35

*Values for these items are zero unless convection is present.

STIF55

□ Output Listing

The following table shows how the data are arranged on the element output listing. Nodal heat flows are automatically listed.

Table 11.55.3 STIF55 Element Output Listing

Line 1	—	EL, NODES, MAT, VOL
Line 2	—	TGX, TGY, TGSUM, TFX, TFY, TFSUM
Line 3*	—	FACE, FACE NODES, AREA, HFILM, TAVG, TBULK, HEAT RATE
Lines 4-9*	—	(Same as line 3 for other convection faces, if defined)

*Listed only if convection is defined

3.7 REFERENCES

1. Cask Systems Development, Initiative 1, Statement Of Work For NuPac 100-Ton Rail/Barge Cask Development, Contract No. DE-AC07-88ID12700, May 10, 1988.
2. NuPac Report No. L-9775, "O-ring Seal Test Program Report", Prepared by, Nuclear Packaging Inc., Federal Way, WA, July 1989.
3. Bulletin S-83N, "Boro-Silicone Shielding", Prepared by, Reactor Experiments, Inc., San Carlos, CA, October 1, 1987. (Include update on thermal expansion and conductivity per telecon October 19, 1988)
4. NuPac Report No. IL-001-NP, "Impact Limiter Polyurethane Foam High Temperature Structural Performance Test Results", Prepared by, Nuclear Packaging Inc., Federal Way, WA, July 19, 1989.
5. "Safety Analysis Report For The NuPac 125-B Fuel Shipping Cask", Docket Number 71-9200, Prepared by Nuclear Packaging Inc., Federal Way, WA, May, 1985.
6. ORNL/CSD/TM-149 TTC-9316, " Scoping Design Analyses for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, or 10-Year-Old PWR Spent Fuel", by J. A. Bucholz, Jan, 1983.
7. Unpublished data sheet transmittal from George Townes of BEI to Dan Kent of NuPac, dated April 3, 1989.
8. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1. 1986 Edition
9. ORNL-NSIC-68, "Cask Designers Guide", by L.B. Shappert, Feb, 1970.
10. SAE Aero-Space Applied Thermodynamics Manual, Section 2

11. "Metals For Supersonic Aircraft And Missiles", American Society of Metals, 1957, Transactions of ASME, Vol 77, October, 1957
12. "Principles Of Heat Transfer", Third Edition, by Frank Kreith, University of Colorado, 1976.
13. EPRI NP-5268, " The MC-10 PWR Spent- Fuel Storage Cask: Testing and Analysis ", Prepared by PNL, VPC, & EG&G Idaho, July, 1987
14. 10 CFR Part 71, 1983
15. "Handbook of Heat Transfer", by Rohsenow and Hartnet, McGraw Hill, 1976.
16. Roark and Young, "Formulas for Stress and Strain", McGraw Hill Fifth Edition.
17. "Developmental Testing of Partially Volatile Neutron Shields For High-Performance Shipping Cask", by R.B. Pope, G.C. Allen, H.J. Rack, and B.J. Joseph, of Sandia National Labs and S.A. Dupree, of Los Alamos Tech. Assoc., paper presented at PATRAM'80, Nov 10-14, 1980.
18. "Safety Analysis Report For The MAC Legal Weight Truck Cask", Docket Number 71-9225, Prepared by Nuclear Assurance Inc., Norcross, GA 30092, March, 1988.
19. Domestic Light Water Reactor Fuel Design Evolution, Prepared by Dept. of Energy, Savannah River Operations Office, Sept, 1981.
20. "ANSYS-PC\THERMAL 4.3" Reference Manual (for Revision 4.3A4), Swanson Analysis System, Inc. Houston, PA, Update 0-2 December 1, 1988.

4.0 CONTAINMENT

4.1 Containment Boundary

The containment boundary consists of the fuel as the primary level of containment and the cask as the secondary level of containment. The cask containment consists of a stainless steel vessel with a mating lid and body, plus the inner main O-ring seal enclosed between lid and body. In addition, the containment boundary includes a stainless steel vent port plug and a stainless steel drain plug each with an O-ring seal. There is also a rupture disc located in the lid.

4.1.1 Containment Vessel

The cask containment boundary consists of the 1.125 inch thick inner shell, the 8.5 inch thick cask bottom, the 16.25 inch thick closure-end forging, and the 8.5 inch thick cask lid. The containment boundary of the lid includes the inner O-ring seal, a vent port with a stainless steel plug and an O-ring seal, a rupture disc port with a nickel rupture disc, and a stainless steel plate which is welded to the inside of the lid to seal the opening that forms the seal test port in the lid. The containment boundary of the body includes a drain port in the bottom with a stainless steel plug and an O-ring seal.

A more detailed description of the cask containment boundary is provided in Section 1.2.1.1 and on the Packaging General Arrangement Drawing (NuPac DWG #2111-201) in Appendix 1.3.2.

The non-stainless steel components utilized in the containment boundary are the inner main O-ring seal, the inner O-ring seals on the vent and drain port plugs, and the nickel rupture disc.

4.1.2 Containment Penetrations

The only containment boundary penetrations into the containment vessel are the vent and drain ports, the rupture disc port, and the lid. Each penetration is designed such that a total system leakage-rate shall not exceed 1×10^{-7} standard cubic centimeters per second (scc/sec), so defined as 'leaktight' per ANSI 14.5-1987, Reference 4.5.1.1.

4.1.3 Seals and Welds

4.1.3.1 Seals

Seals affecting containment are described in section 4.1.1 above. A summary of seal testing prior to first use, during routine maintenance, and upon assembly for transport is as follows.

4.1.3.1.1 Fabrication Verification Leak Test

During fabrication following the pressure testing per Section 8.1.2.2, the containment vessel shall be tested per the Fabrication Verification Leak Test, delineated in Section 8.1.3. The Fabrication Verification Leak Test is consistent with the requirements of Section 6.3 of Reference 4.5.1.1. This test verifies the containment integrity of the NuPac 140-B Cask to a leakage rate not to exceed 1×10^{-7} scc/sec.

4.1.3.1.2 Maintenance Verification Leak Test

Annually, or after repair of the sealing surface, the containment boundary shall be leak tested per the Maintenance Verification Leak Test, delineated in Section 8.2.2. The Maintenance Verification Leak Test is consistent with the requirements of Section 6.4 of Reference

4.5.1.1. This test verifies the sealing integrity of the NuPac 140-B Cask to a leakage rate not to exceed 1×10^{-7} scc/sec.

4.1.3.1.3 Assembly Verification Leak Test

Prior to transport of the loaded Nupac 140-B Cask, the main body O-ring seal and the vent and drain port plug O-ring seals shall be leak tested per the Assembly Verification Leak Test, delineated in Appendix 7.4.2. The Assembly Verification Leak Test is consistent with the requirements of Section 6.5 of Reference 4.5.1.1. This test verifies the proper assembly of the Cask with a test sensitivity of 1×10^{-3} scc/sec, or better.

4.1.3.2 Welds

All containment vessel body welds are full penetration welds and non-destructively tested to ensure structural integrity. All containment boundary welds are confirmed to be leaktight per the Fabrication Verification Leak Test in Section 8.1.3.

4.1.4 Closure

Closure on the NuPac 140-B Cask is affected by thirty-two (32), ASTM A-320, Grade L43, 1 5/8 - 8 UN bolts tightened to 1200-1450 ft-lbs torque each.

4.2 Requirements for Normal Conditions of Transport

The results of the structural and thermal analyses described in Sections 2.0 and 3.0, respectively, verify that there will be no release of radioactive materials per the 'leaktight' criterion of ANSI N14.5-1987 (Reference 4.5.1.1) under any of the normal conditions of transport.

4.2.1 Containment of Radioactive Material

The 'leaktight' containment criteria of ANSI N14.5-1987, Reference 4.5.1.1, specified for the structure and seals of the NuPac 140-B cask ensure that the requirements of 10 CFR 71.51, Reference 4.5.1.2, are satisfied under the normal conditions of transport.

4.2.2 Pressurization of Containment Vessel

The maximum normal operating pressure of the NuPac 140-B Cask is 284 psig. Results from the analyses presented in Section 3.4.4 verify that any increase in pressure, due to an increase in temperature of the initial cask inert gas atmosphere or due to the water vapor pressure from the water assumed present in the cask cavity, will be equal to or less than 284 psig within the containment vessel. A pressure increase to 284 psig will not reduce the effectiveness of the NuPac 140-B Cask to maintain containment integrity per Section 4.2.1. In the unlikely event of an overpressurization, a rupture disk would prevent a catastrophic failure of the cask body structure.

4.2.3 Containment Criterion

The containment boundary shall be leak tested as described in Sections 4.1.3.1.1, 4.1.3.1.2, and 4.1.3.1.3 to demonstrate the containment criterion of Section 4.2.1, at the completion of

fabrication, for annual maintenance, or for assembly verification prior to transport, respectively.

4.3 Containment Requirements for the Hypothetical Accident Conditions

The results of the structural and thermal analyses performed in Sections 2.0 and 3.0, respectively, and the results of the full-scale, structural and thermal testing (to be performed later), verify that there will be no release of radioactive materials per the 'leaktight' criterion of ANSI N14.5-1987 (Reference 4.5.1.1) under any of the hypothetical accident conditions of transport.

4.3.1 Fission Gas Products

The fission gas products in the NuPac 140-B Cask payload will be retained under the hypothetical accident conditions with a leakage rate less than 1×10^{-7} scc/sec due to the 'leaktight' design of cask structure and seals.

These estimates are based on total release of all fission gas products from the 21 PWR spent fuel assemblies. The assemblies are assumed to have an initial enrichment of 3.2 percent, a burnup of 35,000 MWd/MTU and a cooling decay time of 10 years.

4.3.2 Containment of Radioactive Material

The 'leaktight' containment criteria of ANSI N14.5-1987, Reference 4.5.1.1, specified for the structure and seals of the NuPac 140-B Cask ensure that the requirements of 10 CFR 71.51, Reference 4.5.1.2, are satisfied under the hypothetical accident conditions.

4.3.3 Containment Criterion

The NuPac 140-B Cask has been designed to meet the 'leaktight' criteria established in ANSI N14.5-1987, Reference 4.5.1.1, which is verified after fabrication, and before and during use by leak testing.

The results of the structural and thermal analyses described in Sections 2.0 and 3.0, respectively, verify that the NuPac 140-B Cask will meet the 'leaktight' criterion of ANSI N14.5-1987, Reference 4.5.1.1, for the entire hypothetical accident scenario.

4.4 Special Requirements

The NuPac 140-B Cask is designed as a Type B(M) containment package to meet the requirements of 10 CFR 71.63, Reference 4.5.1.2, for plutonium shipments that are in the form of reactor fuel elements.

4.5 Appendix

4.5.1 References

APPENDIX 4.5.1

REFERENCES

4.5.1 References

4.5.1.1 ANSI N14.5-1987, American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment.

4.5.1.2 Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), Packaging and Transportation of Radioactive Materials, August, 24, 1983.

5.0 SHIELDING EVALUATION

The NuPac 140-B Cask was designed to accommodate 21 intact PWR fuel assemblies with a maximum burnup of 35,000 megawatt days per metric ton of uranium (MWD/MTU) or 52 BWR fuel assemblies with a burnup of 30,000 MWD/MTU. The reference fuel age for cask design is 10 years. The cask provides adequate shielding to ensure compliance with the external dose rate requirements of 10 CFR 71.47, and 49 CFR 173.441, for the parameters stated above.

5.1 Discussion and Results

Gamma shielding is provided by the combination of a lead liner in the outer containment vessel, and the stainless steel inner and outer shells which make up the cask wall. There is also a thick layer of Boro-Silicone for neutron thermalization and attenuation. Stainless steel in the package ends provides gamma-ray shielding. A Boro-Silicone insert in the stainless steel ends attenuates neutrons for ALARA consideration during installation of the impact limiters. The impact limiters provide virtually 100 percent neutron attenuation and some gamma shielding during transport.

Normal and hypothetical accident condition shielding requirements for the package are determined by the following criteria:

Normal Condition: Maximum dose rate of 200 millirem/hour at any point in contact with the package, and 10 millirem/hour at any point two (2) meters from the package are the requirements of 10 CFR 71.47(a) and (c) respectively, for dose design-basis normal conditions.

Accident Condition: Maximum dose rate of 1.0 rem/hour at any point one (1) meter from the cask is a requirement of 10 CFR 71.51(a) (2) for dose design-basis accident conditions.

The normal condition limits bound the design basis shielding requirement for the package. Only slight changes in the shielding geometry occur during an accident condition. One notable distortion could occur in the flat end drop. Flat end drop analysis of Section 2.0, Table 2.6.7.1-6 resulted in a maximum lead slump of 1.67 inches. This slump occurs at the cask ends, at a minimum of three inches from the active fuel region. The expected dose rate from streaming through this void during a hypothetical accident is less than the dose limit of 10 CFR 71.51 (a)(2).

The only other potential damage could occur during the 40-inch side drop onto a puncture pin. This scenario could result in local damage at the site of impact, and cause deformation in the lead and Boro-Silicone shields. The extent of deformation and the effects of streaming through the damaged shield will be examined during final design. Because the allowable dose rate is considerably higher for the accident condition, the normal transport conditions govern the shield thickness requirements.

The shielding configuration and dose points of interest are presented in Figure 5.1-1. Table 5.1-1 summarizes the dose rates at the indicated dose points for 21 PWR and 52 BWR fuel assemblies. The dose rate for the top (and not the bottom) of the package is reported since the package ends are symmetric. The fuel elements are positioned near the top of the cask cavity for axial dose rate calculations.

Although the cask can accommodate 2.5 times more BWR fuel elements than PWR fuel elements, the PWR source strength bounds the shielding requirements. The axial dose rate (case D in Figure 5.1-1) for 21 PWR fuel elements with 35,000 MWD/MTU and 10 years decay represents the maximum radioactivity capacity for the 140-B Cask with the current gamma shielding.

TABLE 5.1-1
Summary of Dose Rates at Given Dose Points

SUMMARY OF MAXIMUM DOSE RATES (mrem/hr)				
Dose Point:	<u>Axial Dose Rate</u>		<u>Radial Dose Rate</u>	
	surface(A)	2-meter(B)	surface(C)	2-meter(D)
Normal Conditions				
PWR FUEL				
Gamma	24.9	7.8	23.5	7.0
Neutron	0.2	0.2	8.4	2.4
Total	25.1	8.0	31.9	9.4
BWR FUEL				
Gamma	18.7	6.0	13.2	3.8
Neutron	0.3	0.3	3.1	0.8
Total	19.0	6.3	16.3	4.6
10 CFR 71 LIMIT	200.0	10.0	200.0	10.0
Hypothetical Accident Conditions			1-meter from package <u>side</u> <u>top</u>	
PWR neutron + gamma				
BWR neutron + gamma				
Total				
10 CFR 71 LIMIT	-----	----	1000.0	1000.0

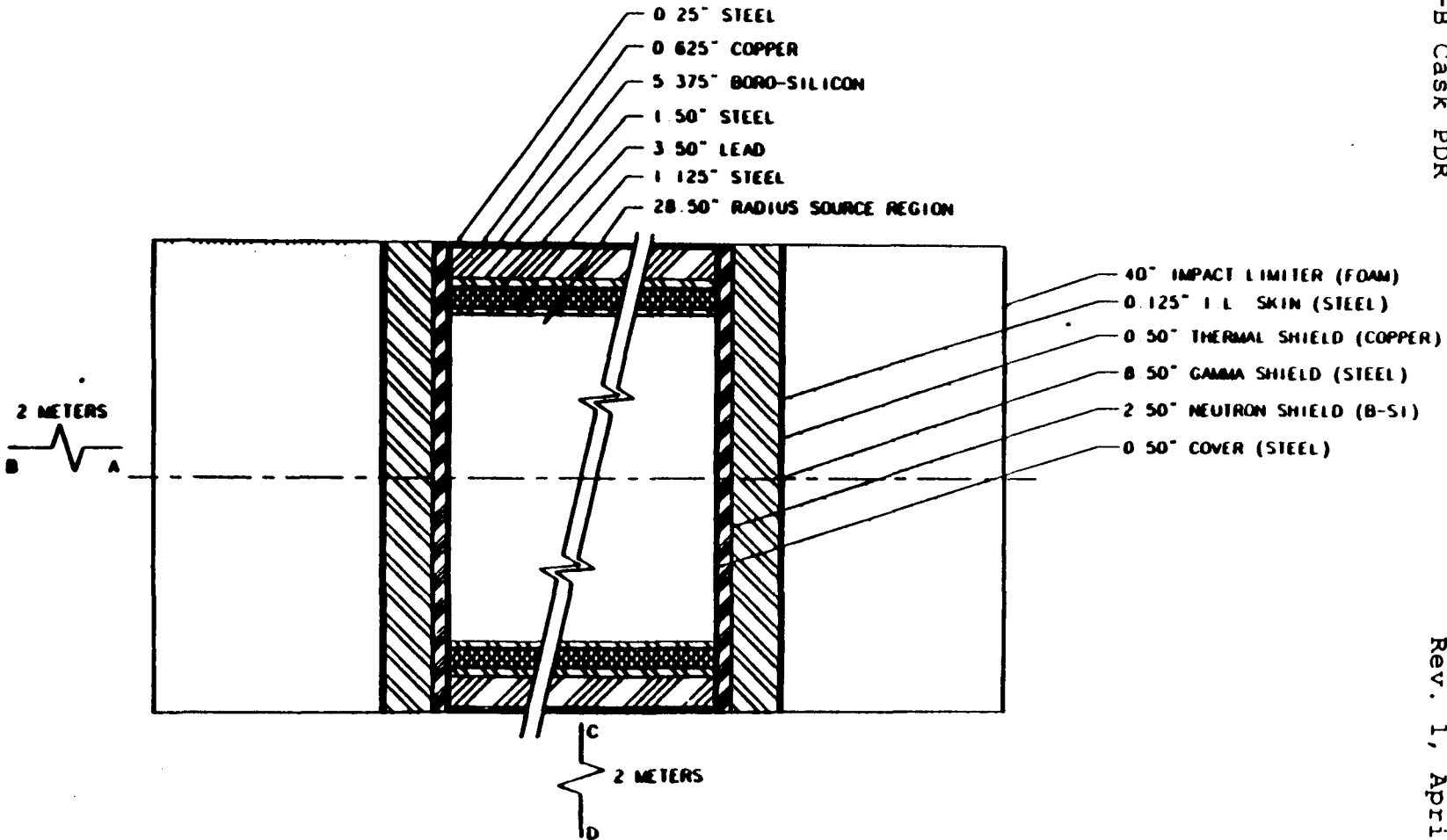


FIGURE 5.1-1

Shielding Configuration and Dose Points of Interest

5.2 Source Specification

The NuPac 140-B Cask and Fuel Package is designed to transport a variety of PWR and BWR fuel assemblies with differing levels of burnup and cooling times. There are many parameters that affect the activity present in a spent fuel assembly. The shielding analysis must consider the fuel assemblies power history, initial enrichment, exposure period, cooling time, heavy metal content and active fuel length.

The physical and radiological characteristics of the fuel elements were evaluated from information provided in DOE/RW-0184 (Reference 5.5.1.1). This report and the associated computerized database provide details about the physical and radiological properties of intact fuel assemblies and their associated hardware.

The database contains the radiological properties of intact spent fuel as a function of burnup and cooling time. The radiological data, generated by the computer code ORIGEN2, includes the inventory of fission, actinide and activation products in an irradiated fuel element.

The characteristics of the PWR and BWR fuel assemblies are sufficiently different that they will be considered separately. The cask can physically accommodate fuel elements with a maximum assembly length of 180 inches. The NuPac 140-B Cask can transport any PWR or BWR fuel assembly that falls within the design basis envelope. This could include shipping higher burnup fuels with an increased cooling time, or a decreased payload. These considerations require further analysis than is presented in this report.

For shielding calculations, the source region (the interior volume of the cask) is a homogenized mixture of the fuel and basket material. A conservative approach was used in determining the density of the source region for self-absorption within the fuel and basket material.

5.2.1 Gamma Source

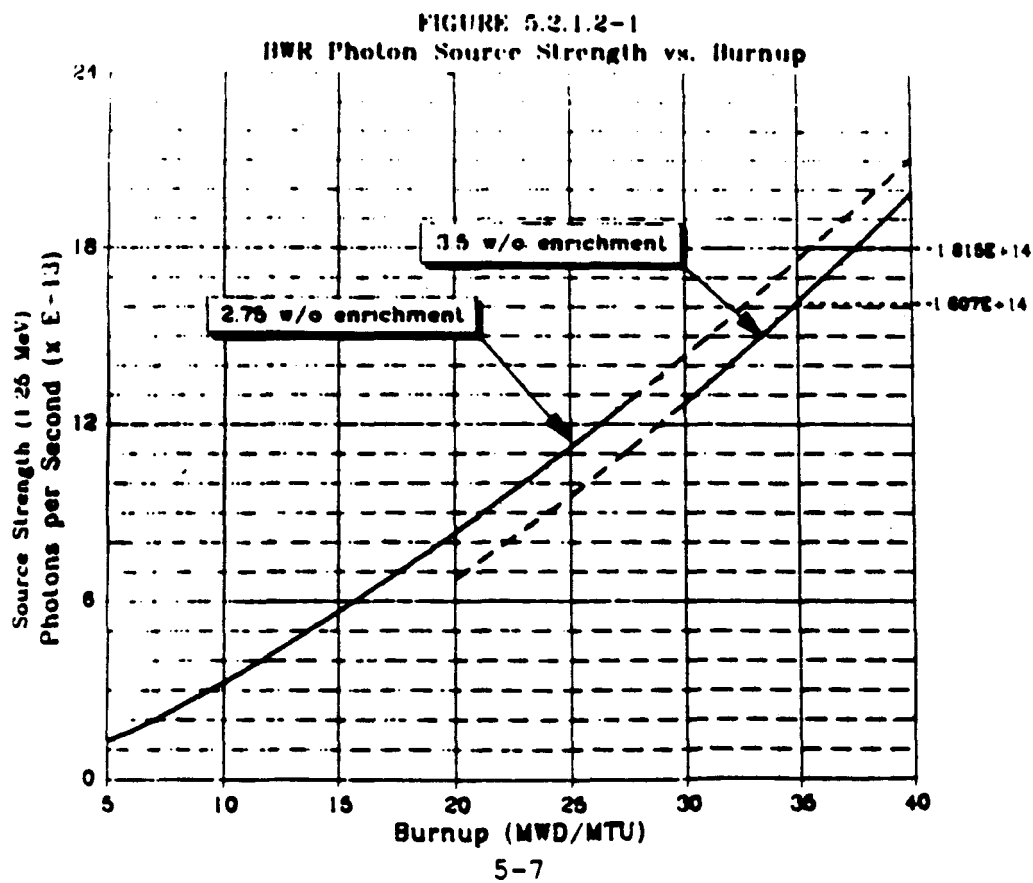
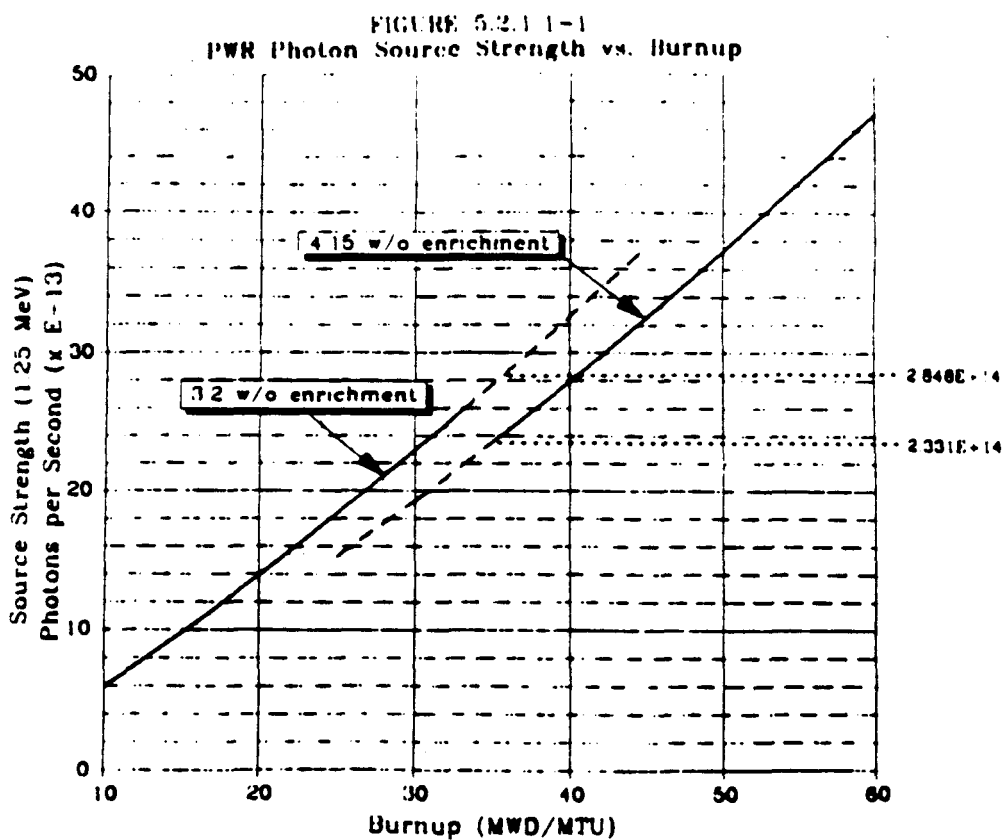
5.2.1.1 PWR Fuel

The input to ORIGEN2 for the PWR source strength determination was a generic 17 x 17 Westinghouse PWR fuel assembly with 35,000 MWD/MTU and 10 years cooling time. The input parameters to ORIGEN2 that established the radiological source activity are as follows:

Power Level	37.5 MW thermal/MTU
Operating Periods	11,000 MWD intervals
Shutdown Periods	106 days
Initial Enrichment	4.15 w/o U ²³⁵
Cooling Time	10 years

Eighty-eight percent of the photon absorbed dose results from photons with an average energy of 1.25 MeV. Reference 5.5.1.1 provides a photon spectra per one metric ton of initial heavy metal (MTIHM) for several PWR burnup periods between 5,000 and 60,000 MWD/MTU. Figure 5.2.1.1-1 is a plot of the 1.25 MeV photon emission rate verses burnup from Reference 5.5.1.1. The initial enrichment input to ORIGEN2 was 3.2 w/o U²³⁵ for burnups of 33,000 MWD/MTU and less, and 4.15 w/o U²³⁵ for burnups of 35,000 MWD/MTU and higher. The dotted line is an extrapolation of each curve.

The graph indicates that the lower enrichment fuel produces an 18 percent higher concentration of radioactivity than the higher enrichment fuel. The dose rate was increased by 25 percent to compensate for non-conservatism in the 35,000 MWD/MTU burnup case, and to account for the higher activity associated with 3.0 w/o enriched fuel.



The ORIGEN2 output divides the photon emission rate into 18 energy groups. Table 5.2.1.1-1, column A, contains the photon emission rate from ORIGEN2 for each energy group. For the shielding analysis, the physical characteristics of a Westinghouse 14 x 14 fuel assembly were used in calculations. This assembly is one of the lighter fuel elements and will yield a less dense source region. An active fuel length of 135.2 inches bounds the source region volume. The fuel region volume is $5.555 \times 10^6 \text{ cm}^3$ based on a basket diameter of 56.5 inches. The uranium content is 0.4614 MTIHM. A sample calculation to determine the photon fluence density, Φ_1 , for a single energy group with a 21 element basket configuration is shown below.

$$\begin{aligned} \Phi_1 \text{ (photons/cm}^2\text{-sec)} &= \text{(photons/sec-MTIHM)} \\ &\quad \times \text{(0.4614 MTIHM/assembly)} \\ &\quad \times \text{(21 assemblies/cask)} \\ &\quad \times \text{(cask/5.555 x 10}^6\text{ cm}^3\text{)} \end{aligned}$$

For example, consider photons with a mean energy of 3.50 MeV (from Table 5.2.1.1-1):

$$\begin{aligned} \Phi_1 &= (4.84 \times 10^6 \text{ photons/sec-MTIHM}) \times (0.4614 \text{ MTIHM/assembly}) \\ &\quad \times (21 \text{ assemblies/cask}) \times (\text{cask/5.555 x 10}^6 \text{ cm}^3) \\ \Phi_1 &= 844 \text{ photons/sec-cm}^2 \end{aligned}$$

Some of the energy groups were combined to coincide with the energy groups in the DLC-23/Cask cross-section library (Reference 5.5.1.2) available from the Radiation Shielding Information Center at ORNL. For example, photons with mean energies of 0.01, 0.025 and 0.0375 MeV were collapsed into Cask Group 40. This combination (shown in Table 5.2.1.1-1 as factor Φ_2) is necessary to apply the DLC-23F/Cask cross-section library. The library contains a 22 neutron, 18 gamma energy, P3 cross-section set specifically designed for the purpose of performing shielding calculations on spent fuel shipping casks.

TABLE 5.2.1.1-1
Photon Spectra For 35,000 MWD/MTU PWR Fuel
10 Year Fuel Decay
(For 0.4614 MTIHM/Assembly x 21 Assemblies)

Mean Energy MeV	gamma/sec/ MTIHM A	gamma/sec/ fuel assy B	gamma/sec/ cm ³ Φ_1	CASE-81 Group #	CASE-81 E Range MeV	gamma/sec/cm ³ for CASE-81 Φ_2
0.01	2.077E+15	9.583E+14	3.623E+09	40	0.005 to 0.05	5.319E+09
0.025	4.462E+14	2.059E+14	7.783E+08	39	0.05 to 0.10	1.154E+09
0.0375	5.260E+14	2.427E+14	9.175E+08	38	0.1 to 0.20	3.872E+08
0.0575	4.176E+14	1.927E+14	7.284E+08	37	0.2 to 0.30	3.525E+08
0.085	2.441E+14	1.126E+14	4.258E+08	36	0.3 to 0.40	1.687E+08
0.125	2.220E+14	1.024E+14	3.872E+08	35	0.4 to 0.60	5.997E+09
0.225	2.021E+14	9.325E+13	3.525E+08	34	0.6 to 0.80	0.0
0.375	9.673E+13	4.463E+13	1.687E+08	33	0.8 to 1.00	4.568E+08
0.575	3.438E+15	1.586E+15	5.997E+09	32	1.0 to 1.33	4.066E+08
0.85	2.619E+14	1.208E+14	4.568E+08	31	1.33 to 1.66	0.0
1.25	2.331E+14	1.076E+14	4.066E+08	30	1.66 to 2.0	5.315E+06
1.75	3.047E+12	1.406E+12	5.315E+06	29	2.0 to 2.5	1.110E+05
2.25	6.364E+10	2.936E+10	1.110E+05	28	2.5 to 3.0	7.040E+03
2.75	4.036E+09	1.862E+09	7.040E+03	27	3.0 to 4.0	8.442E+02
3.5	4.840E+08	2.233E+08	8.442E+02	26	4.0 to 5.0	9.825E+00
5	5.633E+06	2.599E+06	9.825E+00	25	5.0 to 6.5	0.0
7	6.492E+05	2.995E+05	1.132E+00	24	6.5 to 8.0	1.132E+00
9.5	7.456E+04	3.440E+04	1.301E-01	23	8.0 to 10.0	1.301E-01

TABLE 5.2.1.2-1
Photon Spectra For 30,000 MWD/MTU BWR Fuel
(For 0.1743 MTIHM/Assembly x 52 Assemblies)

Mean Energy MeV	gamma/sec/ MTIHM A	gamma/sec/ fuel assy B	gamma/sec/ cm ³ Φ_1	CASE-81 Group #	CASE-81 E Range MeV	gamma/sec/cm ³ for CASE-81 Φ_2
0.010	1.759E+15	3.07E+14	2.670E+09	40	0.005 - 0.05	3.920E+09
0.025	3.788E+14	6.60E+13	5.751E+08	39	0.05 - 0.10	8.472E+08
0.038	4.442E+14	7.74E+13	6.744E+08	38	0.10 - 0.20	2.810E+08
0.058	3.521E+14	6.14E+13	5.346E+08	37	0.20 - 0.30	2.601E+08
0.085	2.059E+14	3.59E+13	3.126E+08	36	0.30 - 0.40	1.248E+09
0.125	1.851E+14	3.23E+13	2.810E+08	35	0.40 - 0.60	4.381E+09
0.225	1.713E+14	2.99E+13	2.601E+08	34	0.60 - 0.80	0.0
0.375	8.222E+14	1.43E+14	1.248E+09	33	0.80 - 1.00	2.959E+08
0.575	2.886E+15	5.03E+14	4.381E+09	32	1.00 - 1.33	1.940E+08
0.85	1.949E+14	3.40E+13	2.959E+08	31	1.33 - 1.66	0.0
1.25	1.278E+14	2.23E+13	1.940E+08	30	1.66 - 2.00	3.755E+06
1.75	2.473E+12	4.31E+11	3.755E+06	29	2.00 - 2.50	6.823E+04
2.25	4.494E+10	7.83E+09	6.823E+04	28	2.50 - 3.00	4.398E+03
2.75	2.897E+09	5.05E+08	4.398E+03	27	3.00 - 4.00	5.361E+02
3.50	3.531E+08	6.15E+07	5.361E+02	26	4.00 - 5.00	5.989E+00
5.00	3.945E+06	6.88E+05	5.989E+00	25	5.00 - 6.50	0.0
7.00	4.546E+05	7.92E+04	6.902E-01	24	6.50 - 8.00	6.902E-01
9.50	5.221E+04	9.10E+03	7.926E-02	23	8.00 - 10.00	7.926E-02

5.2.1.2 BWR Fuel

The input to ORIGEN2 for the BWR source strength determination was a generic General Electric 8 x 8 BWR fuel assembly with 30,000 MWD/MTU and 10 years cooling time. The input parameters to ORIGEN2 that established the radiological source activity are as follows:

Power Level	25.9 MW Thermal/MT
Operating Periods	6,667 MWD
Initial Enrichment	3.5 w/o U ²³⁵
Shutdown Period	106 days
Cooling Time	10 years

For the BWR case, seventy-eight percent of the photon absorbed dose results from photons with an average energy of 1.25 MeV. The emission rate of 1.25 MeV photons verses burnup was also plotted (Figure 5.2.1.2-1) for the BWR fuels with burnups between 5,000 and 40,000 MWD/MTU. The enrichment was 2.75 w/o U²³⁵ for burnups of 27,500 MWD/MTU and less, and 3.50 w/o U²³⁵ for burnups of 30,000 MWD/MTU and higher. The graph shows a 14% higher concentration of radioactivity with 2.75 w/o enrichment and 30,000 MWD/MTU burnup. The contract requires a minimum burnup of 3.0 w/o enrichment. The calculated dose rates were increased by 14% for conservatism.

The BWR photon source strength is reported in the same manner as the PWR source strength. Table 5.2.1.2-1, column A contains the photon emission rate from Reference 5.5.1.1. The shielding analysis used the physical characteristics of Exxon / ANF 8 x 8 JP-3 BWR fuel in calculations. The volume of the source region is based on an active fuel length of 145.24 inches. The uranium content is 0.1743 MTIHM. The fluence density, Φ , was calculated using the same formula used in the PWR analysis.

5.2.2 Neutron Source

5.2.2.1 PWR Fuel

The same method used to determine the gamma source strength applies to the total neutron emission rate for low enrichment high burnup fuel. The effects of low and high enrichment verses burnup are plotted in Figure 5.2.2.1-1. The neutron emission rate is 78% higher for 3.2 w/o enriched fuel verses 4.15 w/o enriched fuel. The neutron dose rate was increased by a factor of 2 in this analysis.

As reported in Reference 5.5.1.1, the neutron emission rate for a PWR fuel assembly with 35,000 MWD/MTU burnup is:

$$1.331 \times 10^8 \text{ neutrons/sec-MTIHM}$$

For a heavy metal weight of 0.4614 MTIHM/assembly, the neutron fluence density, Φ_{nPWR} is:

$$\begin{aligned}\Phi_{\text{nPWR}} &= (1.331 \times 10^8 \text{ neutrons/sec-MTIHM}) \\ &\quad \times (0.4614 \text{ MTIHM/assembly}) \times (21 \text{ assembly/cask}) \\ &\quad \times (\text{cask}/5.555 \times 10^6 \text{ cm}^3) \\ \Phi_{\text{nPWR}} &= 232.2 \text{ neutrons/sec-cm}^3\end{aligned}$$

A PWR spent fuel assembly has a fission spectrum similar to the fission spectrum of Cf^{252} (Reference 5.5.1.2). The fission spectrum of Cf^{252} was used for the PWR energy distribution. Table 5.2.2.1-1 provides the neutron fluence density for 35,000 MWD/MTU PWR fuel.

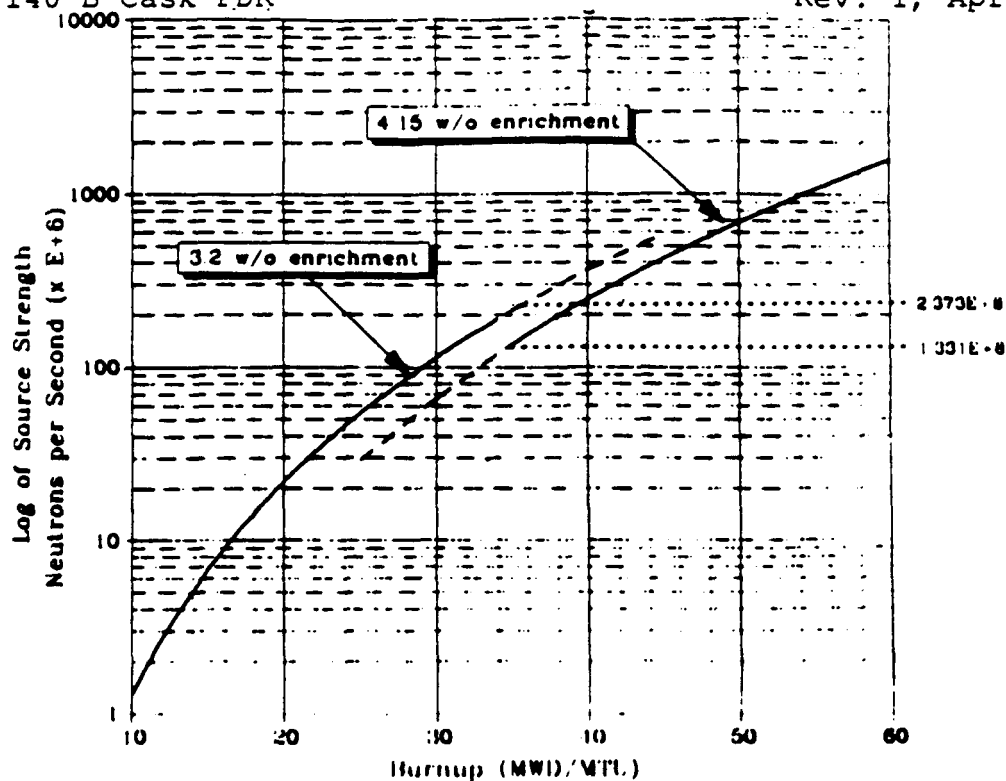


FIGURE 5.2.2.1-1

PWR Neutron Source Strength vs. Burnup

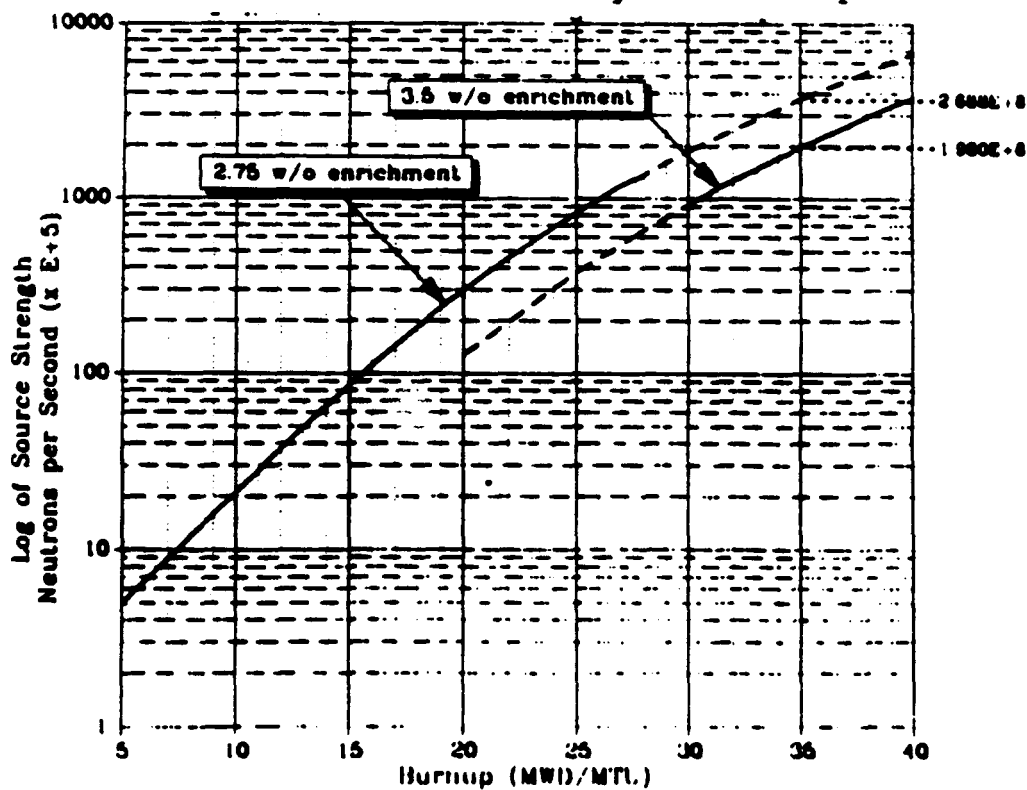


FIGURE 5.2.2.2-1

BWR Neutron Source Strength vs. Burnup

TABLE 5.2.2.1-1
PWR Neutron Energy Distribution

CASE-81 Group #	Energy Range (MeV)	Fraction $\Delta F(E)$	neutrons/ cm^3/sec
1	12.2 to 14.92	4.653E-04	0.437
2	10.0 to 12.2	1.883E-03	1.337
3	8.18 to 10.0	5.756E-03	4.468
4	6.36 to 8.18	1.924E-02	9.288
5	4.96 to 6.36	4.000E-02	12.014
6	4.06 to 4.96	5.174E-02	25.403
7	3.01 to 4.06	1.094E-01	20.443
8	2.46 to 3.01	8.804E-02	4.848
9	2.35 to 2.46	2.088E-02	26.842
10	1.83 to 2.35	1.156E-01	48.507
11	1.11 to 1.83	2.089E-01	44.582
12	0.55 to 1.11	1.920E-01	30.813
13	0.11 to 0.55	1.327E-01	3.123
14	0.003 to 0.11	1.345E-02	0.0
15	5.83E-4 to 0.003	0.0	0.0
16	1.01E-4 to 5.83E-4	0.0	0.0
17	2.90E-5 to 1.01E-4	0.0	0.0
18	1.01E-5 to 2.9E-5	0.0	0.0
19	3.06E-6 to 1.01E-5	0.0	0.0
20	1.12E-6 to 3.06E-6	0.0	0.0
21	4.14E-7 to 1.12E-6	0.0	0.0
22	1.00E-8 to 4.14E-7	0.0	0.0
sum		1.0	232.2

TABLE 5.2.2.2-1
BWR Neutron Energy Distribution

CASE-81 Group #	Energy Range (MeV)	Fraction $\Delta F(E)$	neutrons/ cm^3/sec
1	12.2 to 14.92	4.653E-04	0.061
2	10.0 to 12.2	1.883E-03	0.247
3	8.18 to 10.0	5.756E-03	0.754
4	6.36 to 8.18	1.924E-02	2.519
5	4.96 to 6.36	4.000E-02	5.238
6	4.06 to 4.96	5.174E-02	6.775
7	3.01 to 4.06	1.094E-01	14.326
8	2.46 to 3.01	8.804E-02	11.529
9	2.35 to 2.46	2.088E-02	2.734
10	1.83 to 2.35	1.156E-01	15.138
11	1.11 to 1.83	2.089E-01	27.355
12	0.55 to 1.11	1.920E-01	25.142
13	0.11 to 0.55	1.327E-01	17.377
14	0.003 to 0.11	1.345E-02	1.761
15	5.83E-4 to 0.003	0.0	0.0
16	1.01E-4 to 5.83E-4	0.0	0.0
17	2.90E-5 to 1.01E-4	0.0	0.0
18	1.01E-5 to 2.9E-5	0.0	0.0
19	3.06E-6 to 1.01E-5	0.0	0.0
20	1.12E-6 to 3.06E-6	0.0	0.0
21	4.14E-7 to 1.12E-6	0.0	0.0
22	1.00E-8 to 4.14E-7	0.0	0.0
sum		1.0	130.937

5.2.2.2 BWR Neutron Source Strength

The same method used to calculate PWR neutron fluence density applies to BWR fuels. Figure 5.2.2.2-1 is a plot of the total neutron emission rate verses burnup. The neutron emission rate is 36% higher for 2.75 w/o enriched fuel verses 3.5 w/o enriched fuel. The neutron source strength was increased by 36% for conservatism in the BWR analysis. The Cf²⁵² spectrum was again used for the BWR neutron energy distribution.

The BWR neutron emission rate, Φ_{nBWR} for 30,000 MWD/MTU is:

$$\begin{aligned}\Phi_{nBWR} &= (9.345 \times 10^7 \text{ neutrons/sec-MTIHM}) \\ &\quad \times (0.1743 \text{ MTIHM/assembly}) \times (52 \text{ assembly/cask}) \\ &\quad \times (\text{cask}/5.97 \times 10^6 \text{ cm}^3) \\ \Phi_{nBWR} &= 141.9 \text{ neutrons/sec-cm}^3\end{aligned}$$

The BWR neutron fluence density is presented in Table 5.2.2.2-1.

5.3 Model Specification

The neutron shielding requirements for the NuPac 140-B Cask were determined using the computer code ANISN/PC (Reference 5.5.1.3). The gamma shielding requirements were determined with the computer codes ANISN/PC and QAD-CGGP (Reference 5.5.1.4). The application of these two codes is described in more detail below.

ANISN/PC

ANISN/PC is a one-dimensional discrete ordinates finite difference code widely used for deep penetration shielding problems. The source geometry can be either cylindrical, slab or spherical. Radial shielding calculations are based on cylindrical geometry. The radial dose rate from neutrons and gammas was determined with ANISN/PC. Using slab geometry, the ANISN/PC code calculated the axial dose rate from neutrons, and gammas from the (n,γ) reaction. The ANISN input files are included in Appendix 5.5.2.

The source and shield dimensions in ANISN/PC are input as zones divided into mesh intervals. A particle flux is determined at each mesh interval for each neutron and gamma energy group. The dose rate at each mesh interval is included as Appendix 5.5.3. The DLC-23/Cask (Reference 5.5.1.2) 22 neutron, 18 gamma-ray group cross-section library was input to ANISN/PC for the cask materials. These cross-sections were compiled for the purpose of performing calculations on spent fuel shipping casks.

QAD-CGGP

QAD-CGGP is a point-kernel shielding code that uses three dimensional geometry to define the source and shielding configurations. The code allows the combination of geometric shapes by "and/or" operators into zones which can then be assigned an amalgamation of elemental densities. QAD-CGGP was used to determine the gamma dose rate in the axial direction, and to verify the ANISN/PC gamma results. The buildup factor was determined by the geometric progression (G-P) fitting function

(Reference 5.5.1.4) with buildup in steel. The QAD-CGGP input files are included in Appendix 5.5.4.

ANSI standard flux-to-dose conversion factors corresponding to the coupled (22n-18γ) DLC-23/CASK Cross-Section Library were applied to obtain dose rates from ANISN/PC. These conversion factors are not applicable to QAD-CGGP fluxes. The ANSI standard values are based on a known spectrum of gamma-rays impinging upon a tissue phantom, while a point kernel calculation is based on knowledge of the source spectrum (rather than the impinging spectrum). The formula below was used to determine the flux-to-dose conversion factors, K, for QAD-CGGP. Mass energy absorption coefficients (μ_a/ρ) values were selected from Reference 5.5.1.5, and are based on ICRP¹ tissue.

$$K = (1.6E-6 \text{ ergs/MeV}) \times (1 \text{ rad}/(100 \text{ ergs/grams})) \\ \times (3600 \text{ sec/hour}) \times (1000 \text{ mrad/rad})$$

$$K = (5.76E-2) \times (\mu_a/\rho), ([\text{mrad/hr}]/\text{MeV/cm}^2/\text{sec})$$

These flux-to-dose conversion factors are presented in Table 5.3-1. The buildup factor is derived from an infinite medium. The correction factor accounts for the effect of multiple scattering in tissue rather than in an infinite medium of shielding material alone (Reference 5.5.1.5).

¹The International Commission on Radiological Protection (ICRP) was established in 1928. Since its inception, the ICRP has been the one internationally recognized body responsible for recommending values of maximum permissible exposure to ionizing radiation.

TABLE 5.3-1
Flux to Dose Conversion Factors
for QAD-CGGP

Energy (MeV)	MU/RHO (cm ² /g)	Correction Factor, C	Conversion Factor, K*
5	0.01892	1.01	1.101E-03
3.5	0.02151	1.01	1.251E-03
2.8	0.02324	1.01	1.352E-03
2.25	0.02504	1.02	1.471E-03
1.75	0.02698	1.025	1.593E-03
1.25	0.02945	1.05	1.781E-03
0.85	0.03157	1.06	1.928E-03
0.575	0.03268	1.08	2.033E-03
0.375	0.03235	1.12	2.087E-03
0.225	0.03001	1.2	2.074E-03
0.15	0.02741	1.31	2.068E-03
0.0575	0.03293	1.65	3.130E-03
0.01	4.40300	1.36	3.449E-01

* K = 0.0576 [(mrad/hr)/MeV/cm²-sec] * MU/RHO (cm²/g) * C

5.3.1 Description of Radial and Axial Shielding Configuration

Shielding in the NuPac 140-B Cask includes the following:

Package Wall:

The cylindrical cask wall consists of a total stainless steel thickness of 2.875 inches, 3.50 inches of lead, 0.50 inch copper fins corrugated through 5.875 inches of Boro-Silicone, and a 0.125 inch copper fin extension. A 0.50 inch air gap exists between the basket and cask wall. This air gap is ignored and the source region is homogenized in the 56.5 inch basket diameter. This is a conservative assumption since the detectors are then closer to the source.

The Boro-Silicone begins a transition at 52.25 inches from the cask end closure from 5.875 inches to 3.75 inches. This does not present a neutron shielding problem. The impact limiters extend over this transition during transport so that the boro-silicon decreases to no less than 4.6 inches before the neutron shielding benefits of the impact limiters are available. The neutron source strength also decreases toward the ends of the fuel assembly further decreasing the neutron dose rate outside the cask in this region.

Package Ends:

The package ends are symmetrical except for the location of the ports. The ends consist of a total stainless steel thickness of 9.25 inches, with 2.5 inches of Boro-Silicone inset into the stainless steel. The impact limiters contain 0.50 inches of copper and 40 inches of foam. The Boro-Silicone thickness was included in the ends for ALARA to reduce worker exposure during impact limiter installation. Without Boro-Silicone, the neutron dose rate at the end surface of the cask is 60 mrem/hour (without impact limiters). With 2.5 inches of Boro-Silicone the neutron dose rate is reduced to 23.0 mrem/hour at the end surface of the cask.

Ports:

The vent port is located in the top plate, 18.6 inches from the cask centerline; within the fuel region radius. The port is shielded by 2.25 inches of steel and 1.75 inches of Boro-Silicone. The possibility for streaming exists through five 3/16 inch tapped holes. A similar situation occurs with the test port and rupture port. Further analysis is needed to assure a streaming hazard does not exist with the current design. Potential problems could be resolved by using depleted uranium wafers around each bolt, or by placing a depleted uranium plug within the steel lid.

Drain Port:

The drain port is positioned in the side of the lid. The drain consists of a 1.5 inch diameter port in the axial lid portion, the bore turns 80 degrees, and the plug is in the side of the lid. There is a possibility for excessive shine through the drain port plug, or through the lid in the axial direction. Additional shielding or a more tortuous path may be required in this region.

5.3.2 Shield Regional Densities

The lightest fuel assembly was chosen to determine the density for self absorption in the active fuel region. For PWR fuel this a Westinghouse 14 x 14. The assembly characteristics listed below were used to determine the source region densities for the PWR basket:

Assembly Weight:	1096 lbs./assembly
Heavy Metal Weight:	741 lbs. heavy metal/assembly
Active Fuel Length:	135.2 inches

The top and bottom nozzle combined weigh 11.73 kilograms. This weight is subtracted for radial dose rate calculations. The basket is assumed to contain 23.5% by weight copper and 76.5% by weight steel. The density of the PWR fuel region is calculated as follows:

The total weight, W, of the active fuel region:

$$\begin{array}{rcl}
 (13,300 \text{ lb. basket weight}) \times (135.2"/180") & = & 9,990 \text{ lbs} \\
 (1,070 \text{ lbs./assembly}) \times (21 \text{ assemblies}) & = & + \underline{22,470 \text{ lbs}} \\
 & & 32,460 \text{ lbs}
 \end{array}$$

$$W = (32,460 \text{ lbs}) \times (453.8 \text{ grams/lb})$$

$$W = 1.473 \times 10^7 \text{ grams}$$

The total volume, V, of the active fuel region:

$$\begin{array}{l}
 V = 3.142 (56.5 \text{ inches}/2)^2 \times (135.2 \text{ inches}) \times (2.54 \text{ cm/in})^3 \\
 V = 5.555 \times 10^6 \text{ cm}^3
 \end{array}$$

Total Density, ρ , of the active fuel region is W/V:

$$\begin{array}{l}
 \rho = 1.4473 \times 10^7 \text{ grams}/5.555 \times 10^6 \text{ cm}^3 \\
 \rho = 2.652 \text{ g/cm}^3
 \end{array}$$

The active fuel region was homogenized, and the total density, ρ , was divided into the density of the constituent elements in the source region.

The same type of calculation was performed for the BWR active fuel region. The assumed BWR basket weight for calculations is 15,500 pounds. Table 5.3.2-1 provides a summary of the elemental densities of each shield region included in the shielding analysis.

TABLE 5.3.2-1
Shield Regional Densities
(g/cm³)

Atomic Number	Element	PWR Source	BWR Source	Top Nozzle	Steel	Lead	Boro-Silicone	Copper	Impact Limiter	Air
1	Hydrogen						0.0752		0.014	
5	Boron						0.0169		0.1242	
6	Carbon						0.1716			2.997E-04
7	Nitrogen								0.018	9.765E-04
8	Oxygen	0.171	0.2043				0.7463		0.004	
11	Sodium						0.0001			
13	Aluminum						0.2999			
14	Silicone						0.2789			
26	Iron	0.6242	0.7924	0.228	7.87					
29	Copper	0.1919	0.1585					8.96		
40	Zirconium	0.394	0.429							
82	Lead					11.34				
92	Uranium	1.271	1.519							
Total Density:		2.6521	3.1032	0.228	7.87	11.34	1.5889	8.96	0.1602	0.0012762

5.4 Shielding Evaluation

Shielding attenuation was determined using standard formulae. The methodologies used in this analysis were previously benchmarked with available measured test data for a similar cask. The benchmark problem is a 24 element spent fuel shipping cask. Measured values were compared with calculated values. The calculated values were bounded by the measured values in all cases.

Sufficient conservatism has been used in this analysis to insure the calculated dose rates in this analysis are bounding. The assumptions include the following:

- 1) The neutron energy spectrum of Cf^{252} was used for the neutron energy distribution. The spectrum of Cf^{252} has a greater high-energy component than U^{235} ; consequently the fission neutrons are more penetrating than those of U^{235} .
- 2) The axial calculations conservatively neglected any shielding benefits from the basket. This includes the 0.25 inch end plate.
- 3) The radial dose rate was multiplied by a safety factor of 1.08 to account for uneven burnup along the fuel's radial profile. This safety factor is for fuel aged 5 years and older (Reference 5.5.1.6). The dose rate was then multiplied by 2 for PWR neutron dose rates and 1.25 for PWR gamma dose rates (1.4 for BWR gamma dose rates).
- 4) The 10 CFR 71.47 dose limit is for two meters from the edge of the conveyance (or the personnel barrier). The two meter dose limits in this analysis were determined at two meters from the edge of the cask body.

These analyses have shown that the shielding provided by the 140-B Cask is adequate for 21 PWR fuel assemblies with 35,000 MWD/MTU and 10 years decay. The limiting factor is the dose rate limit of 10 mrem/hr at two meters. For 21 PWR assemblies, the calculated two meter dose rate is 9.4 mrem/hr in the radial direction and 8.0 in the axial direction. The 30,000 MWD/MTU BWR fuel resulted in dose rates well below the 10 CFR 71

limit. For 52 BWR assemblies, the calculated two meter dose rate is 4.7 mrem/hr in the radial direction and 6.3 mrem/hr in the axial direction. BWR fuel with a higher burnup or shorter decay period could legally be shipped in this cask. The burnup limits for BWR fuel require further analysis.

APPENDIX 5.5

Appendix 5.5.1 References

Appendix 5.5.2 ANISN Input Files

Appendix 5.5.3 ANISN Dose Rates

Appendix 5.5.4 QAD-CGGP Input Files

APPENDIX 5.5.1

References

- 5.5.1.1 U.S. Department of Energy, "Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation", Office of Civilian Radioactive Waste Management, DOE/RW-0184, December 1987.
- 5.5.1.2 RSIC Data Library Collection, "Cask-81", Oak Ridge National Laboratory, DLC-23, 1987.
- 5.5.1.3 RSIC Computer Code Collection, "ANISN/PC", Oak Ridge National Laboratory, CCC-514 MICRO, 1988.
- 5.5.1.4 RSIC Computer Code Collection, "QAD-CGGP", Oak Ridge National Laboratory, CCC-493, 1986.
- 5.5.1.5 D.K. Trubey, "New Gamma Ray Buildup Factor Data for Point Kernel Calculations", ANS-6.43 Standard Reference Data, Oak Ridge National Laboratory, ORNL/RSIC-49, 1988.
- 5.5.1.6 Reg. Guide 3.5.4, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation", September 1984.

APPENDIX 5.5.2
ANISN Input Files

- 5.5.2.1 PWR Radial
- 5.5.2.2 PWR Axial (neutron input only)
- 5.5.2.3 BWR Radial
- 5.5.2.4 BWR Axial (neutron input only)

(PACRAD2 INP)DCRWM 21PWR CASK 35K BURNUP 10 Y DECAY FUEL REVISED

```

15** 1 0 1 8 2 1 0 8 0 0
      40 3 4 43 44 32 0 44 0 0
      1 0 0 20 0 0 0 0 5 1
      0 0 0 1 1 0

```

```

16** 2R0.0 0.0001 1.420892 347.22
      4R0.0 0.5000 0.0002 F0.0
      T

```

```

14* 0 +30165- 9 0 + 0+ 0 0 +73389- 5 0 +51569- 639R+ 0+ 0 0 +31936- 9
    0 + 0+ 0 0 +87732- 5 0 +64911- 6 0 +12394- 538R+ 0+ 0 0 +33346- 9
    0 -11615- 4 0 -65185- 4 0 -76103- 4 0 -62275- 4 0 -47349- 4 0 -37530- 4
    0 -29534- 4 0 -23285- 4 0 -17638- 4 0 -13124- 4 0 -99318- 5 0 -76285- 5
    0 -60054- 528R+ 0+ 0 0 +30171- 3 0 -91028- 4 0 -15370- 3 0 -91044- 4
    0 -32490- 4 0 -39781- 537R+ 0+ 0 0 +19624- 3 0 -11244- 341R+ 0+ 0
    0 +68748- 4 0 -61356- 438R+ 0+ 0
    T

```

Not a Complete 14* array

```

17** 33R0.11 47R0.0
      33R0.43 47R0.0
      33R1.33 47R0.0
      33R4.47 47R0.0
      33R9.29 47R0.0
      33R12.01 47R0.0
      33R25.40 47R0.0
      33R20.44 47R0.0
      33R4.84 47R0.0
      33R26.84 47R0.0
      33R48.50 47R0.0
      33R44.58 47R0.0
      33R30.80 47R0.0
      33R3.12 47R0.0

```

640R0.0

```

      33R0.13 47R0.0
      33R1.13 47R0.0
      33R0.0 47R0.0
      33R9.83 47R0.0
      33R8.440E+02 47R0.0
      33R7.050E+03 47R0.0
      33R1.110E+05 47R0.0
      33R5.310E+06 47R0.0
      33R0.0 47R0.0
      33R4.060E+08 47R0.0
      33R4.560E+08 47R0.0
      33R0.0 47R0.0
      33R5.990E+09 47R0.0
      33R1.690E+08 47R0.0
      33R3.520E+08 47R0.0
      33R3.870E+08 47R0.0
      33R1.160E+09 47R0.0
      33R5.310E+09 47R0.0

```

T

```

3** 33R1.0 47R0.0 39080 T
1** F0.0
4** 3210.0 3172.390 9175.250 3184.150 10187.960 21101.61
      21103.200 11103.840 91105.110 305.11
5** F1.0
6** 0.0 0604938 0453704 0453704 0604938 0604938 0453704
      0453704 0604938 0.00 0453704 0462962 0453704 0453704
      0462962 0453704 0.00 0453704 0453704 0453704 0453704
      0.00 0604938 0604938
7** - 9759000 - 9511897 - 7867958 - 5773503 - 2182179 + 2182179
      + 5773503 + 7867958 + 9511897 - 8164965 - 7867958 - 5773503
      - 2182179 + 2182179 + 5773503 + 7867958 - 6172134 - 5773503
      - 2182179 + 2182179 + 5773503 - 3086057 - 2182179 + 2182179
8** 33R1 4R2 10R3 4R4 11R5 3R6 3R7
      12R8

```

REVISION 5
 PREPARED BY JF6/5-24-89
 CHECKED BY RJA/5-25-89
 PAGE A.3.20 A.3.14

948	33	35	39	35	41	43	35
	37						

/ MAT 1. 2 = H
 / MAT 3. 4 = B
 / MAT 5. 6 = C
 / MAT 7. 8 = N
 / MAT 9. 10 = O
 / MAT 11. 12 = Na
 / MAT 13. 14 = AL
 / MAT 15. 16 = SI
 / MAT 17. 18 = CR
 / MAT 19. 20 = FE
 / MAT 21. 22 = NI
 / MAT 23. 24 = CU
 / MAT 25. 26 = ZR
 / MAT 27. 28 = PB
 / MAT 29. 30 = U235
 / MAT 31. 32 = U238
 /
 / MIXTURE 33. 34 = FUEL
 / MIXTURE 35. 36 = STEEL
 / MIXTURE 37. 38 = AIR
 / MIXTURE 39. 40 = LEAD
 / MIXTURE 41. 42 = B-SI
 / MIXTURE 43. 44 = CU
 /

1088

/FUEL-

33 34
 33 34
 33 34
 33 34
 33 34
 33 34
 33 34
 33 34

/STEEL-

35 36
 35 36
 35 36

/AIR-

37 38
 37 38

/LEAD-

39 40

/B-SI

41 42
 41 42
 41 42
 41 42
 41 42
 41 42
 41 42

/COPPER

43 44

1188

/FUEL-

9 10

REVISION 0 PAGE A.3.3 A.3.14
 PREPARED BY 986/5-24-81
 CHECKED BY RTM 515-81

17	18
19	20
21	22
23	24
25	26
29	30
31	32
/ STEEL-	
17	18
19	20
21	22
/ AIR-	
7	8
9	10
/ LEAD-	
27	28
/ B-SI	
1	2
3	4
5	6
9	10
11	12
13	14
15	16
/ COPPER	
23	24
12**	
/ FUEL	
2R6. 434E-3	
2R1. 374E-3	
2R4. 848E-3	
2R5. 766E-4	
2R1. 819E-3	
2R2. 599E-3	
2R1. 303E-4	
2R3. 088E-3	
/ STEEL	
2R1. 721E-2	
2R6. 071E-2	
2R7. 218E-3	
/ AIR	
2R1. 98E-5	
2R5. 28E-6	
/ LEAD	
2R3. 296E-2	
/ B-SI	
2R4. 494E-2	
2R9. 400E-4	
2R8. 600E-3	
2R2. 810E-2	
2R2. 500E-5	
2R6. 690E-3	
2R5. 980E-3	
/ COPPER	
2R8. 493E-2	
1986	F1
	T

REVISION 0 PAGE A.3.4
 PREPARED BY JG/S-24-87
 CHECKED BY Rock 87

(PACAX12 IMP)DCRMH 21PMR CASK 35K SUP 10Y DECAY TOP AXIAL IMP LIMITER.

```

15** 3      0      1      16      1      1      0      7      100      0
      40      3      4      43      60      32      0      48      0      0
      1      0      0      20      0      0      0      0      5      1
      0      0      0      1      1      0

```

```

16** 2R0.0      0.0001      1.420892      347.22
      4R0.0      0.5000      0.0002      F0.0

```

T

```

14**
0 +30165- 9 0 +      0+ 0 0 +73389- 5 0 +51549- 639R+      0+ 0 0 +31936- 9
0 +      0+ 0 0 +87732- 5 0 +64911- 6 0 +12394- 538R+      0+ 0 0 +33346- 9
0 +      0+ 0 0 +10376- 4 0 +81680- 6 0 +14759- 5 0 +10147- 537R+      0+ 0
0 +35005- 9 0 +      0+ 0 0 +12554- 4 0 +11697- 5 0 +21378- 5 0 +14878- 5
0 -65988- 5 0 -55292- 5 0 -46390- 529R+      0+ 0 0 +31075- 3 0 +17834- 3
0 -11615- 4 0 -65185- 4 0 -76103- 4 0 -62275- 4 0 -47349- 4 0 -37530- 4
0 -29534- 4 0 -23285- 4 0 -17638- 4 0 -13124- 4 0 -99318- 5 0 -76285- 5
0 -60054- 528R+      0+ 0 0 +30171- 3 0 -91028- 4 0 -15370- 3 0 -91044- 4
0 -32490- 4 0 -39781- 537R+      0+ 0 +19624- 3 0 -11244- 341R+      0+ 0
0 +68748- 4 0 -61356- 438R+      0+ 0

```

T

```

17** 33R0.11      67R0.0
      33R0.43      67R0.0
      33R1.33      67R0.0
      33R4.47      67R0.0
      33R9.29      67R0.0
      33R12.01      67R0.0
      33R25.40      67R0.0
      33R20.44      67R0.0
      33R4.84      67R0.0
      33R26.84      67R0.0
      33R48.50      67R0.0
      33R44.58      67R0.0
      33R30.80      67R0.0
      33R3.12      67R0.0

```

800R0.0

```

      33R0.13      67R0.0
      33R1.13      67R0.0
      33R0.0      67R0.0
      33R9.83      67R0.0
      33R8.440E+02      67R0.0
      33R7.050E+03      67R0.0
      33R1.110E+05      67R0.0
      33R5.310E+06      67R0.0
      33R0.0      67R0.0
      33R4.060E+08      67R0.0
      33R4.560E+08      67R0.0
      33R0.0      67R0.0
      33R5.990E+09      67R0.0
      33R1.690E+08      67R0.0
      33R3.520E+08      67R0.0
      33R3.870E+08      67R0.0
      33R1.160E+09      67R0.0
      33R5.310E+09      67R0.0

```

T

```

3** 33R1.0      67R0.0      390100      T
1** F0.0
4** 3210.0      191171.70      141229.24      31258.45      31259.08
      151259.56      11361.16      51362.43      662.43
5** F1.0
6** 0.0      0244936      0413296      0392569      0400796      0643754      0442097
      1090850      1371702      1371702      1090850      0442097      0643754
      0400796      0392569      0413296      0244936
7** - 9952984 - 9805009 - 9092855 - 8319966 - 7467506 - 6504264
      - 5370966 - 3922893 - 1389568 + 1389568 + 3922893 + 5370966
      + 6504264 + 7467506 + 8319966 + 9092855 + 9805009
8** 33R1      20R2      15R3      4R4      4R5      16R6      8R7
9** 33      45      25      43      35      47      37

```

REVISION 0
 PAGE A.31104.3.14
 PREPARED BY JES-24-87
 CHECKED BY R. 5-16-87

```

/ MAT 1. 2 = H
/ MAT 3. 4 = B
/ MAT 5. 6 = C
/ MAT 7. 8 = N
/ MAT 9. 10 = O
/ MAT 11. 12 = Na
/ MAT 13. 14 = AL
/ MAT 15. 16 = SI
/ MAT 17. 18 = CR
/ MAT 19. 20 = FE
/ MAT 21. 22 = NI
/ MAT 23. 24 = CU
/ MAT 25. 26 = ZR
/ MAT 27. 28 = PB
/ MAT 29. 30 = U235
/ MAT 31. 32 = U238
/
/ MIXTURE 33. 34 = FUEL
/ MIXTURE 35. 36 = STEEL
/ MIXTURE 37. 38 = AIR
/ MIXTURE 39. 40 = LEAD
/ MIXTURE 41. 42 = B-SI
/ MIXTURE 43. 44 = CU
/ MIXTURE 45. 46 = BASKT
/ MIXTURE 47. 48 = IMPLI

```

1000

/FUEL-

```

33 34
35 36
37 38
39 40
41 42
43 44
45 46
47 48

```

/STEEL-

```

35 36
37 38
39 40

```

/AIR-

```

37 38
39 40

```

/LEAD-

```

39 40

```

/B-SI

```

41 42
41 43
41 44
41 45
41 46
41 47
41 48

```

/COPPER

```

43 44

```

/TOP BASKET

```

45 46
45 46
45 46

```

REVISION 0 PAGE A.3.12 of A.3.14
 PREPARED BY 4845-24-83
 CHECKED BY Rm/6-25-87

45 46

/IMP LIMITER
47 48
47 48
47 48
47 48

1199
/ FUEL-
9 10
17 18
19 20
21 22
23 24
25 26
29 30
31 32

/ STEEL-
17 18
19 20
21 22

/ AIR-
7 8
9 10

/LEAD-
27 28

/B-SI
1 2
3 4
5 6
9 10
11 12
13 14
15 16

/COPPER
23 24

/TOP BASKET
17 18
19 20
21 22
23 24

/IMP LIMITER
1 2
5 6
7 8
9 10

1299
/ FUEL
2R6. 434E-3
2R1. 374E-3
2R4. 848E-3
2R5. 744E-4
2R1. 817E-3
2R2. 997E-3
2R1. 303E-4
2R3. 088E-3

/ STEEL
2R1. 721E-2
2R4. 071E-2
2R7. 218E-3

/ AIR
2R1. 98E-3
2R3. 28E-4

/LEAD
2R3. 296E-2

REVISION 0 PAGE A.3.13 of A.3.14
PREPARED BY 4/28/5-24-89
CHECKED BY Ryan 5-25-89

2R3. 296E-2
 /B-SI 2R4. 494E-2
 2R9. 400E-4
 2R8. 600E-3
 2R2. 810E-2
 2R2. 500E-3
 2R6. 690E-3
 2R5. 980E-3
 /COPPER 2R8. 493E-2
 /TOP BASKET 2R1. 372E-3
 2R4. 842E-3
 2R5. 758E-4
 2R1. 797E-3
 /IMP LIMITER 2R8. 378E-3
 2R6. 227E-3
 2R7. 752E-4
 2R1. 659E-3
 1988 F1
 T T

REVISION 0 PAGE A.3.14 OF 8 A.3.14
 PREPARED BY JKB/S-24-87
 CHECKED BY AT-5-49A

[BWR30] OCRWM 52 BWR / 30K BURNUP / 10 Y DECAY / RADIAL 9-14-89

15**	1	0	1	6	2	1	0	8	100	0
	40	3	4	43	46	28	0	44	0	0
	1	0	0	20	0	0	0	0	5	0
	0	0	0	1	1	0				

16**	2R0.0	0.0001	1.420892	347.22
	4R0.0	0.5000	0.0002	F0.0

T

14* (Cross section data not included)

T

17**	33R0.06093	67R0.0
	33R0.2466	67R0.0
	33R0.7537	67R0.0
	33R2.52	67R0.0
	33R5.238	67R0.0
	33R6.776	67R0.0
	33R14.33	67R0.0
	33R11.53	67R0.0
	33R2.734	67R0.0
	33R15.14	67R0.0
	33R27.36	67R0.0
	33R25.14	67R0.0
	33R17.38	67R0.0
	33R1.761	67R0.0

800R0.0

	33R0.0793	67R0.0
	33R0.6902	67R0.0
	33R0.0	67R0.0
	33R5.989	67R0.0
	33R5.361E+02	67R0.0
	33R4.398E+03	67R0.0
	33R6.823E+04	67R0.0
	33R3.755E+06	67R0.0
	33R0.0	67R0.0
	33R1.940E+08	67R0.0
	33R2.959E+08	67R0.0
	33R0.0	67R0.0
	33R4.381E+09	67R0.0
	33R1.248E+09	67R0.0
	33R2.601E+08	67R0.0
	33R2.810E+08	67R0.0
	33R8.472E+08	67R0.0
	33R3.920E+09	67R0.0

T

3** 33R1.0 67R0.0 39Q100 T

1** F0.0

4**	3210.0	8172.390	15175.250	9184.150	10187.96	41101.61
	31103.200	11103.840	71105.110	11303.34	305.88	

5** F1.0

6**	0.0	.0880632	.0786035	.0880632	.0880632	.0786035	.0880632
	0.0	.0786035	.0786035	.0786035	.0786035	0.00	.0880631

7**	-.9637974	-.9261808	-.6815076	-.2666355	+.2666355	+.6815076
	+.9261808	-.7318110	-.6815076	-.2666355	+.2666355	+.6815076
	-.3770795	-.2666355	+.2666355			

8**	33R1	9R2	16R3	10R4	11R5	5R6

9**	4R7	12R8				
	29	31	35	31	37	39
	31	33				

/ MAT 1, 2 = H
 / MAT 3, 4 = B
 / MAT 5, 6 = C
 / MAT 7, 8 = N
 / MAT 9, 10 = O
 / MAT 11, 12 = Na
 / MAT 13, 14 = AL
 / MAT 15, 16 = SI
 / MAT 17, 18 = FE
 / MAT 19, 20 = CU
 / MAT 21, 22 = ZR
 / MAT 23, 24 = PB
 / MAT 25, 26 = U235
 / MAT 27, 28 = U238
 /
 / MIXTURE 29, 30 = FUEL
 / MIXTURE 31, 32 = STEEL

```

/ MIXTURE 33, 34 = AIR
/ MIXTURE 35, 36 = LEAD
/ MIXTURE 37, 38 = B-SI
/ MIXTURE 39, 40 = CU
/ MIXTURE 41, 42 = TOP-NZL
/ MIXTURE 43, 44 = IMPL

```

10\$\$

/FUEL-

```

29 30
29 30
29 30
29 30
29 30
29 30

```

/STEEL-

31 32

/AIR-

```

33 34
33 34

```

/LEAD-

35 36

/B-SI

```

37 38
37 38
37 38
37 38
37 38
37 38
37 38

```

/COPPER

39 40

/TOP BASKET

41 42

/IMPACT LIMITER

```

43 44
43 44
43 44
43 44

```

11\$\$

/ FUEL-

```

9 10
17 18
19 20
21 22
25 26
27 28

```

/ STEEL-

17 18

/ AIR-

```

7 8
9 10

```

/LEAD-

23 24

/B-SI

```

1 2
3 4
5 6
9 10
11 12
13 14
15 16

```

/COPPER

19 20

/TOP-NZL

17 18

/IMPACT LIMITER

```

1 2
5 6
7 8
9 10

```

12**

/ FUEL

```

2R7.691E-3
2R8.546E-3
2R1.502E-3
2R2.834E-3

```

2R1.168E-4
2R3.729E-3
/ STEEL 2R8.487E-2
/ AIR 2R1.98E-5
2R5.28E-6
/LEAD 2R3.296E-2
/B-SI 2R4.494E-2
2R9.400E-4
2R8.600E-3
2R2.810E-2
2R2.500E-5
2R6.690E-3
2R5.980E-3
/COPPER 2R8.493E-2
/TOP BASKET 2R2.459E-3
/IMPACT LIMITER 2R8.378-3
2R6.227-3
2R7.752-4
2R1.659-3
19SS F1
T T

[BWRAX30] 140-B 52 ASSY BWR / 30K BURNUP / 10 Y DECAY / AXIAL 9-14-89

```

15$$ 2 0 1 16 1 1 0 8 115 0
      40 3 4 43 46 28 0 44 0 0
      1 0 0 20 0 0 0 0 5 0
      0 0 0 1 1 0

```

```

16** 2R0.0 0.0001 1.420892 368.9
      4R0.0 0.5000 0.0002 F0.0

```

14* (Cross-section data not included)

```

T
17** 33R0.0609 82R0.0
      33R0.2466 82R0.0
      33R0.7537 82R0.0
      33R2.519 82R0.0
      33R5.238 82R0.0
      33R6.775 82R0.0
      33R14.33 82R0.0
      33R11.53 82R0.0
      33R2.734 82R0.0
      33R15.14 82R0.0
      33R27.36 82R0.0
      33R25.14 82R0.0
      33R17.38 82R0.0
      33R1.761 82R0.0
      920R0.0
      33R1.0E-20 82R0.0
      1955R0.0

```

```

T
3** 33R1.0 82R0.0 39Q115 T
1** F0.0
4** 3210.0 31184.46 51185.73 241192.08 51213.67
      241214.94 31316.54 11317.17 71318.44 11517.44
      518.71
5** F1.0
6** 0.0 .0244936 .0413296 .0392569 .0400796
      .0643754 .0442097 .1090850 .1371702 .1371702
      .1090850 .0442097 .0643754 .0400796 .0392569
      .0413296 .0244936
7** -.9902984 -.9805009 -.9092855 -.8319966 -.7467506
      -.6504264 -.5370966 -.3922893 -.1389568 +.1389568
      +.3922893 +.5370966 +.6504264 +.7467506 +.8319966
      +.9092855 +.9805009
8$$ 33R1 4R2 6R3 25R4 6R5
      25R6 4R7 12R8
9$$ 29 31 37 31 39
      43 31 33

```

```

/ MAT 1, 2 = H
/ MAT 3, 4 = B
/ MAT 5, 6 = C
/ MAT 7, 8 = N
/ MAT 9, 10 = O
/ MAT 11, 12 = Na
/ MAT 13, 14 = AL
/ MAT 15, 16 = SI
/ MAT 17, 18 = FE
/ MAT 19, 20 = CU
/ MAT 21, 22 = ZR
/ MAT 23, 24 = PB
/ MAT 25, 26 = U235
/ MAT 27, 28 = U238
/
/ MIXTURE 29, 30 = FUEL
/ MIXTURE 31, 32 = STEEL
/ MIXTURE 33, 34 = AIR
/ MIXTURE 35, 36 = LEAD
/ MIXTURE 37, 38 = B-SI
/ MIXTURE 39, 40 = CU
/ MIXTURE 41, 42 = TOP-NZL
/ MIXTURE 43, 44 = IMPL

```

10\$\$

/FUEL-

```

29 30
29 30
29 30
29 30
29 30
29 30

```

```

31 32
/AIR-
33 34
33 34
/LEAD-
35 36
/B-SI
37 38
37 38
37 38
37 38
37 38
37 38
37 38
/COPPER
39 40
/TOP BASKET
41 42
/IMPACT LIMITER
43 44
43 44
43 44
43 44
11$$
/ FUEL-
9 10
17 18
19 20
21 22
25 26
27 28
/ STEEL-
17 18
/ AIR-
7 8
9 10
/LEAD-
23 24
/B-SI
1 2
3 4
5 6
9 10
11 12
13 14
15 16
/COPPER
19 20
/TOP-NZL
17 18
/IMPACT LIMITER
1 2
5 6
7 8
9 10
12**
/ FUEL
2R7.691E-3
2R8.546E-3
2R1.502E-3
2R2.834E-3
2R1.168E-4
2R3.729E-3
/ STEEL
2R8.487E-2
/ AIR
2R1.98E-5
2R5.28E-6
/LEAD
2R3.296E-2
/B-SI
2R4.494E-2
2R9.400E-4
2R8.600E-3
2R2.810E-2
2R2.500E-5
5.5.2-13

```

2R6.690E-3
2R5.980E-3
/COPPER 2R8.493E-2
/TOP BASKET 2R2.459E-3
/IMPACT LIMITER
2R8.378-3
2R6.227-3
2R7.752-4
2R1.659-3
1988 F1
T T

APPENDIX 5.5.3
ANISN Dose Rates

- 5.5.3.1 PWR Radial
- 5.5.3.2 BWR Radial
- 5.5.3.3 BWR Axial (neutron input only)

OCRWM CASK 21PWR ASSEM. 33000MWD/MTIHM 10YEAR DECAY RADIAL DOSE

INTERV	DOSE	G DOSE	TOTAL
1	4638.813	33102905	33107544
2	4635.930	33102913	33107549
3	4629.984	33102895	33107525
4	4621.168	33102876	33107497
5	4609.470	33102839	33107449
6	4594.895	33102823	33107418
7	4577.362	33102719	33107297
8	4556.822	33102681	33107238
9	4533.222	33102510	33107043
10	4506.467	33102390	33106896
11	4476.495	33102022	33106499
12	4443.197	33101731	33106174
13	4406.476	33101221	33105628
14	4366.236	33100518	33104884
15	4322.333	33099482	33103805
16	4274.656	33098091	33102366
17	4223.045	33096077	33100300
18	4167.344	33093019	33097187
19	4107.382	33088974	33093081
20	4042.981	33082549	33086592
21	3973.925	33074137	33078111
22	3900.010	33060220	33064120
23	3820.983	33042145	33045966
24	3736.560	33011512	33015249
25	3646.446	32970679	32974325
26	3550.317	32900304	32903854
27	3447.746	32801503	32804951
28	3338.242	32628667	32632005
29	3221.141	32355144	32358365
30	3095.457	31883117	31886213
31	2959.526	30892834	30895793
32	2810.010	29316593	29319403
33	2639.515	22876906	22879546
34	2456.465	13616923	13619379
35	2285.644	7939018.	7941303.
36	2134.054	5207091.	5209225.
37	1995.867	3370413.	3372409.
38	1877.827	1192772.	1194650.
39	1774.794	360911.4	362686.2
40	1673.622	147673.0	149346.6
41	1574.707	64378.32	65953.03
42	1478.108	30263.53	31741.64
43	1383.735	14550.67	15934.40
44	1291.432	6983.284	8274.716
45	1200.989	3514.714	4715.703
46	1112.114	1769.697	2881.812
47	1024.346	908.6022	1932.948
48	912.0921	595.3840	1507.476
49	777.6452	449.9689	1227.614
50	646.4172	320.9466	967.3639
51	512.2294	229.0104	741.2398
52	353.1276	184.9480	538.0757
53	219.1119	168.2616	387.3735

REVISION 0 PAGE 41 of 50
 PREPARED BY RS/S-24-93
 CHECKED BY RS/S-24-93

<i>Interval</i>	NDOSE	Δ DOSE	TOTAL
54	146.0517	152.2010	298.2527
55	97.83780	137.1850	235.0228
56	66.44365	123.3121	189.7557
57	45.61402	110.5799	156.1939
58	31.69261	98.91299	130.6056
59	22.32727	88.22576	110.5530
60	15.99349	78.41613	94.40962
61	11.71002	69.34281	81.05284
62	9.021459	60.66810	69.68956
63	7.670415	48.25634	55.92676
64	6.797494	36.48538	43.28287
65	5.763845	28.26167	34.02552
66	5.006879	23.48335	28.49023
67	4.584156	21.20489	25.78905
68	4.130333	18.78074	22.91108
69	3.864391	17.44588	21.31027
70	3.805218	17.23947	21.04469
71	3.188509	14.95216	18.14067
72	2.568526	12.23474	14.80327
73	2.147577	10.40086	12.54843
74	1.865018	9.075716	10.94073
75	1.647861	8.067909	9.715771
76	1.479127	7.264978	8.744105
77	1.341742	6.610886	7.952629
78	1.227306	6.064785	7.292092
79	1.130184	5.601110	6.731294
80	1.046379	5.201609	6.247989

REVISION 0 PAGE 42 of 50
 PREPARED BY JAB/S-44-73
 CHECKED BY RT-15-15-87

ZONE MATERIAL	MESH INTERVAL	OUTER MESH INTERVAL RADIUS	NEUTRON (nrem/hr)	GAMMA (nrem/hr)	TOTAL (nrem/hr)
FUEL	1	2.194000E+00	2.668458E+03	2.491631E+07	2.491897E+07
FUEL	2	4.387000E+00	2.666820E+03	2.491630E+07	2.491896E+07
FUEL	3	6.581000E+00	2.663413E+03	2.491628E+07	2.491894E+07
FUEL	4	8.775000E+00	2.658371E+03	2.491630E+07	2.491896E+07
FUEL	5	1.097000E+01	2.651686E+03	2.491627E+07	2.491892E+07
FUEL	6	1.316000E+01	2.643343E+03	2.491630E+07	2.491894E+07
FUEL	7	1.536000E+01	2.633305E+03	2.491626E+07	2.491889E+07
FUEL	8	1.755000E+01	2.621534E+03	2.491629E+07	2.491891E+07
FUEL	9	1.974000E+01	2.607985E+03	2.491623E+07	2.491884E+07
FUEL	10	2.194000E+01	2.592610E+03	2.491626E+07	2.491885E+07
FUEL	11	2.413000E+01	2.575335E+03	2.491619E+07	2.491876E+07
FUEL	12	2.632000E+01	2.556104E+03	2.491619E+07	2.491875E+07
FUEL	13	2.852000E+01	2.534842E+03	2.491603E+07	2.491856E+07
FUEL	14	3.071000E+01	2.511476E+03	2.491601E+07	2.491852E+07
FUEL	15	3.290000E+01	2.485900E+03	2.491549E+07	2.491817E+07
FUEL	16	3.510000E+01	2.458034E+03	2.491547E+07	2.491793E+07
FUEL	17	3.729000E+01	2.427761E+03	2.491510E+07	2.491752E+07
FUEL	18	3.949000E+01	2.394979E+03	2.491438E+07	2.491682E+07
FUEL	19	4.168000E+01	2.359544E+03	2.491371E+07	2.491607E+07
FUEL	20	4.387000E+01	2.321374E+03	2.491203E+07	2.491435E+07
FUEL	21	4.607000E+01	2.280270E+03	2.491023E+07	2.491251E+07
FUEL	22	4.826000E+01	2.236094E+03	2.490604E+07	2.490629E+07
FUEL	23	5.045000E+01	2.188683E+03	2.490168E+07	2.490387E+07
FUEL	24	5.265000E+01	2.137844E+03	2.489155E+07	2.489368E+07
FUEL	25	5.484000E+01	2.083360E+03	2.487997E+07	2.488205E+07
FUEL	26	5.703000E+01	2.025016E+03	2.485417E+07	2.485420E+07
FUEL	27	5.923000E+01	1.962528E+03	2.482084E+07	2.482280E+07
FUEL	28	6.142000E+01	1.895551E+03	2.475144E+07	2.475334E+07
FUEL	29	6.362000E+01	1.823444E+03	2.463716E+07	2.463898E+07
FUEL	30	6.581000E+01	1.746151E+03	2.443086E+07	2.443261E+07
FUEL	31	6.800000E+01	1.662002E+03	2.388023E+07	2.388189E+07
FUEL	32	7.020000E+01	1.569250E+03	2.309221E+07	2.309378E+07
FUEL	33	7.239000E+01	1.464131E+03	1.795958E+07	1.796104E+07
STEEL	34	7.271000E+01	1.386198E+03	1.195062E+07	1.195200E+07
STEEL	35	7.303000E+01	1.344091E+03	9.239394E+06	9.240738E+06
STEEL	36	7.334000E+01	1.304202E+03	7.344982E+06	7.346286E+06
STEEL	37	7.364000E+01	1.266178E+03	5.932500E+06	5.933766E+06
STEEL	38	7.398000E+01	1.229754E+03	4.848670E+06	4.849900E+06
STEEL	39	7.430000E+01	1.194733E+03	3.988970E+06	3.990165E+06
STEEL	40	7.461000E+01	1.160967E+03	3.285838E+06	3.286999E+06
STEEL	41	7.493000E+01	1.128349E+03	2.688958E+06	2.690086E+06
STEEL	42	7.525000E+01	1.096811E+03	2.139435E+06	2.140532E+06
LEAD	43	7.581000E+01	1.063073E+03	9.854150E+05	9.864781E+05
LEAD	44	7.634000E+01	1.026824E+03	3.894542E+05	3.904830E+05
LEAD	45	7.688000E+01	9.910084E+02	1.925401E+05	1.935311E+05
LEAD	46	7.747000E+01	9.554481E+02	1.018451E+05	1.028007E+05
LEAD	47	7.809000E+01	9.207475E+02	6.061452E+04	6.153327E+04
LEAD	48	7.890000E+01	8.863059E+02	3.605535E+04	3.694165E+04
LEAD	49	7.914000E+01	8.523138E+02	2.257589E+04	2.342821E+04
LEAD	50	7.970000E+01	8.187594E+02	1.417886E+04	1.499762E+04
LEAD	51	8.026000E+01	7.854242E+02	9.058534E+03	9.844160E+03
LEAD	52	8.081000E+01	7.528849E+02	5.814443E+03	6.567330E+03
LEAD	53	8.137000E+01	7.205254E+02	3.765027E+03	4.485528E+03
LEAD	54	8.193000E+01	6.885104E+02	2.448561E+03	3.137071E+03
LEAD	55	8.248000E+01	6.568073E+02	1.602645E+03	2.259453E+03
LEAD	56	8.304000E+01	6.253741E+02	1.051882E+03	1.677254E+03
LEAD	57	8.359000E+01	5.941538E+02	6.942116E+02	1.288345E+03
LEAD	58	8.415000E+01	5.630744E+02	4.640805E+02	1.029155E+03
STEEL	59	8.453000E+01	5.325263E+02	3.853449E+02	9.178712E+02
STEEL	60	8.491000E+01	5.021080E+02	3.528593E+02	8.549673E+02
STEEL	61	8.529000E+01	4.723382E+02	3.171673E+02	7.895055E+02
STEEL	62	8.567000E+01	4.429225E+02	2.815093E+02	7.244318E+02
STEEL	63	8.604000E+01	4.137616E+02	2.480275E+02	6.617890E+02
STEEL	64	8.644000E+01	3.847444E+02	2.175187E+02	6.022651E+02
STEEL	65	8.682000E+01	3.557487E+02	1.902083E+02	5.459570E+02
STEEL	66	8.720000E+01	3.266140E+02	1.660419E+02	4.926560E+02

STEEL	67	8.758000E+01	2.971521E+02	1.450992E+02	4.422513E+02
STEEL	68	8.796000E+01	2.671356E+02	1.281449E+02	3.952804E+02
B-SI	69	8.920000E+01	2.003392E+02	1.172668E+02	3.176060E+02
B-SI	70	9.044000E+01	1.230624E+02	1.072616E+02	2.303240E+02
B-SI	71	9.168000E+01	8.095203E+01	9.768652E+01	1.786385E+02
B-SI	72	9.292000E+01	5.385474E+01	8.850621E+01	1.423609E+02
B-SI	73	9.416000E+01	3.625494E+01	7.986913E+01	1.161241E+02
B-SI	74	9.541000E+01	2.469875E+01	7.183527E+01	9.653401E+01
B-SI	75	9.665000E+01	1.703855E+01	6.438573E+01	8.142429E+01
B-SI	76	9.789000E+01	1.192705E+01	5.749119E+01	6.941824E+01
B-SI	77	9.913000E+01	8.496534E+00	5.107781E+01	5.957434E+01
B-SI	78	1.004000E+02	6.197592E+00	4.505180E+01	5.124940E+01
B-SI	79	1.016000E+02	4.752408E+00	3.917985E+01	4.393226E+01
CU	80	1.019000E+02	4.100875E+00	3.227734E+01	3.637821E+01
CU	81	1.022000E+02	3.831165E+00	2.682616E+01	3.065733E+01
CU	82	1.026000E+02	3.532373E+00	2.283759E+01	2.636996E+01
CU	83	1.029000E+02	3.208668E+00	1.950803E+01	2.271670E+01
CU	84	1.032000E+02	2.860562E+00	1.665542E+01	1.951598E+01
STEEL	85	1.034000E+02	2.603629E+00	1.480309E+01	1.740672E+01
STEEL	86	1.035000E+02	2.447425E+00	1.370436E+01	1.615178E+01
STEEL	87	1.037000E+02	2.285150E+00	1.258131E+01	1.486646E+01
STEEL	88	1.038000E+02	2.116013E+00	1.140429E+01	1.352030E+01
AIR	89	1.045000E+02	2.016228E+00	1.075344E+01	1.276967E+01
AIR	90	1.051000E+02	1.989876E+00	1.063835E+01	1.262822E+01
AIR	91	1.299000E+02	1.643981E+00	9.059655E+00	1.070364E+01
AIR	92	1.547000E+02	1.271073E+00	7.147407E+00	8.418480E+00
AIR	93	1.794000E+02	1.054594E+00	5.972860E+00	7.027454E+00
AIR	94	2.042000E+02	8.951204E-01	5.114981E+00	6.010101E+00
AIR	95	2.290000E+02	7.827420E-01	4.492056E+00	5.274798E+00
AIR	96	2.538000E+02	6.954982E-01	4.005735E+00	4.701234E+00
AIR	97	2.786000E+02	6.252897E-01	3.613776E+00	4.239066E+00
AIR	98	3.033000E+02	5.672722E-01	3.290185E+00	3.857458E+00
AIR	99	3.046000E+02	5.395952E-01	3.136103E+00	3.675699E+00
AIR	100	3.059000E+02	5.371214E-01	3.122350E+00	3.659472E+00

[BWRX30] 140-B 52 ASSY BWR / 30K BURNUP / 10 Y DECAY / AXIAL 9-14-89

ZONE MATERIAL	MESH INTERVAL	OUTER MESH INTERVAL RADIUS	NEUTRON (mrem/hr)	GAMMA (mrem/hr)	TOTAL (mrem/hr)
FUEL	1	5.590000E+00	3.234547E+03	1.548830E+01	3.250035E+03
FUEL	2	1.118000E+01	3.234295E+03	1.48617E+01	3.249781E+03
FUEL	3	1.677000E+01	3.233782E+03	1.48188E+01	3.249264E+03
FUEL	4	2.236000E+01	3.232983E+03	1.547521E+01	3.248458E+03
FUEL	5	2.795000E+01	3.231863E+03	1.546597E+01	3.247329E+03
FUEL	6	3.354000E+01	3.230377E+03	1.545378E+01	3.245831E+03
FUEL	7	3.913000E+01	3.228462E+03	1.543827E+01	3.243901E+03
FUEL	8	4.472000E+01	3.226049E+03	1.541879E+01	3.241468E+03
FUEL	9	5.031000E+01	3.223025E+03	1.539482E+01	3.238420E+03
FUEL	10	5.590000E+01	3.219268E+03	1.536531E+01	3.234634E+03
FUEL	11	6.149000E+01	3.214628E+03	1.532952E+01	3.229957E+03
FUEL	12	6.708000E+01	3.208915E+03	1.528593E+01	3.224200E+03
FUEL	13	7.267000E+01	3.201890E+03	1.523349E+01	3.217124E+03
FUEL	14	7.826000E+01	3.193280E+03	1.517004E+01	3.208450E+03
FUEL	15	8.385000E+01	3.182737E+03	1.509410E+01	3.197831E+03
FUEL	16	8.944000E+01	3.169842E+03	1.500272E+01	3.184844E+03
FUEL	17	9.502000E+01	3.154078E+03	1.489379E+01	3.168972E+03
FUEL	18	1.006000E+02	3.134844E+03	1.476335E+01	3.149607E+03
FUEL	19	1.062000E+02	3.111376E+03	1.460852E+01	3.125984E+03
FUEL	20	1.118000E+02	3.082786E+03	1.442416E+01	3.097210E+03
FUEL	21	1.174000E+02	3.047983E+03	1.420618E+01	3.062189E+03
FUEL	22	1.230000E+02	3.005652E+03	1.394841E+01	3.019600E+03
FUEL	23	1.286000E+02	2.954231E+03	1.364497E+01	2.967876E+03
FUEL	24	1.342000E+02	2.891819E+03	1.328958E+01	2.905109E+03
FUEL	25	1.397000E+02	2.816164E+03	1.287307E+01	2.829037E+03
FUEL	26	1.453000E+02	2.724560E+03	1.239335E+01	2.736954E+03
FUEL	27	1.509000E+02	2.613781E+03	1.183343E+01	2.625614E+03
FUEL	28	1.565000E+02	2.479978E+03	1.121274E+01	2.491191E+03
FUEL	29	1.621000E+02	2.318449E+03	1.049161E+01	2.328941E+03
FUEL	30	1.677000E+02	2.123862E+03	9.787425E+00	2.133649E+03
FUEL	31	1.733000E+02	1.887903E+03	8.996110E+00	1.896899E+03
FUEL	32	1.789000E+02	1.605143E+03	8.675086E+00	1.613818E+03
FUEL	33	1.845000E+02	1.239376E+03	7.893237E+00	1.247270E+03
STEEL	34	1.848000E+02	9.902283E+02	7.004333E+00	9.972326E+02
STEEL	35	1.851000E+02	9.181264E+02	6.618315E+00	9.247447E+02
STEEL	36	1.854000E+02	8.473871E+02	6.469494E+00	8.538566E+02
STEEL	37	1.857000E+02	7.754203E+02	6.765565E+00	7.821859E+02
B-SI	38	1.868000E+02	6.193567E+02	7.380344E+00	6.267371E+02
B-SI	39	1.878000E+02	4.392478E+02	7.682098E+00	4.469299E+02
B-SI	40	1.889000E+02	3.340598E+02	7.624689E+00	3.416845E+02
B-SI	41	1.900000E+02	2.611181E+02	7.324307E+00	2.684424E+02
B-SI	42	1.910000E+02	2.141434E+02	6.821877E+00	2.209653E+02
B-SI	43	1.921000E+02	1.925034E+02	6.057252E+00	1.985606E+02
STEEL	44	1.929000E+02	1.883649E+02	4.349217E+00	1.927141E+02
STEEL	45	1.938000E+02	1.832320E+02	3.068508E+00	1.863006E+02
STEEL	46	1.947000E+02	1.745619E+02	2.330749E+00	1.768926E+02
STEEL	47	1.955000E+02	1.652841E+02	1.875798E+00	1.671599E+02
STEEL	48	1.964000E+02	1.560131E+02	1.552053E+00	1.575651E+02
STEEL	49	1.973000E+02	1.469660E+02	1.311992E+00	1.482780E+02
STEEL	50	1.981000E+02	1.382227E+02	1.130274E+00	1.393530E+02
STEEL	51	1.990000E+02	1.298187E+02	9.849556E-01	1.308036E+02
STEEL	52	1.999000E+02	1.217643E+02	8.683576E-01	1.226326E+02
STEEL	53	2.007000E+02	1.140576E+02	7.710494E-01	1.148287E+02
STEEL	54	2.016000E+02	1.066876E+02	6.897125E-01	1.073773E+02
STEEL	55	2.024000E+02	9.963934E+01	6.201427E-01	1.002595E+02
STEEL	56	2.033000E+02	9.289533E+01	5.606657E-01	9.345600E+01
STEEL	57	2.042000E+02	8.643697E+01	5.093185E-01	8.694628E+01
STEEL	58	2.050000E+02	8.024519E+01	4.652535E-01	8.071044E+01
STEEL	59	2.059000E+02	7.430122E+01	4.276182E-01	7.472884E+01
STEEL	60	2.068000E+02	6.858551E+01	3.961466E-01	6.898166E+01
STEEL	61	2.076000E+02	6.308006E+01	3.707431E-01	6.345080E+01
STEEL	62	2.085000E+02	5.776682E+01	3.517653E-01	5.811858E+01
STEEL	63	2.094000E+02	5.262865E+01	3.399623E-01	5.296861E+01
STEEL	64	2.102000E+02	4.764861E+01	3.366651E-01	4.798528E+01
STEEL	65	2.111000E+02	4.281011E+01	3.440612E-01	4.315417E+01
STEEL	66	2.119000E+02	3.809581E+01	3.654733E-01	3.846128E+01

STEEL	67	2.128000E+02	3.349258E+01	4.070587E-01	3.389964E-01
STEEL	68	2.137000E+02	2.896611E+01	4.857307E-01	2.945184E-01
CU	69	2.139000E+02	2.596591E+01	5.544300E-01	2.652034E+01
CU	70	2.141000E+02	2.444931E+01	5.837964E-01	2.505311E+01
CU	71	2.143000E+02	2.296630E+01	6.038084E-01	2.357011E+01
CU	72	2.145000E+02	2.144267E+01	6.133354E-01	2.205600E+01
CU	73	2.147000E+02	1.988039E+01	6.086751E-01	2.048906E+01
CU	74	2.149000E+02	1.825799E+01	5.834261E-01	1.884142E+01
INPL	75	2.190000E+02	1.525858E+01	5.558710E-01	1.581445E+01
INPL	76	2.231000E+02	1.182862E+01	5.343585E-01	1.236498E+01
INPL	77	2.271000E+02	9.601061E+00	5.195194E-01	1.012058E+01
INPL	78	2.312000E+02	7.864098E+00	5.037639E-01	8.369862E+00
INPL	79	2.353000E+02	6.490300E+00	4.880837E-01	6.978384E+00
INPL	80	2.393000E+02	5.379474E+00	4.719472E-01	5.851421E+00
INPL	81	2.434000E+02	4.467298E+00	4.551488E-01	4.922447E+00
INPL	82	2.475000E+02	3.714431E+00	4.377018E-01	4.152133E+00
INPL	83	2.515000E+02	3.087947E+00	4.197362E-01	3.507684E+00
INPL	84	2.556000E+02	2.566127E+00	4.014585E-01	2.967585E+00
INPL	85	2.596000E+02	2.130147E+00	3.830915E-01	2.513238E+00
INPL	86	2.637000E+02	1.766287E+00	3.648418E-01	2.131129E+00
INPL	87	2.678000E+02	1.462476E+00	3.468949E-01	1.809371E+00
INPL	88	2.718000E+02	1.209275E+00	3.294009E-01	1.538676E+00
INPL	89	2.759000E+02	9.983627E-01	3.124841E-01	1.310847E+00
INPL	90	2.800000E+02	8.229589E-01	2.962168E-01	1.119176E+00
INPL	91	2.840000E+02	6.771153E-01	2.806592E-01	9.577744E-01
INPL	92	2.881000E+02	5.558989E-01	2.658375E-01	8.217363E-01
INPL	93	2.922000E+02	4.549798E-01	2.517496E-01	7.067295E-01
INPL	94	2.962000E+02	3.707626E-01	2.383897E-01	6.091523E-01
INPL	95	3.003000E+02	3.000264E-01	2.257357E-01	5.257621E-01
INPL	96	3.043000E+02	2.401485E-01	2.137570E-01	4.539035E-01
INPL	97	3.084000E+02	1.885783E-01	2.024161E-01	3.909944E-01
INPL	98	3.125000E+02	1.434347E-01	1.916437E-01	3.350984E-01
INPL	99	3.165000E+02	1.021787E-01	1.813837E-01	2.835624E-01
STEEL	100	3.167000E+02	7.891373E-02	1.693251E-01	2.482388E-01
STEEL	101	3.169000E+02	7.286919E-02	1.572996E-01	2.301688E-01
STEEL	102	3.170000E+02	6.677305E-02	1.461666E-01	2.129397E-01
STEEL	103	3.172000E+02	6.025033E-02	1.347383E-01	1.949887E-01
AIR	104	3.178000E+02	5.682673E-02	1.289037E-01	1.857304E-01
AIR	105	3.184000E+02	5.680893E-02	1.288926E-01	1.857015E-01
AIR	106	3.433000E+02	5.645426E-02	1.286494E-01	1.851237E-01
AIR	107	3.682000E+02	5.576975E-02	1.282347E-01	1.840065E-01
AIR	108	3.931000E+02	5.509818E-02	1.278063E-01	1.829045E-01
AIR	109	4.179000E+02	5.443842E-02	1.273780E-01	1.818164E-01
AIR	110	4.428000E+02	5.378945E-02	1.269518E-01	1.807413E-01
AIR	111	4.677000E+02	5.315012E-02	1.265273E-01	1.796777E-01
AIR	112	4.926000E+02	5.251943E-02	1.261053E-01	1.786247E-01
AIR	113	5.174000E+02	5.189655E-02	1.256850E-01	1.775816E-01
AIR	114	5.423000E+02	5.157910E-02	1.254708E-01	1.770499E-01
AIR	115	5.672000E+02	5.156345E-02	1.254401E-01	1.770235E-01

APPENDIX 5.5.4
QAD-CGGP Input Files

- 5.5.4.1 PWR Axial/Fuel Region
- 5.5.4.2 PWR Axial/Top Nozzle Region
- 5.5.4.3 BWR Axial/Fuel Region

(The BWR source strength was doubled to account for activation products in the end fittings. The PWR analysis and other studies verify that the dose rate from the end fittings is approximately 50% of the dose rate from the active fuel region.)

(PWRFUEL) G DOSE, with IMPACT LIMITER, 9/12/89, FUEL, AXIAL

```

17 34 10 11 7 99 13 99 0 1 1 0 0 0 0 0
5.555E+6 0.0 0.0 0.0 0.0 0.0 0.0
0.0 5.0 10.0 15.0 20.0 25.0 30.0 35.0
40.0 45.0 50.0 55.0 60.0 62.5 65.0 67.5
70.0 72.39
0.00 30.00 60.00 90.00 120.00 150.00 180.00 205.00
215.00 225.00 235.00 245.00 255.00 265.00 275.00 285.00
295.00 300.00 305.00 310.00 315.00 320.00 325.00 330.00
334.00 335.00 336.00 337.00 338.00 339.00 340.00 341.50
342.50 343.00 343.40
0.0000 0.3142 0.6283 0.9425 1.2566 1.5808 1.8850 2.1991
2.5133 2.8274 3.1416
0 0 OCRM 21 PWR ELEMENT AXIAL DOSE RATE
RCC 1 0.00 0.00 0.00 0.00 0.00 0.00 343.40
72.39
RCC 2 0.00 0.00 343.40 0.00 0.00 0.00 33.40
72.39
RCC 3 0.00 0.00 376.80 0.00 0.00 0.00 1.27
72.39
RCC 4 0.00 0.00 378.07 0.00 0.00 0.00 6.35
72.39
RCC 5 0.00 0.00 384.42 0.00 0.00 0.00 21.59
72.39
RCC 6 0.00 0.00 406.01 0.00 0.00 0.00 1.27
72.39
RCC 7 0.00 0.00 407.28 0.00 0.00 0.00 0.64
72.39
RCC 8 0.00 0.00 407.92 0.00 0.00 0.00 101.60
72.39
RCC 9 0.00 0.00 -999.00 0.00 0.00 0.00 9999.00
9999.00
END
FUL 1
TNZ 2
BSI 4
IRN OR 3OR 5OR 7
COP 6
IMP 8
AIR 9 -8 -7 -6 -5 -4 -3 -2 -1
END
1 2 3 4 5 6 7
1 2 3 4 5 6 7
99 1 5 6 8 13 14 26 29 40 92 7
IRON ABS
0.000 0.000 0.000 0.171 0.000 0.000 .6242 .1919 0.394 1.271 0.0
0.000 0.000 0.000 0.000 0.000 0.000 0.228 0.000 0.000 0.000 0.0
.0752 .0169 .1716 .7443 .2999 .2789 .0003 0.000 0.000 0.000 0.0
0.000 0.000 0.000 0.000 0.000 0.000 7.870 0.000 0.000 0.000 0.0
0.000 0.000 0.000 0.000 0.000 0.000 0.000 8.960 0.000 0.000 0.0
0.014 0.000 .1242 0.044 0.000 0.000 0.000 0.000 0.000 0.000 0.0
0.00 0.0 0.0 3.23-4 0.0 0.0 0.0 0.0 0.0 0.0 9.70-4
5.0 3.5 2.8 2.25 1.75 1.25
0.85 .575 .375 .225 .125 .0575
0.01
1.11E+1 8.44E+2 7.05E+3 1.11E+5 5.31E+6 4.04E+8
4.54E+8 5.99E+9 1.69E+8 3.53E+8 3.87E+8 1.14E+9
5.31E+9
.001101 .001251 .001352 .001471 .001593 .001781
.001928 .002033 .002087 .002074 .002048 .003138
.3449
0.0 0.0 0.0 0.0 0.0 0.0
0.0 0.0 0.0 0.0 0.0 0.0
0.0
0.01-6.5
5.0-6.5 3.0-4.0 2.5-3.0 2.0-2.5 1.66-2.0 1.0-1.33
0.8-1.00 0.4-0.6 0.3-0.4 0.2-0.3 0.16-0.2 0.05-0.1
0.01-0.05
PHOTONS/CM^2-SEC HREM/HR W/G
0.00 508.73 0.0 1 0 0 0
0.00 708.73 0.0 1 0 0 0
0.00 0.0 0.0 -1 0 0 0

```

(PWRTPN2), G DOSE, with IMPACT LIMITER, 9/12/89, ACT.PROD TOP NO2, AXIAL

17 24 10 10 6 99 1 99 0 1 1 0 0 0 0 0

5.403E+5 0.0 0.0 0.0 0.0 0.0 0.0

0.0 5.0 10.0 15.0 20.0 25.0 30.0 35.0

40.0 45.0 50.0 55.0 60.0 62.5 65.0 67.5

70.0 72.39

0.00 3.00 6.00 9.00 12.00 15.00 18.00 20.50

21.50 22.50 23.50 24.50 25.50 26.50 27.50 28.50

29.50 30.00 30.50 31.00 31.50 32.00 32.50 33.00

33.40

0.0000 0.3142 0.6283 0.9425 1.2566 1.5808 1.8850 2.1991

2.5133 2.8274 3.1416

0 0 21 PWR OCRM CSK 35KBUP 10YEAR

RCC 1 0.00 0.00 0.00 0.00 0.00 33.40

RCC 2 0.00 0.00 33.40 0.00 0.00 1.27

RCC 3 0.00 0.00 34.67 0.00 0.00 6.35

RCC 4 0.00 0.00 41.02 0.00 0.00 21.59

RCC 5 0.00 0.00 62.61 0.00 0.00 1.27

RCC 6 0.00 0.00 63.88 0.00 0.00 0.64

RCC 7 0.00 0.00 64.52 0.00 0.00 101.60

RCC 8 0.00 0.00 -999.00 0.00 0.00 9999.00

9999.00

END

TNZ 1

BSI 3

IRN OR 2OR 4OR 6

COP 5

IMP 7

AIR 8 -7 -6 -5 -4 -3 -2 -1

END

1 2 3 4 5 6

1 2 3 4 5 6

99 1 5 6 7 8 11 13 14 26 29

IRON ABS

0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.228 0.000

.0752 .0169 .1716 0.000 .7443 0.001 .2999 .2789 .0003 0.000

0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 7.876 0.000

0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 8.940

0.014 0.000 .1262 0.018 0.044 0.000 0.000 0.000 0.000 0.000

0.0 0.0 0.0 9.7-4 3.23-4 0.0 0.0 0.0 0.0 0.0

1.25

4.16E+7

.001496

0.0

1.0-1.33

1.0-1.33

PHOTONS/CM²-SEC

0.00 166.12 0.0 1 0 0 0

0.00 346.12 0.0 1 0 0 0

0.00 0.0 0.0 -1 0 0 0

MREM/HR

W/G

6.0 CRITICALITY EVALUATION

The NuPac 140-B Cask and Fuel Package complies with the requirements of 10 CFR 71.55 and 71.61 (Reference 6.6.1.1) for a Fissile Class III Package for the contents described in Section 6.2. This section describes the basis for establishing the NuPac 140-B Cask and Fuel Package as a Fissile Class III package.

6.1 Discussion and Results

The NuPac 140-B Cask has been designed to transport irradiated reactor fuel assemblies from both pressurized water reactors (PWR) and boiling water reactors (BWR). Section 1.2 contains a description of the NuPac 140-B Cask, and Appendix 1.4.1 contains the Packaging General Arrangement Drawings. The NuPac 140-B Cask utilizes interchangeable inserts or baskets in the cask cavity for fuel assembly support. The cask capacity is 21 PWR or 52 BWR fuel assemblies. The basket for PWR assemblies achieves criticality control through the use of plates made of boron-copper material. The poison plates are sandwiched between two sheets of stainless steel that enclose a basket cell. They are a permanent part of the PWR basket structure. The basket for BWR assemblies uses a similar structural arrangement, but uses sheets of borated stainless steel for the neutron poison material.

The value of k_{eff} (the neutron multiplication factor) for 21 PWR fuel assemblies was calculated from two different approaches. The first approach was to determine the maximum enrichment of U^{235} in fresh (unirradiated) fuel to maintain k_{eff} less than 0.95. The second approach examined what minimum burnup would be required of 4.5% enriched fuel to maintain k_{eff} less than 0.95.

The maximum k_{eff} , with 21 PWR fuel assemblies containing 3.2% enriched fresh fuel is 0.928¹. For 3.4% enriched fresh fuel, k_{eff} is also expected to be less than 0.95; this case will be examined during final design. The maximum k_{eff} for 21 PWR fuel assemblies with 4.5% enrichment and 18,000 (MWD/MTU (megawatt days per metric tons of uranium) burnup, is 0.944².

The maximum k_{eff} is 0.887 for an infinite array of casks with 52 BWR fuel assemblies. All k_{eff} values were determined with the most reactive fuel optimally moderated and reflected. The requirements of 10 CFR 71.55 are satisfied when the contents are limited to the types of fuel assemblies identified in Section 6.2.

An array of two undamaged NuPac 140-B Casks with PWR fuel assemblies, as defined in Section 6.3, is also expected to be safely subcritical ($k_{eff} \leq 0.95$) due to the small interactive effects between undamaged casks. (The array calculations will be performed for the PWR fuel in the final design). Therefore, the requirements of 10 CFR 71.61 will be satisfied for both the normal and hypothetical accident conditions for both fuel types.

The results of the criticality calculations are summarized in Table 6.1-1. The maximum calculated k_{eff} value for a single NuPac 140-B Cask occurs when Westinghouse 15 x 15 array fuel assemblies are optimally moderated and fully reflected. For case 1 with 3.2% enriched fresh fuel, the maximum k_{eff} is 0.928. For case 2 with 4.5% enriched fuel and 18,000 MWD/MTU burnup, the maximum k_{eff} is 0.944.

¹All reported k_{eff} values include 2 σ for a 95% confidence level and a validation bias.

²This value was obtained from a Single Cell model (described in Section 6.3) with a correction factor to adjust to a Quarter Cask model.

TABLE 6.1-1
Criticality Summary for the NuPac 140-B Cask and Fuel Package
Fissile Class III

<u>Description</u>	Calculated k_{eff} Value	
	<u>Normal</u>	<u>Accident</u>
Single NuPac 140-B Cask with water ingress as required by 10 CFR 71.55		
o PWR fuel assemblies (case 1) ⁽¹⁾	0.928	0.928
o PWR fuel assemblies (case 2) ⁽²⁾	0.944	0.944
o BWR fuel assemblies ⁽³⁾	0.887	0.887
Array of two undamaged NuPac 140-B Casks as required by 10 CFR 71.61		
o PWR fuel assemblies	0.928 ⁽⁴⁾	N/A
o BWR fuel assemblies ⁽³⁾	0.887	N/A

(1) (Case 1) 3.2% enriched fresh (unirradiated) fuel.

(2) (Case 2) 4.5% enriched fuel with 18,000 MWD/MTU burnup and 10 years decay.

(3) Fully reflected cask.

(4) Any increase in k_{eff} is expected to be negligible due to the small interactive affects between undamaged casks.

6.2 Package Fuel Loading

The payload cavity of the NuPac 140-B Cask can accommodate a basket for either 21 PWR or 52 BWR spent fuel assemblies. Canisters of consolidated fuel with the same external cross-section as the intact PWR and BWR fuel assemblies are also expected to be an acceptable payload for the cask. The reactivity of the payload is expected to decrease when the fuel to water ratio increases as compared to the analyses for an intact fuel assembly.

The fuel assemblies listed in Table 6.2-1 were evaluated to determine the assembly of each fuel type with the highest reactivity in the cask and basket. The criticality evaluation for the PWR fuel was performed on the Westinghouse 15 x 15 array assembly. The characteristics for this fuel assembly are provided in Table 6.2-2. The criticality evaluation for the BWR fuel was performed on the General Electric 8 x 8R array assembly. The characteristics of this fuel assembly are provided in Table 6.2-3. Section 6.4.2 explains the basis for the selection of the PWR and BWR fuel assembly type for the criticality evaluation.

TABLE 6.2-1
Fuel Assemblies Acceptable for Transport in
the NuPac 140-B Cask

Standard PWR Spent Fuels

Westinghouse Electric	17 x 17
Westinghouse Electric	15 x 15
Westinghouse Electric	14 x 14
Babcock & Wilcox	17 x 17
Babcock & Wilcox	15 x 15
Combustion Engineering	16 x 16
Combustion Engineering	15 x 16
Combustion Engineering	15 x 15
Combustion Engineering	14 x 14
Exxon Nuclear	17 x 17
Exxon Nuclear	15 x 16
Exxon Nuclear	15 x 15
Exxon Nuclear	14 x 14

Standard BWR Spent Fuels

General Electric	8 x 8
General Electric	7 x 7
General Electric	6 x 6
Exxon Nuclear	10 x 10
Exxon Nuclear	9 x 9
Exxon Nuclear	8 x 8
Exxon Nuclear	7 x 7
Exxon Nuclear	6 x 6
Westinghouse	8 x 8

TABLE 6.2-2
Design Characteristics for
Westinghouse PWR Fuel

Fuel Assembly Type	15 x 15 STD/ZC PWR
Assembly Transverse Dimension (inches)	8.426
Maximum Enrichment (case 1)	3.2% U ²³⁵
Maximum Enrichment (case 2)	4.5% U ²³⁵
Minimum Burnup (case 2)	18,000 MWD/MTU
Fueled Rods/Assembly	204
Rod Pitch (inches)	0.563
Fuel Rod Diameter (inches)	0.422
Active Fuel Height (inches)	144.0
Clad Thickness (inches)	0.0243
Cladding Material	Zircaloy-4
Fuel Pellet Diameter (inches)	0.3659
Fuel Density	95% TD

TABLE 6.2-3
Design Characteristics for
General Electric BWR Fuel

Fuel Assembly Type	8 x 8R (two water rods)
Flow Channel Material	Zircaloy-4
Flow Channel Inside Width (inches)	5.278
Flow Channel Thickness (inches)	0.100
Maximum Enrichment	4.5% U ²³⁵
Fueled Rods/Assembly	62
Rod Pitch (inches)	0.640
Active Fuel Length (inches)	150.00
Cladding O.D. (inches)	0.483
Cladding I.D. (inches)	0.419
Cladding Material	Zircaloy-2
Fuel Pellet Diameter (inches)	0.416
Fuel Density	95% TD

6.3 Model Specification

Criticality calculations for the NuPac Model 140-B Cask and Fuel Package are performed using KENO-IV, a three dimensional Monte Carlo code (Reference 6.6.1.2). Descriptions of the calculational models are given in Section 6.3.1. A summary of the materials and atom densities used in the evaluation are included in Section 6.3.2.

6.3.1 Description of the Calculational Model

The multiplication factor, k_{eff} , for the limiting type of PWR or BWR fuel assembly that may be transported in a NuPac 140-B Cask for a Fissile Class III shipment is established from the results of KENO-IV calculations. As discussed in Section 6.1, an array of two undamaged NuPac 140-B Casks with PWR fuel assemblies is expected to be safely subcritical due to small interactive effects between casks. The k_{eff} of an infinite array of casks in an air/void volume may be approximated by the formula:

$$k_{eff(array)} = k_{eff(cask)} [1 + L_{cask}]$$

where L_{cask} is the leakage factor. Total neutron leakage from the PWR cask model was 3.263×10^{-5} ; which amounts to a negligible change in k_{eff} . The model of a single undamaged NuPac 140-B Cask was developed and utilized for the PWR fuel assemblies. For BWR fuel assemblies, the model was an infinite array of undamaged NuPac 140-B Casks.

All models use pure, unborated water for all void regions. No credit is taken for absorption in water impurities. These calculations assume a temperature of 20°C.

Pure elemental zirconium is assumed for the zircaloy material in the fuel rod cladding and flow channels. No credit is taken for additional neutron absorption in zircaloy impurities.

Pure elemental copper and lead are assumed for the copper material in the basket assembly and the lead shielding in the cask. This is a reasonable assumption due to the high purity of these items as they are commercially available.

Damage to the impact limiters on the cask ends may occur as a result of the hypothetical accident conditions- including the 30 foot drop, the puncture drop and the thermal transient. However, the cask criticality models for both PWR and BWR assemblies exclude the impact limiter structures or foam for normal and accident conditions. The analytical results reported in the structural evaluation of the NuPac 140-B Cask demonstrate that the cask body and basket assembly components experience no distortions under hypothetical accident conditions. Thus, both normal and accident condition geometries for the package are identical. Hypothetical accident condition analyses also assumes intact fuel assemblies as a result of the accident conditions. This is conservative since the normal pitch of a fuel assembly is at or near the maximum k_{eff} pitch (Ref. 6.6.1.11).

6.3.1.1 Models Used for PWR Fuel Assemblies in the Cask

The NuPac 140-B Cask with a basket for 21 PWR fuel assemblies was designed to maintain a calculated k_{eff} plus biases and uncertainties to be less than 0.95. This was accomplished by maintaining sufficient space between fuel assemblies to thermalize the neutrons so that an adequate number are absorbed by a boron poison before they reach adjacent fuel assemblies. Each PWR fuel assembly is enclosed in a basket cell within the cask body. Each basket cell consists of a copper and boron alloy plate sandwiched between two sheets of stainless steel. The cell pitch is 9.85 inches, with a 0.59 inch gap separating the PWR basket cells. The gap contains a corrugated sheet of stainless steel. A stainless steel cruciform is located at the intersection of adjacent PWR basket cells. The PWR basket is assumed flooded with water in the model. A sketch of

the basket design is provided in Figure 6.3-1³. The dimensions of the PWR basket and a list of materials are provided in Table 6.3-1. The PWR basket is positioned in the center of the cask body which consists of layers of stainless steel plate, lead, and a Boro-Silicone neutron/gamma shielding material. A sketch of the cask body design is provided in Figure 6.3-2. The dimensions of the cask body and a list of the materials in the cask body are provided in Table 6.3-2.

The poison contained in the PWR basket cells is a mixture of boron (1.58W%), carbon (0.53W%) and copper (97.89W%). The boron is enriched to 95 atom percent with boron-10. The cask is assumed to be fully submerged in water and flooded. Water occupies all open areas in the cask cavity. These areas include the corrugated channel between adjacent cells, the hollow tube in the cruciform, and the unoccupied sealed regions near the outer basket shell. The corrugated stainless steel in the gaps between adjacent PWR basket cells are homogenized with the water.

The copper fins in the NuPac 140-B Cask are homogenized with the Boro-Silicone material. The thickness of the Boro-Silicone material and the outer stainless steel varies. Both have a maximum thickness at the center line of the cask and a minimum thickness at the ends. The cask was modeled to reflect the maximum number of neutrons back into the PWR basket. This is accomplished using the maximum value of thickness for the reflecting stainless steel material and the minimum value of thickness for the absorbing Boro-Silicone material. A twelve inch thick layer of water was placed around the cask to simulate the cask being submerged in water for a single cask evaluation after the hypothetical accident conditions.

³Minor changes in some dimensions have occurred since the criticality analysis. These dimensions are shown in parenthesis in all figures.

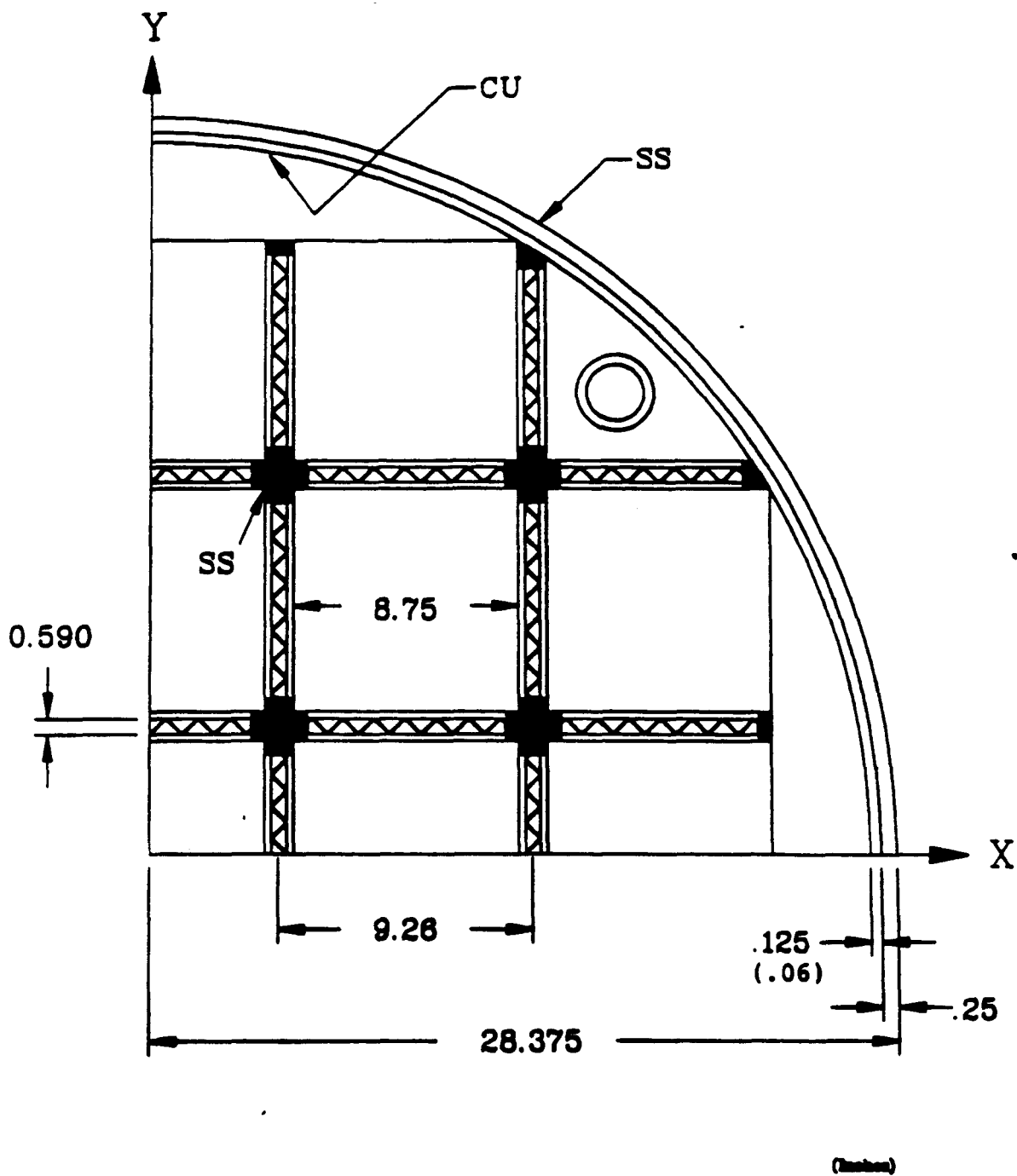


FIGURE 6.3-1
21 PWR Basket Design (Not drawn to scale)

TABLE 6.3-1
Dimensions and Materials Modeled in the PWR Fuel Assembly Basket

	<u>Dimensions</u>	
	<u>(inches)</u>	<u>(cm)</u>
Fuel assembly (cross section)	8.426	21.4020
Basket cell (cross section)	8.750	22.2250
Stainless steel inner plate to basket cell (thickness)	0.030	0.0762
Copper/Boron (thickness)	0.165	0.4191
Stainless steel outer plate to basket cell (thickness)	0.060	0.1524
Water gap between outer plates of adjacent basket cells	0.590	1.4986
Corrugated stainless steel in water gap (thickness)	0.035	0.0889
Cruciform extension into water channel	0.750	1.9050
Radius in cruciform	0.1875	0.47625
Water channel thickness between basket outer cell and copper liner to basket	3.570	9.0678
Copper liner to basket (thickness) (actual thickness is 0.06 inches)	0.125	0.3175
Stainless steel liner to basket (thickness)	0.250	0.6350

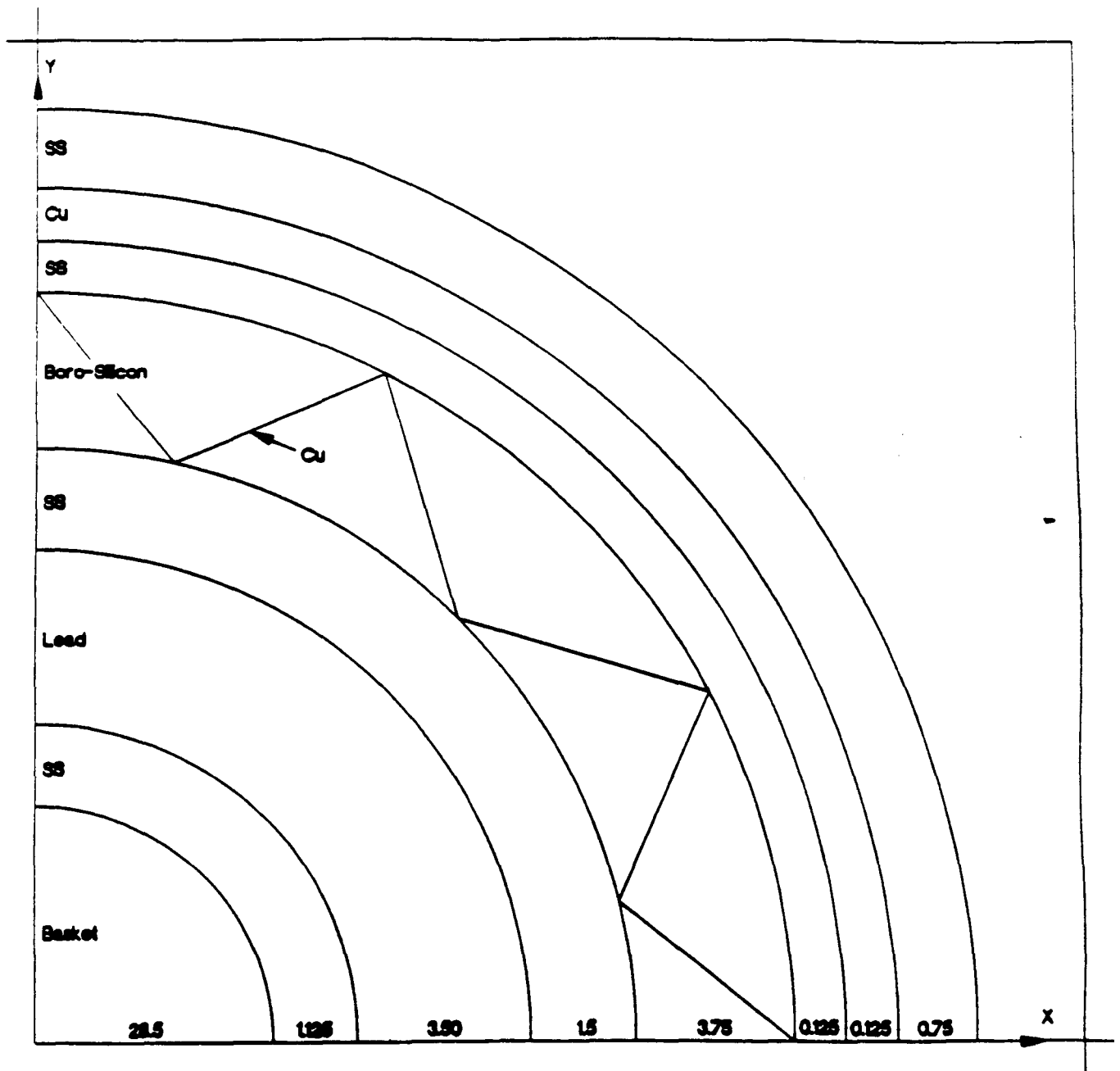


FIGURE 6.3-2
NuPac 140-B Cask Dimensions Used in PWR Criticality Model
(Not Shown to scale)

TABLE 6.3-2
Dimensions and Materials for the NuPac 140-B Cask Body

	Dimensions	
	<u>(inches)</u>	<u>(cm)</u>
Inner stainless steel liner to cask (thickness)	1.125	2.8575
Lead liner to cask (thickness)	3.500	8.8900
Middle stainless steel liner to cask (thickness)	1.500	3.8100
Boro-Silicone liner to cask		
(maximum thickness)	5.880	14.935
(minimum thickness)	3.750*	9.525
Stainless steel liner adjacent to Boro-Silicone material	0.125	0.3175
Copper wire (Current cask uses 0.134 inches of stainless steel wire followed by a 0.125 inch stainless steel liner)	0.125	0.3175
Outer stainless steel liner to cask		
(maximum thickness)	0.750*	1.905
(minimum thickness)	0.250	0.635

*The maximum B-Si and the minimum stainless steel values were used in the PWR model for conservatism.

Two models were used to evaluate the criticality of the PWR basket design. The first model, called the Single Cell Model, simulates one PWR basket cell and half the space to the adjacent cell, with specular albedo conditions specified on all six sides. This model is shown in Figure 6.3-3. This model yielded a conservative criticality result and allowed design changes to be made quickly and efficiently to evaluate their sensitivity to reactivity changes.

The second model, called the Quarter Cask Model, simulates a quarter section of the cask and basket. Figure 6.3-4 illustrates the basket model including 10 box types. Figure 6.3-5 illustrates the cuboid model of the cask body. Neutrons exiting this model along the x and y axis center lines were constrained to be reflected into the quarter basket and cask. This model allowed more detail in the criticality model. The Single Cell Model was used to determine the most efficient design, then the Quarter Cask Model was used to verify the criticality results.

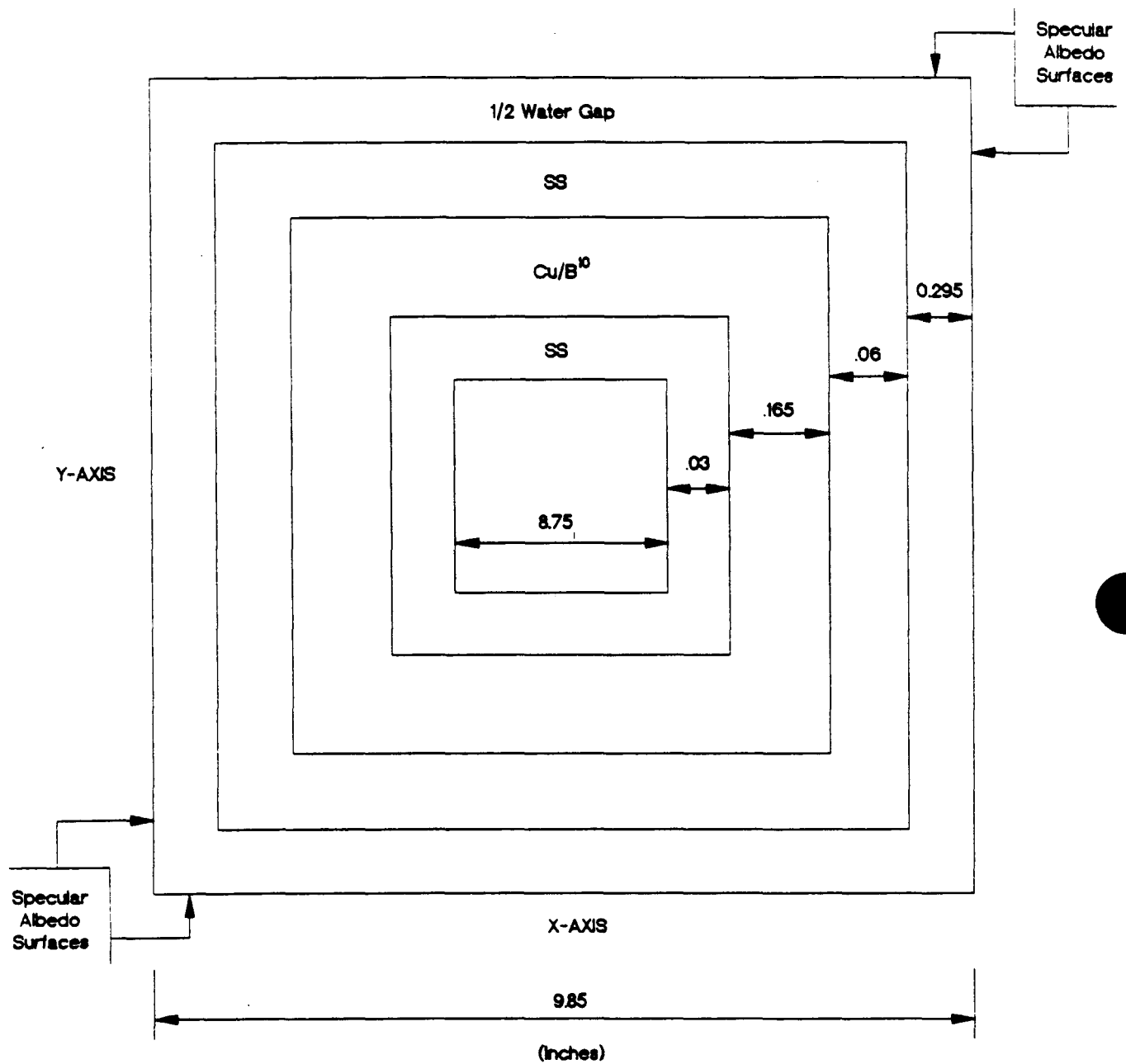


FIGURE 6.3-3
PWR Basket Single Cell Model
(Not drawn to scale)

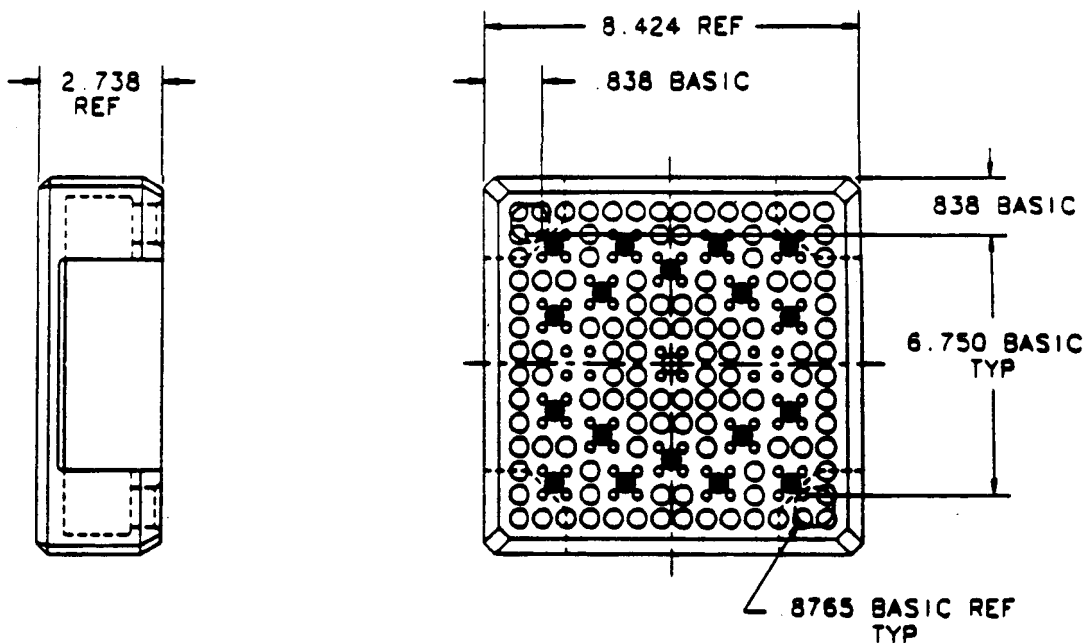


FIGURE 6.3-3A

Westinghouse 15x15 PWR
Fuel Assembly Dimensions

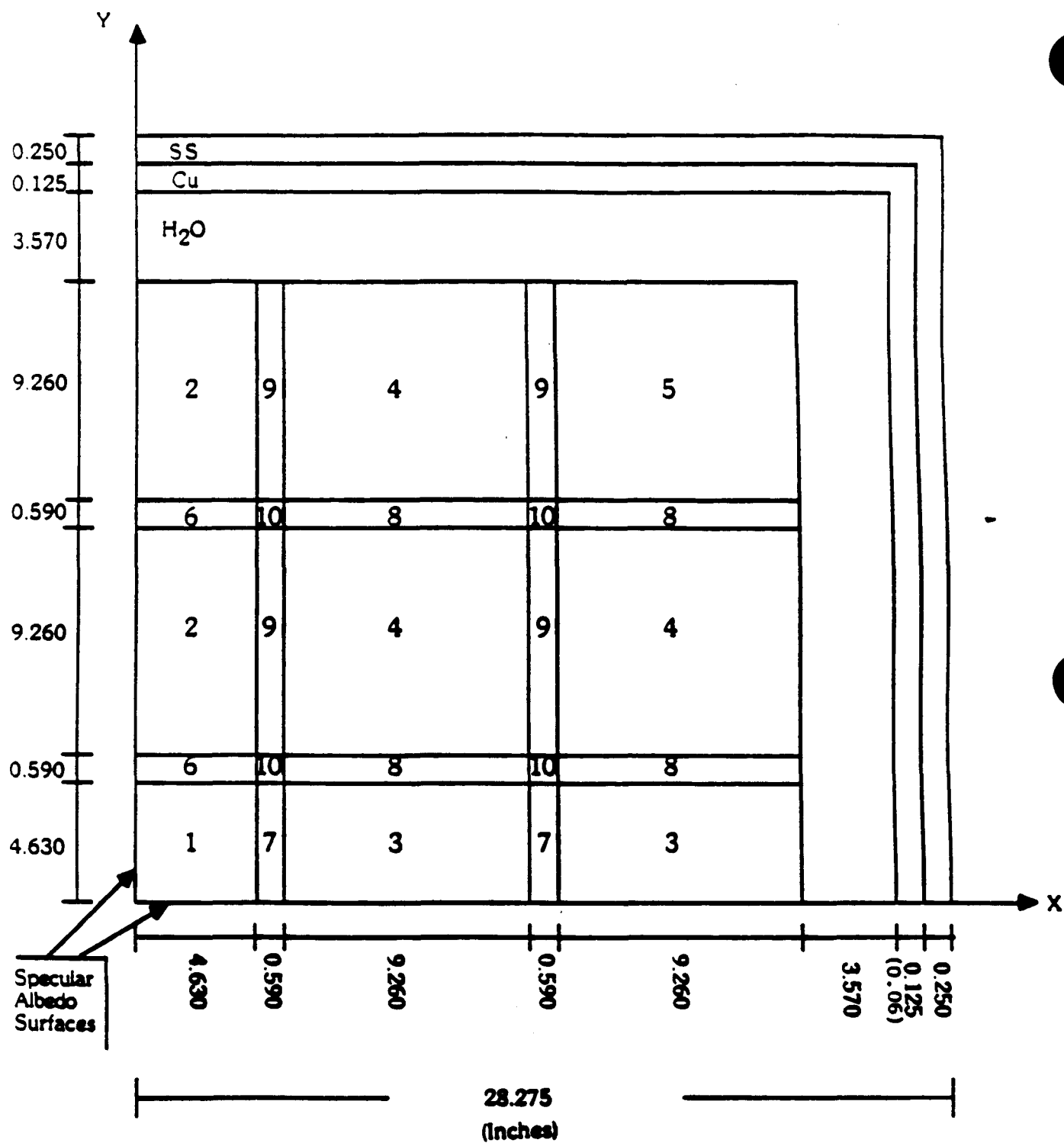


FIGURE 6.3-4
Quarter Cask Model of the PWR Basket
(Basket Box Types)

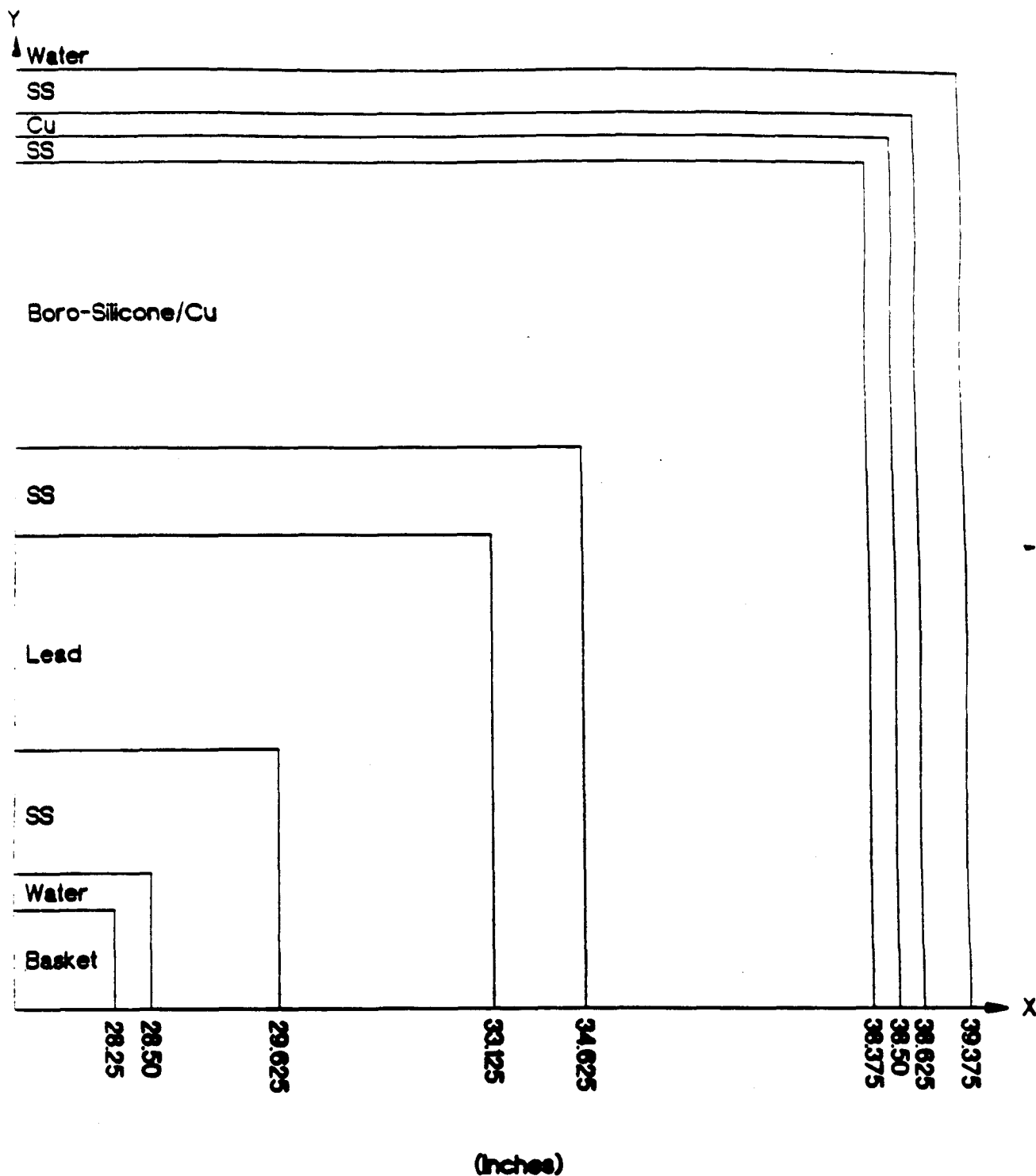


FIGURE 6.3-5
Quarter Cask Model of the NuPac 140-B Cask Body
(Cask Cuboids)

6.3.1.2 Models Used for BWR Fuel Assemblies in the Cask

The NuPac 140-B Cask with a basket for 52 BWR fuel assemblies was designed to maintain a calculated k_{eff} plus biases and uncertainties to be less than 0.95. This was accomplished by providing sufficient neutron poison materials between adjacent fuel assemblies. The basket design was evaluated for fresh (unirradiated) BWR fuel with a maximum U^{235} enrichment of 4.5 percent in whole (unconsolidated) assemblies. Each BWR fuel assembly is enclosed in a basket cell within the cask body. Each BWR basket cell consists of a stainless steel and boron plate sandwiched between layers of copper and stainless steel plate. The pitch of the BWR basket cells is 6.26 inches. A stainless steel cruciform is located at the intersection of adjacent BWR basket cells. The BWR basket is assumed flooded in the model. A sketch of the BWR basket design is provided in Figure 6.3-6. The BWR basket is in the center of the cask body shown previously in Figure 6.3-2. Three different computer models were developed to calculate the reactivity of the shipping cask with 52 BWR fuel assemblies in the basket.

The heterogeneous pin infinite array model was constructed to closely reproduce an infinite array of General Electric 8 x 8R BWR fuel assemblies separated by a poison basket assembly. The model included a discrete representation of the fuel pellet, zircaloy cladding, water rods, zircaloy flow channel, and surrounding water.

The homogeneous pin infinite array model reproduced the same fuel and BWR basket geometric configuration, however smeared materials and cross sections were used in the core zone of the fuel assembly. This model was used to benchmark the homogenization technique used in the cask model.

The third computer model was constructed to calculate the reactivity of an infinite array of NuPac 140-B shipping casks, each loaded with 52 BWR fuel assemblies. Homogenized materials and cross section were used in the core regions of the BWR fuel assemblies. Reflector regions were included to represent the shipping cask and a thin layer of water outside each cask.

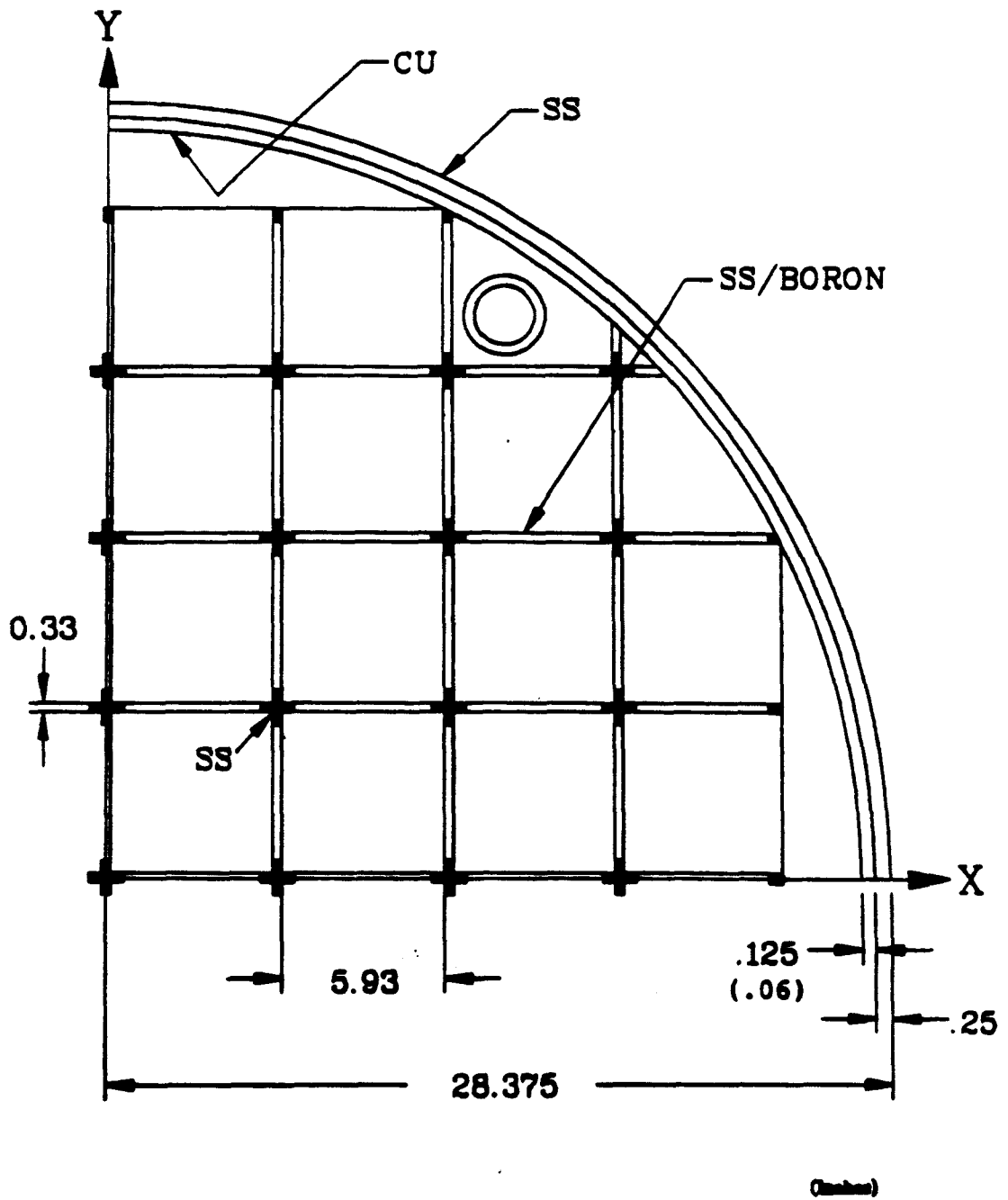


FIGURE 6.3-6
BWR Basket Design for the NuPac 140-B Cask
(Not drawn to scale)

In all three models, minor structural elements such as fuel assembly grid straps and small items of BWR basket hardware are neglected. They are instead replaced by water in the models. This is customary due to the difficulties in modeling them, and conservative since such materials generally absorb rather than moderate neutrons.

All models are constructed using nominal BWR basket cell dimensions with BWR fuel assemblies centered in their storage locations. No allowances for fabrication tolerances, manufacturing tolerances on the poison material, off-centered fuel loading, or accident conditions have been analyzed at this time for the shipping cask with BWR fuel assemblies.

The design basis fuel for all three KENO-IV models was fresh General Electric 8 x 8R fuel. No credit was taken for fuel burn up. No credit was taken for burnable poison, or for the axial and radial variations in fuel enrichment typically found in BWR fuel. All fuel rods are assumed to be uniformly enriched with 4.5% U^{235} .

6.3.1.2.1 Heterogeneous Pin Infinite Array

The heterogeneous pin infinite array model closely represents an infinite array of design basis fuel assemblies surrounded by the BWR basket poison assembly. The KENO-IV model shown in Figure 6.3-7 illustrates the overall model including two box types. Figure 6.3-8 shows details of box type 1 (fuel rods) and box type 2 (water rods).

Specular albedo conditions were specified for all six sides of the model. An arbitrary length was chosen for the model's geometrical elements. This results in a rectangular infinite array model with an infinite length.

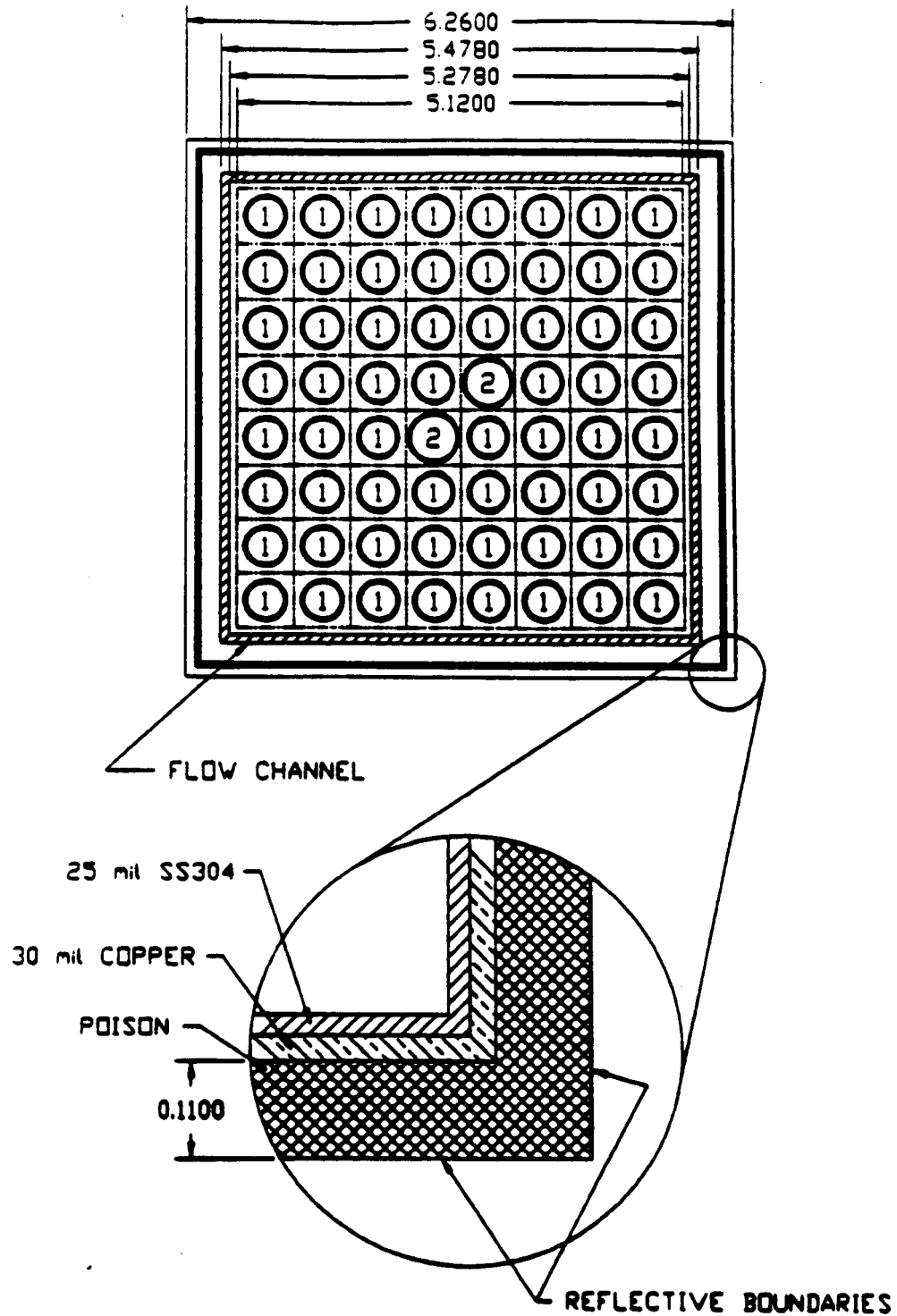


FIGURE 6.3-7
Heterogeneous Infinite Array KENO Model
for BWR Fuel

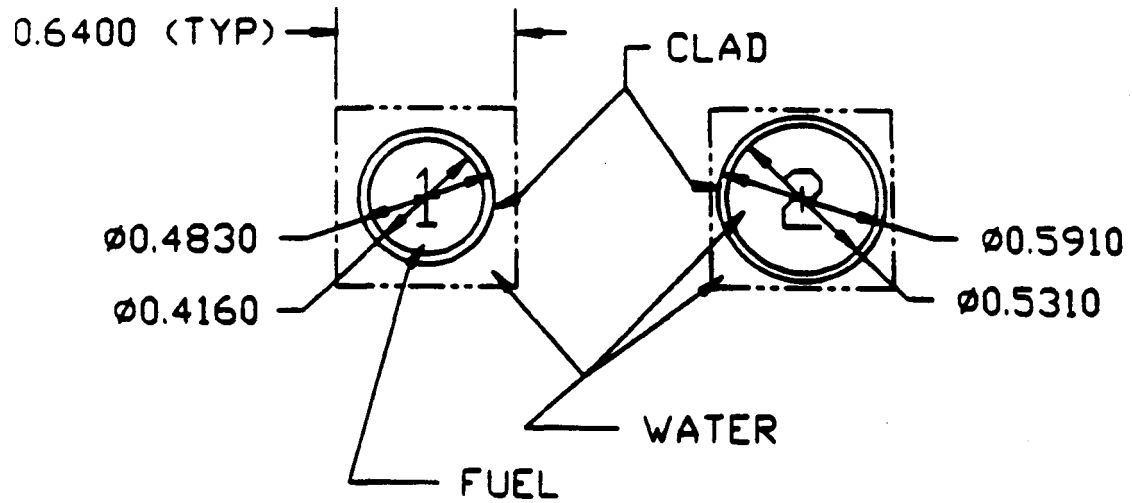


FIGURE 6.3-8
Heterogeneous Infinite Array KENO Model
Box Type Details

The core region of the model contained 64 cells, as indicated in Figure 6.3-7. Surrounding the core, a water region, flow channel region, and another water region were included. The poisoned basket assembly surrounds the outermost water region. The model is in a half-symmetrical fashion with a reflective boundary condition.

6.3.1.2.2 Homogeneous Pin Infinite Array

This KENO-IV model is identical to the heterogeneous infinite array model except that the core zone, which was made up of 64 discrete boxes in the heterogeneous model, was replaced by a single zone containing homogenized fuel. This model is necessary to benchmark the homogenization process since the cask model was executed using homogenized cross sections. Figure 6.3-9 shows the homogenized infinite array model dimensions and materials.

6.3.1.2.3 Array of Casks with BWR Fuel Assemblies

This model represents an infinite array of closely spaced NuPac 140-B Casks each loaded with 52 design basis BWR fuel assemblies. Since the BWR basket is symmetrical about two of its axes, one quarter of the cask and basket was modeled and reflective boundary conditions were specified on all six faces. As in the infinite array models, the specification of a specular albedo on the z-axis of the model results in an infinitely long fuel assembly. This is conservative since end leakage is ignored.

Figure 6.3-10 shows the general arrangement of the cask model. The core area is modeled using ten box types in a 13 x 13 array. Box type one is a homogenized fuel assembly, a water layer, the zircaloy flow channel, and another water layer. Since the actual basket will not have neutron poison throughout the corners, no credit for neutron poison has been taken there. Box type seven is a dummy box which is filled with water and has the same dimensions as box type one.

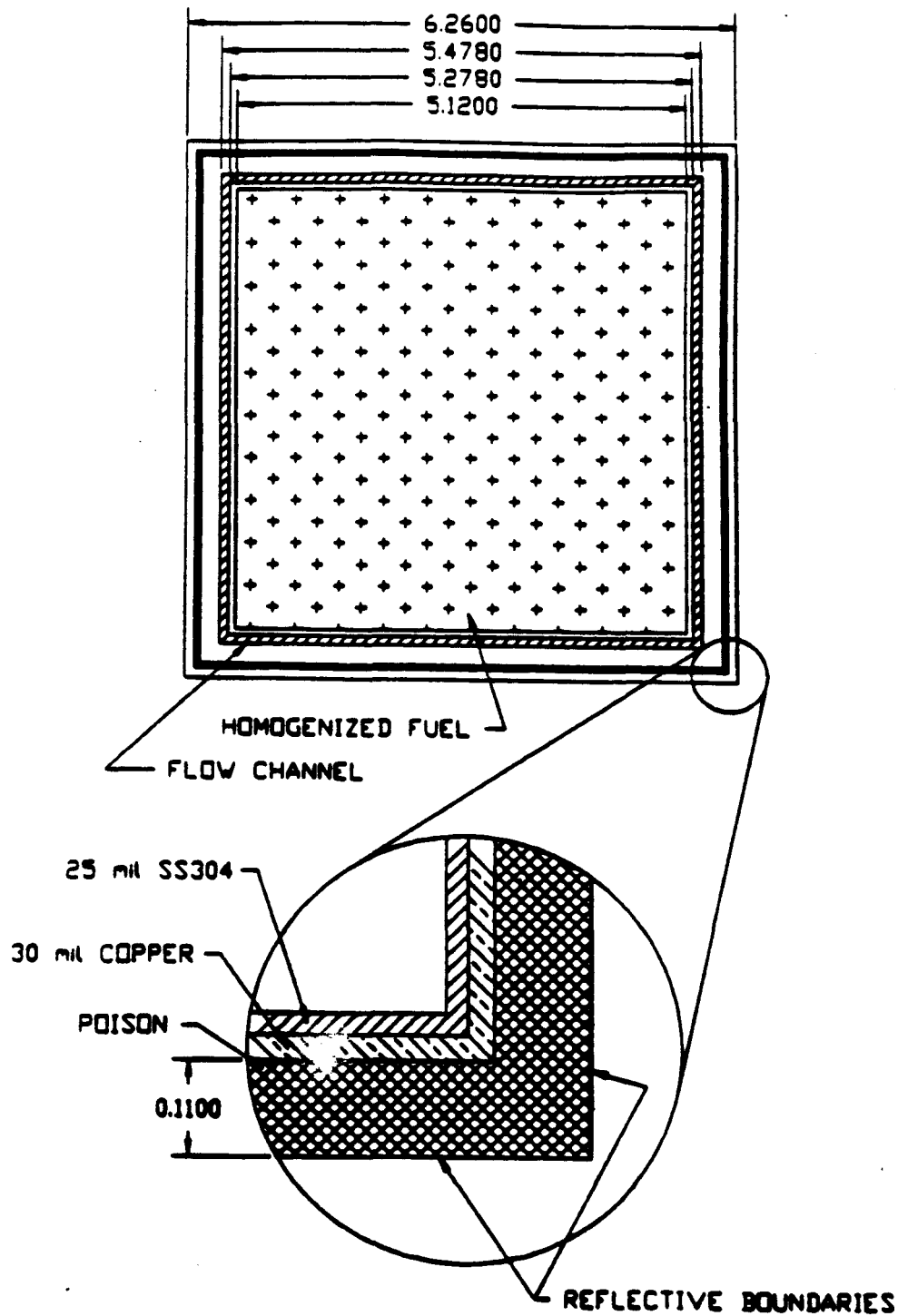


FIGURE 6.3-9
Homogeneous Infinite Array KENO Model
for BWR Fuel

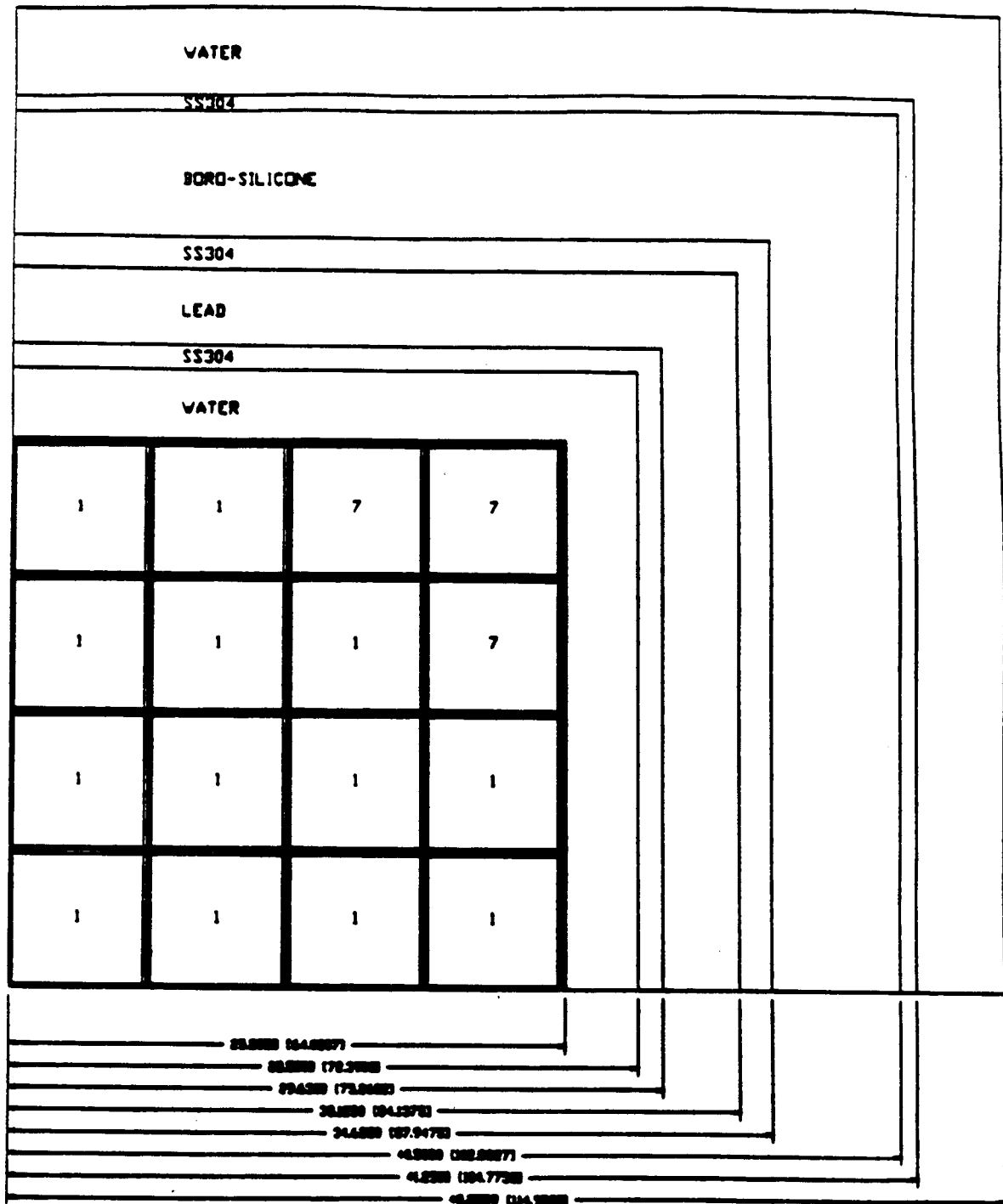


FIGURE 6.3-10
 Array of Casks KENO Model
 General Model Arrangement

6.3.2 Package Regional Densities

A summary of the material compositions used in the criticality evaluations of the NuPac 140-B Cask and Fuel Package is given in Section 6.3.2.1 for the PWR basket design and Section 6.3.2.2 for the BWR basket design. Different sources were used to determine the material densities in the PWR and BWR analysis. Slight differences exist between the values used in the two models.

6.3.2.1 Package Regional Densities for the PWR Basket and Fuel Assemblies

Table 6.3-3 contains a summary of the atom densities for the materials included in the Single Cell Model for the PWR criticality evaluations.

Table 6.3-4 contains a summary of the atom densities for the materials included in the Quarter Cask Model for the PWR criticality evaluations.

6.3.2.2 Package Regional Densities for the BWR Basket and Fuel Assemblies

Table 6.3-5 contains a summary of the atom densities for the materials included in the heterogeneous BWR fuel pin infinite array KENO-IV model.

Table 6.3-6 contains a summary of the atom densities for the materials included in the homogenized infinite array KENO-IV model.

Table 6.3-7 contains a summary of the atom densities for the materials included in the homogenous infinite array KENO-IV model for NuPac 140-B Casks with PWR fuel.

TABLE 6.3-3
Regional Densities for the Single Cell Model
for PWR Fuel

<u>Mixture</u>	<u>Nuclide</u>	<u>Atom Density</u> <u>(Atoms/barns - 'cm)</u>
Fuel Region	Hydrogen	3.9965×10^{-2}
	Oxygen	1.9983×10^{-2}
	Uranium-238	6.7952×10^{-3}
	Uranium-235	2.2751×10^{-4}
	Oxygen	1.4041×10^{-2}
	Zirconium	3.7115×10^{-3}
Water	Hydrogen	3.3427×10^{-2}
	Oxygen	6.6854×10^{-2}
Stainless Steel	Chromium	1.7630×10^{-2}
	Iron ¹	6.2361×10^{-2}
	Nickel	7.3296×10^{-3}
Poison	Boron ²	7.7104×10^{-3}
	Carbon ²	2.2598×10^{-3}
	Copper ³	7.9254×10^{-2}
Water Gaps (Water and Stainless Steel)	Chromium	1.4791×10^{-3}
	Iron ¹	5.2321×10^{-3}
	Nickel	6.1495×10^{-4}
	Hydrogen	6.1245×10^{-2}
	Oxygen	3.0622×10^{-2}

- 1) All other elements are added to iron.
- 2) The density of aluminum (2.699 grams/cm³) is used to calculate this atom density.
- 3) The density of copper (8.96 grams/cm³) is used to calculate this atom density.

TABLE 6.3-4
Regional Densities for the Quarter Cask Model
for PWR Fuel

<u>Mixture</u>	<u>Nuclide</u>	<u>Atom Density</u> <u>(Atoms/barns - cm)</u>
Fuel Region	Hydrogen	3.9965×10^{-2}
	Oxygen	1.9983×10^{-2}
	Uranium-238	6.7952×10^{-3}
	Uranium-235	2.2751×10^{-4}
	Oxygen	1.4041×10^{-2}
	Zirconium	3.7115×10^{-3}
Stainless Steel	Chromium	1.7630×10^{-2}
	Iron ¹	6.2361×10^{-2}
	Nickel	7.3296×10^{-3}
Poison	Boron ²	7.7104×10^{-3}
	Carbon ²	2.2598×10^{-3}
	Copper ³	7.9254×10^{-2}
Water Gaps (Water and Stainless Steel)	Chromium	1.4791×10^{-3}
	Iron ¹	5.2321×10^{-3}
	Nickel	6.1495×10^{-4}
	Hydrogen	6.1245×10^{-2}
	Oxygen	3.0622×10^{-2}
Water	Hydrogen	3.343×10^{-2}
	Oxygen	6.686×10^{-2}
Copper	Copper	8.4918×10^{-2}

TABLE 6.3-4 (continued)

<u>Mixture</u>	<u>Nuclide</u>	Atom Density (Atoms/barn - cm)
Lead	Lead	3.2960×10^{-2}
Boro-Silicone and Copper	Oxygen	2.4083×10^{-2}
	Aluminum	5.7378×10^{-3}
	Silicon	5.1264×10^{-3}
	Carbon	7.3741×10^{-3}
	Hydrogen	3.8520×10^{-2}
	Boron	8.0484×10^{-3}
	Copper	1.2118×10^{-2}

- 1) All other elements are added to iron.
- 2) The density of aluminum (2.699 grams/cm³) is used to calculate this atom density.
- 3) The density of copper (8.96 grams/cm³) is used to calculate this atom density.

TABLE 6.3-5
Regional Densities for Heterogeneous
Infinite Array KENO Model

<u>Mixture</u>	<u>Nuclide</u>	<u>Atom Density</u> <u>(Atom/barn - cm)</u>
Fuel Pellet	Oxygen	4.6555×10^{-2}
	Uranium-238	2.2230×10^{-2}
	Uranium-235	1.0475×10^{-3}
Cladding	Zirconium	3.7189×10^{-2}
Water	Hydrogen	6.6736×10^{-2}
	Oxygen	3.3368×10^{-2}
Fuel Channel	Zirconium	4.2910×10^{-2}
304 Stainless Steel	Iron	6.0579×10^{-2}
	Chromium	1.7230×10^{-2}
	Nickel	7.4303×10^{-3}
Copper	Copper	8.4930×10^{-2}
Poison	Iron	5.5644×10^{-2}
	Chromium	1.7076×10^{-2}
	Nickel	1.0747×10^{-2}
	Natural Boron	4.3228×10^{-3}

TABLE 6.3-6
Regional Densities for the Homogeneous
Infinite Array Model

<u>Mixture</u>	<u>Nuclide</u>	<u>Atom Density</u> <u>(Atoms/barn - cm)</u>
Fuel Pellet	Hydrogen	3.7575×10^{-2}
	Oxygen	3.3753×10^{-2}
	Uranium-238	3.3673×10^{-2}
	Uranium-235	7.1461×10^{-3}
	Zirconium	4.2953×10^{-3}
Water	Hydrogen	6.6736×10^{-2}
	Oxygen	3.3368×10^{-2}
Fuel Channel	Zirconium	4.2910×10^{-2}
304 Stainless Steel	Iron	6.0579×10^{-2}
	Chromium	1.7230×10^{-2}
	Nickel	7.4303×10^{-3}
Copper	Copper	8.4930×10^{-2}
Poison	Iron	5.5644×10^{-2}
	Chromium	1.7076×10^{-2}
	Nickel	1.0747×10^{-2}
	Natural Boron	4.3228×10^{-3}

TABLE 6.3-7
Regional Densities for the Homogenous Infinite Array
of NuPac 140-B Casks KENO Model

<u>Mixture</u>	<u>Nuclide</u>	<u>Atom Density (Atoms/barn -cm)</u>
Fuel Pellet	Hydrogen	3.7575×10^{-2}
	Oxygen	3.3753×10^{-2}
	Uranium-238	3.3673×10^{-2}
	Uranium-235	7.1461×10^{-3}
	Zirconium	4.2953×10^{-3}
Water	Hydrogen	6.6736×10^{-2}
	Oxygen	3.3368×10^{-2}
Fuel Channel	Zirconium	4.2910×10^{-2}
304 Stainless Steel	Iron	6.0579×10^{-2}
	Chromium	1.7230×10^{-2}
	Nickel	7.4303×10^{-3}
Copper	Copper	8.4930×10^{-2}
Poison	Iron	5.5644×10^{-2}
	Chromium	1.7076×10^{-2}
	Nickel	1.0747×10^{-2}
	Natural Boron	4.3228×10^{-3}
Lead	Lead	3.2960×10^{-2}
Boro-Silicone	Hydrogen	4.4900×10^{-2}
	Carbon	8.6016×10^{-3}
	Oxygen	2.8092×10^{-2}
	Natural Boron	9.3890×10^{-4}
	Aluminum	6.6929×10^{-3}
	Silicon	5.9798×10^{-3}

6.4 Criticality Calculations

This section presents a description of the criticality calculational methods performed for the NuPac 140-B Cask and Fuel Package. The maximum fuel loading for the PWR and BWR baskets were determined using the computer codes described within Section 6.4.1. Section 6.4.2 discusses the fuel loading optimization and Section 6.4.3 the results of the calculations.

6.4.1 Calculational Method

The effective neutron multiplication factor (k_{eff}) was calculated using the KENO-IV computer code with a mixed box geometry. The NITAWL computer code was used to retrieve cross section data from the AMPX library and perform resonance self-shielding calculations on the isotope U^{238} using the Nordhem Integral Treatment. The DANCOFF correction factor was calculated using the method described by Sauer. To simplify the mixed box model, the fuel assemblies were homogenized using the computer code XSDRNPM. The XSDRN model consisted of a single rod cell surrounded by water. The fuel-cladding gap was neglected and the cladding inner radius and atomic densities were adjusted to compensate. All computer codes are maintained, verified and validated by Power Computing Company, a subsidiary of the Babcock & Wilcox Company of Lynchburg, Virginia. These codes are in compliance with quality assurance requirements of the Appendices of 10 CFR Part 71.

Before the KENO-IV code was executed, a cross section file containing weighted neutron cross sections for the homogenized fuel assemblies, transportation basket and cask materials were generated. This was accomplished by executing the NITAWL (Reference 6.6.1.3) and XSDRNPM (Reference 6.6.1.4) codes in the following sequence. NITAWL was executed to retrieve UO_2 cross section data from the AMPX library and perform resonance self-shielding calculations on the U^{238} . This data was then written to a neutron cross section working file. The NITAWL code was executed a second time to retrieve neutron cross section data from the AMPX library for fuel assembly cladding materials, and to write this data

to the working file. The XSDRNPM code was executed to homogenize the fuel assembly, and to write the weighted neutron cross section file. The NITAWL code was executed a third time to retrieve neutron cross section data for the basket and cask, and to write these values to the working file. The KENO-IV model was executed after generating the neutron cross section file. An example of a typical KENO-IV input deck is shown as Input Deck 5 in Appendix 6.6.2.

Typical input decks of the calculational sequence are given in Appendix 6.4.2. Appendix 6.6.2 contains the NITAWL, XSDRNPM and KENO-IV input decks for the BWR criticality analysis. Input deck 1 contains the NITAWL and XSDRN input files to create BWR homogenized cross-sections. Input deck 2 contains the NITAWL input file to create the heterogenous cross sections. Input deck 3 is the KENO-IV input file for the heterogenous infinite array assemblies. Input deck 4 is the KENO-IV input file for the homogenous infinite array of assemblies. Input deck 5 is the KENO-IV input file for the homogenous infinite array of casks.

All binary and ASCII files generated in this project were written to an eight track magnetic tape which was stored and maintained by Power Computing Company in Dallas, Texas.

6.4.2 Fuel Loading or Other Contents Loading Optimization

The type of PWR and BWR fuel chosen for the NuPac 140-B criticality evaluations was selected as the most reactive fuel assembly considered for transport in the NuPac 140-B Cask.

From the standard PWR fuel assemblies listed in Table 6.2-1, the Westinghouse 15 X 15 fuel assembly was selected as having among the highest infinite neutron multiplication factor, k_{∞} . Studies in the nuclear industry have shown consistently that Westinghouse PWR fuel assemblies yield the highest k_{eff} values when evaluating fuel assemblies for criticality optimization. Most of these studies are documented in company proprietary reports. Two of these reports, References 6.6.1.6 and 6.6.1.7 support the fact that the W15 x 15 PWR fuel assembly is

justifiable for consideration as the optimum fuel assembly when used in criticality safety analysis.

Conference transactions of the American Nuclear Society (ANS) from technical papers presented by members of the Nuclear Criticality Safety Division (NCSD) similarly support the selection of the Westinghouse fuel assemblies for optimum criticality calculations for PWR fuel.

From the standard BWR fuel assemblies listed in Table 6.2-1, the General Electric 8 x 8R fuel assembly was selected as having among the highest infinite neutron multiplication factor based on infinite array calculations performed in Reference 6.6.1.8. The Exxon Nuclear fuels closely resemble the General Electric fuels with the same array size.

6.4.3 Criticality Results

The results of the calculations for the NuPac 140-B Cask and Fuel Package criticality evaluation are presented in two parts. The results of PWR fuel calculations are presented in Section 6.4.3.1 and the BWR fuel calculations in Section 6.4.3.2.

6.4.3.1 Criticality Results for PWR Fuel

KENO-IV was used to determine the reactivity ($k_{eff} < 0.95$) with fresh (unirradiated) PWR fuel, and with 4.5% enriched irradiated PWR fuel. Table 6.4-1 summarizes the key input parameters and the results for the KENO-IV analysis with these conditions.

The NuPac 140-B Cask PWR basket design was determined to be critically safe for both normal and hypothetical accident conditions. The system reactivity was calculated to be 0.928 for 3.2% enriched fresh (unirradiated) PWR fuel including all calculational uncertainties at a 95% confidence level.

The calculated "k" for case 1 in Table 6.4-1 was determined with a total poison panel thickness of 0.125 inches and a water gap of 0.75 inches.

The total flux trap thickness for this analysis was the same as the current PWR basket design, but the poison panel was narrower, and the water gap wider. Tradeoff calculations using this flux trap dimension show that as the poison panel thickness increases and the water channel thickness decreases, k_{eff} also

TABLE 6.4-1
PWR KENO Calculations Summary

KENO Model	Case 1	Case 2	Case 3
Fuel	Homogenous	Homogenous	Homogenate
Geomatic Model	Infinite Array of PWR Assemblies	Infinite Array of PWR Assemblies	Quarter Cask
Enrichment	4.5 ¹	3.2 ²	3.2 ²
Burnup	18,000 MWD/MTU	0	0
Energy Groups	123	123	123
Neutrons/Generation	500	500	500
Particle Histories	51,500	51,500	51,500
Calculate k	0.95226	0.95088	0.91662
Deviation (1 σ)	0.003299	0.003277	0.003624

- 1) A Quarter Cask model will be analyzed for the final design.
- 2) A Quarter Cask model with 3.4% enrichment will be examined for the final design.

decreases. This model was therefore conservative. For the analyses performed using 4.5% enriched fuel, the Quarter Cask model k_{eff} was found to be 0.018 lower than the Single Cell model. The maximum system reactivity for 4.5% enriched fuel with 18,000 MWD/MTU burnup is expected to be 0.944 including the 0.018 correction factor for the Quarter Cask⁴ and all calculational uncertainties at a 95% confidence level.

⁴The PWR Quarter Cask KENO results with a 0.165 inch poison panel and 0.59 inch water gap will be included in the final design.

6.4.3.2 Criticality Results for BWR Fuel

KENO-IV was executed for the three models described in Section 6.3.1.2. Table 6.4-2 summarizes the key input parameters and the results for each of the three KENO-IV models.

TABLE 6.4-2
BWR KENO Calculations Summary

KENO Model	Case 1	Case 2	Case 3
Fuel	Heterogeneous	Homogeneous	Homogeneous
Geometric Model	Infinite Array of BWR Assemblies	Infinite Array of BWR Assemblies	Infinite Array of Casks
Energy Groups	123	123	123
Neutrons/Generation	500	500	500
Particle Histories	27,500	40,000	44,000
Calculated k	1.05696 ^a	1.05192 ^a	0.85231 ^b
Deviation (1 σ)	± 0.00481	± 0.00339	± 0.00319

(a) Represents k_{∞} .

(b) Represents k_{eff} .

The NuPac 140-B Cask BWR basket design was determined to be critically safe for both the normal and hypothetical accident conditions. The system reactivity was calculated to be 0.88706 including all calculational uncertainties at a 95% confidence level.

The cask array KENO-IV computer run was reviewed to determine the stability and statistical behavior of the model. Figure 6.4-1 shows the running average k_{eff} (neglecting the first 3 generations) as a line. Each independent generation k_{eff} is indicated by a dot. As shown in Figure 6.4-1, the average k_{eff} settled out quickly and appears to have stabilized at about 0.85 after 20,000 histories. Execution out to the final number of histories produced only minor fluctuations in the calculated average.

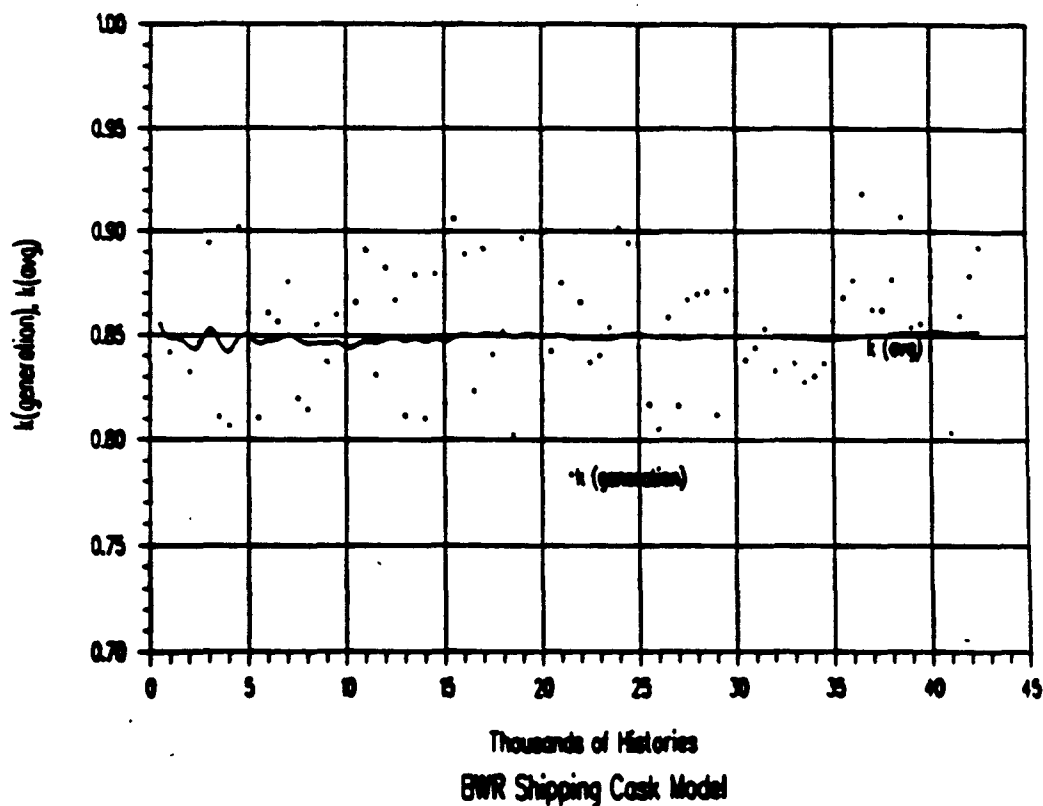


FIGURE 6.4-1
BWR Cask Array KENO-IV Results by Generation

In order to arrive at the reported k_{eff} of 0.88706, two sources of computational bias have been considered. The first is derived from a series of KENO-IV calculations performed against several critical benchmark experiments. The second is a bias which is due to the fuel homogenization technique used in the cask array KENO-IV model.

The KENO-IV bias is calculated to be -0.01423 (non-conservative direction) for this geometry (discussed in Section 6.5.2).

The homogenization bias was determined by running similar models of infinite arrays using heterogeneous and homogeneous fuel assemblies. The results, reported with 1σ uncertainties, are as follows:

$$\begin{aligned} K_{\text{homogeneous}} &= 1.05192 \quad 0.00339 \\ K_{\text{heterogeneous}} &= 1.05696 \quad 0.00481 \end{aligned}$$

The results are statistically equivalent, however a small nonconservative bias in the homogenization technique is indicated and has been accounted for as shown below.

$$\begin{aligned} k_{\text{method}} &= k_{\text{homogeneous}} - k_{\text{heterogeneous}} \\ &= (1.05192 \quad 0.00339) - (1.05696 \quad 0.00481) \\ &= 1.05192 - 1.05696 \\ &\quad (0.00339^2 + 0.00481^2)^{1/2} \\ &= -0.00504 \pm 0.00588 \\ &\quad (\text{non-conservative direction, } 1\sigma) \end{aligned}$$

The following formula was used to determine k_{eff} . Uncertainties were combined statistically and then multiplied by two and added to the average k_{eff} in order to determine the maximum reactivity for the system.

$$k_{eff} = k_{\text{KENO}} - k_{\text{bias}} - k_{\text{method}}$$

$$\begin{aligned} \text{where: } k_{\text{KENO}} &= \text{Calculated reactivity} \\ k_{\text{bias}} &= \text{Bias due to code and cross sections} \\ k_{\text{method}} &= \text{Bias due to homogenization method} \\ k_{\text{KENO}} &= 0.85231 \quad 0.00319 \quad (1\sigma) \end{aligned}$$

$$\begin{aligned}
 k_{\text{bias}} &= -0.01423 \\
 &\quad \text{(non-conservation direction)} \\
 k_{\text{method}} &= -0.00504 \quad 0.00588 \\
 &\quad \text{(non-conservative direction, } 1\sigma)
 \end{aligned}$$

Combining k_{KENO} , k_{bias} , and k_{method} :

$$\begin{aligned}
 k_{\text{eff}} &= k_{\text{KENO}} - k_{\text{bias}} - k_{\text{method}} \\
 &= (0.85231 \quad 0.00319) + 0.01423 \\
 &= + (0.00504 \quad 0.00588) \\
 &= 0.85231 + 0.01423 + 0.000504 \\
 &\quad (0.00504^2 + 0.00588^2)^{1/2} \\
 &= 0.87158 \quad 0.00774 \quad (1\sigma)
 \end{aligned}$$

For a 95% confidence level, the maximum effective reactivity of the loaded cask is therefore:

$$\begin{aligned}
 k_{\text{eff}} &= 0.87158 \quad 2(0.00774) \\
 &= 0.88706
 \end{aligned}$$

6.5 Critical Benchmark Experiments

The methods used to evaluate the criticality safety of the NuPac 140-B Cask and Fuel Package were also used to evaluate a series of low-enriched, uranium-dioxide fuel pins in a water moderated lattice with poisons.

The KENO-IV bias for both PWR and BWR fuels was estimated by comparing the KENO results with those of 21 critical experiments as documented in Reference 6.6.1.9. A brief review of published data available to benchmark the KENO code indicated that the data in Reference 6.6.1.9 is the most applicable.

The KENO code was used to model the critical configurations in Reference 6.6.1.9 with a pin-by-pin representation of the fuel assemblies, and with explicit models of the material interspersed between assemblies. The cross sections for these cases were prepared by NITAWL utilizing the XSDRN cross section set. All pertinent data for each critical configuration are documented in Reference 6.6.1.9 to permit use of these data for validating calculational methods in accordance with ANSI N16.9-1976 (Reference 6.6.1.10). Table 6.5-1 contains the results of both the critical measurements and the KENO calculations.

As reported in Reference 6.6.1.9, twenty-one criticality experiments were performed on nine PWR fuel assemblies. In each experiment, the fuel assemblies were grouped in a 3 x 3 array. The spacing and the material used to separate the fuel assemblies were varied for each experiment. In experiments I through IX, B₂C pins were placed between the fuel assemblies. In experiments X, XI, and XII stainless steel sheets were used, and in experiments XIII through XXI borated aluminum sheets were placed between the fuel assemblies. The average bias from these criticality experiments is 0.004 (0.40%).

TABLE 6.5-1
Measured vs. Calculated Values of k_{eff}

Spacing Between Assys, inches	Core	Material Between Assys	KENO Measured $k_{eff} \pm \sigma$	Measured $k_{eff} \pm \sigma$	KENO Minus Measured $k_{eff} \pm \sigma$
0.000	I	H2O	0.998 $\pm$ 0.006	1.0002 $\pm$ 0.0005	-0.002 $\pm$ 0.006
0.000	II	H2O	1.007 $\pm$ 0.004	1.0001 $\pm$ 0.0005	0.007 $\pm$ 0.004
0.644	III	H2O	0.999 $\pm$ 0.004	1.0000 $\pm$ 0.0006	-0.001 $\pm$ 0.004
0.644	IV	84 B4C Pins	1.004 $\pm$ 0.007	0.9999 $\pm$ 0.0006	0.004 $\pm$ 0.007
0.644	XI	SS Plate	1.015 $\pm$ 0.004	1.0000 $\pm$ 0.0006	0.015 $\pm$ 0.004
0.644	XIII	1.6/Al Plate*	1.008 $\pm$ 0.005	1.0000 $\pm$ 0.0010	0.008 $\pm$ 0.005
0.644	XIV	1.3B/Al Plate	1.003 $\pm$ 0.004	1.0001 $\pm$ 0.0010	0.003 $\pm$ 0.004
0.644	XV	0.41B/Al Plate	0.995 $\pm$ 0.005	0.9998 $\pm$ 0.0016	-0.005 $\pm$ 0.005
0.644	XVII	0.24B/Al Plate	0.993 $\pm$ 0.005	1.0000 $\pm$ 0.0010	-0.007 $\pm$ 0.005
0.644	XIX	0.1B/Al Plate	0.991 $\pm$ 0.004	1.0002 $\pm$ 0.0010	-0.009 $\pm$ 0.004
1.288	V	64 B4C Pins	1.005 $\pm$ 0.005	1.0000 $\pm$ 0.0007	0.005 $\pm$ 0.005
1.288	VI	64 B4C Pins	0.998 $\pm$ 0.004	1.0097 $\pm$ 0.0012	-0.012 $\pm$ 0.004
1.288	XII	SS Plate	0.991 $\pm$ 0.005	1.0000 $\pm$ 0.0007	-0.009 $\pm$ 0.005
1.288	XVI	0.41B/Al Plate	0.990 $\pm$ 0.005	1.0001 $\pm$ 0.0019	-0.010 $\pm$ 0.005
1.288	XVIII	0.24B/Al Plate	1.005 $\pm$ 0.005	1.0002 $\pm$ 0.0011	0.005 $\pm$ 0.005
1.288	XX	0.1B/Al Plate	0.997 $\pm$ 0.005	1.0003 $\pm$ 0.0011	-0.003 $\pm$ 0.005
1.932	VII	34 B4C Pins	0.994 $\pm$ 0.005	0.9998 $\pm$ 0.0009	-0.006 $\pm$ 0.005
1.932	VIII	34 B4C Pins	1.003 $\pm$ 0.005	1.0083 $\pm$ 0.0012	-0.005 $\pm$ 0.005
1.932	X	H2O	0.988 $\pm$ 0.004	1.0001 $\pm$ 0.0009	-0.012 $\pm$ 0.004
1.932	XXI	0.1B/Al Plate	0.981 $\pm$ 0.004	0.9997 $\pm$ 0.0015	-0.019 $\pm$ 0.004
2.576	IX	H2O	0.987 $\pm$ 0.005	1.0030 $\pm$ 0.0009	-0.016 $\pm$ 0.005

*1.6 is the average weight percent of boron in the borated aluminum plate.

Comparing the BWR basket design to the criticals in Table 6.5-1 shows that cores XVI, XVIII, XX, and XXI have similar spacing and materials. The average bias from these criticals is -0.00925 (0.92%). This, in conjunction with the large number of criticals modeled, and the small biases measured (range of 0.001 to 0.019 with most biases being less than 1%), shows that the KENO-IV code and the cross section generating procedure is similar to the BWR basket.

A review of the data indicates no direct correlation between the KENO-to-measured differences and the type of material placed between assemblies. The only obvious correlation is between the inter-assembly spacing. This spacing correlation is illustrated in the upper curve of Figure 6.5-1 in which the average of the differences for a particular assembly spacing are plotted versus assembly spacing. The average values are given in column one of Table 6.5-2. These values represent a best estimate of the bias between the KENO values and the critical experiments. Using this curve and the 1.14 inch assembly spacing in the BWR basket would result in a bias of -0.0025. However, this bias does not give any consideration to the uncertainty in the KENO calculation. An alternate method is to obtain a conservative bias curve that includes all the KENO uncertainties.

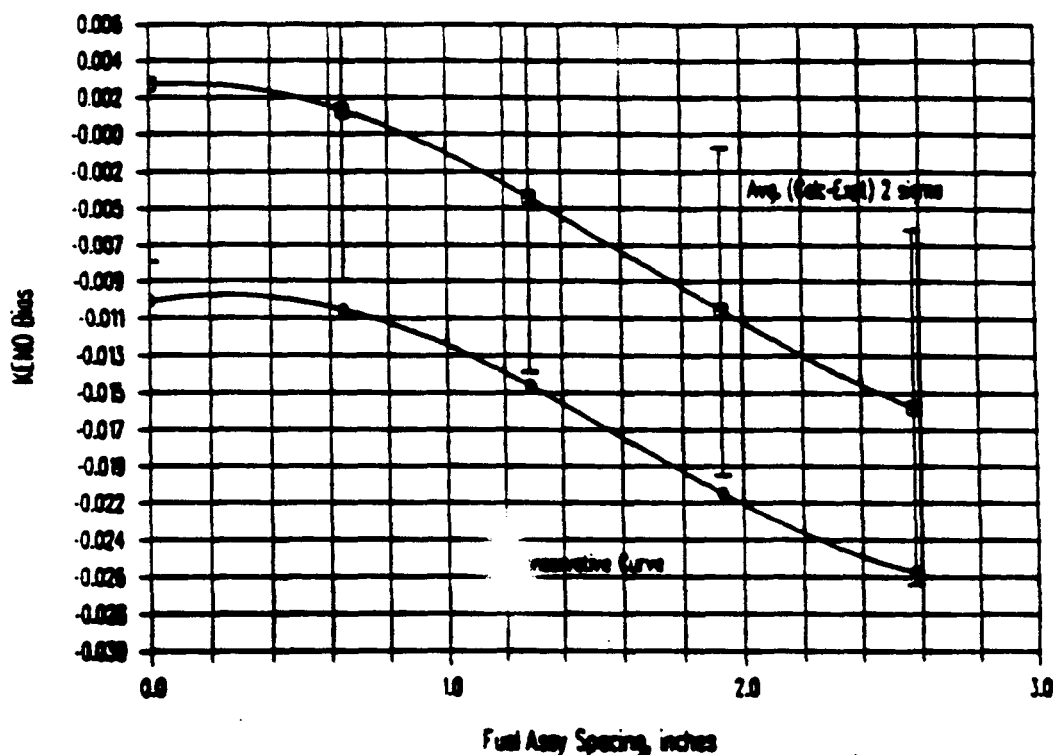


FIGURE 6.5-1
KENO Bias Versus Fuel Assembly Spacing

TABLE 6.5-2
 k_{eff} vs, Assembly Spacing

Spacing Inches	Average $\Delta k_{eff}^{(1)}$ (KENO - Exp)	Average KENO ⁽¹⁾ $k_{eff} + \sigma$	$\Delta k_{eff}^{(2)}$	$\Delta k_{eff} - 2\sigma^{(3)}$
0.000	+0.0025 0.0051	1.0025 0.0051	-----	-0.0100
0.644	+0.0010 0.0049	1.0010 0.0048	-0.0015	-0.0105
1.288	-0.0040 0.0050	0.9977 0.0048	-0.0048	-0.0148
1.932	-0.0105 0.0047	0.9915 0.0045	-0.0110	-0.0210
2.576	-0.0160 0.0051	0.9870 0.0050	-0.0155	-0.0255

Notes: (1) $\sigma = (\Sigma \sigma_i/k)^{1/2}$, where k is the number of calculated KENO values.

(2) Δk_{eff} is the difference between the k_{eff} at the 0.00 inch spacing and at the i^{th} spacing.

(3) σ is the assumed average sigma of the KENO values, i.e., 0.005.

The second column in Table 6.5-2 lists the average KENO calculated k_{eff} at the 0.0 inch spacing and the k_{eff} 's at the other spacings. The conservative bias curve is obtained from this data. To provide a slightly conservative curve, the positive 0.0025 bias of the 0.0 inch spacing will be neglected and the k_{eff} assumed to be 1.0. However, the calculated biases of column three will be retained. Averaging the uncertainties of the k_{eff} 's at the various spacings an approximate value of 0.005 is obtained. To obtain a 95% certainty that the largest bias is obtained, 2σ will be subtracted from the spacing bias of column three.

The fourth column contains conservative 95% uncertainty biases. These values were plotted as the lower curve of Figure 6.5-1 to obtain the recommended bias curve. This curve provides a conservative estimate of the KENO bias for fuel assemblies in a array with spacing between 0.0 and 2.576 inches.

It is noted that this curve bounds all average calculational minus experimental differences including a 1σ uncertainty and all individual data point differences. A simple equation that exactly fits the 0.0, 0.664, and 1.932 values on the curve and provides conservative values for the remaining two points is:

$$\text{Bias} = -0.01 - 0.0015 * (\text{Spacing}^2 / 0.644^2)^{1.8136}$$

For the 1.424 inch spacing of the PWR canister, the KENO bias of 0.01633 is calculated from this equation which is approximately the value read from Figure 6.5-1. Similarly, for the 1.14 inch spacing of the BWR canister, the KENO bias of -0.01423 is obtained. This total conservative bias includes the biases from the code, cross section generation procedure, spacing variation, and code statistics in the benchmark cases.

APPENDIX 6.6

Appendix 6.6.1

References

Appendix 6.6.2

Input Decks for Calculations For the NUPAC 140-B Cask
with BWR Fuel

APPENDIX 6.6.1

6.6.1 References

- 6.6.1.1 Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), "Packaging and Transportation of Radioactive Materials", August 24, 1983.
- 6.6.1.2 KENO-IV User Manual, BWNP-20373-4, April 1987, Babcock and Wilcox Co., Lynchburg, VA.
- 6.6.1.3 NITAWL User's Manual, NPGD-TM-505, Revision 8, December, 1988, Babcock & Wilcox Co., Lynchburg, VA.
- 6.6.1.4 XSDRNPM User's Manual, NPG-TM-13, April, 1985, Babcock & Wilcox Co., Lynchburg, VA.
- 6.6.1.5 M.N. Baldwin, et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, Babcock & Wilcox Company, Lynchburg, Virginia, July, 1979.
- 6.6.1.6 NAC-C-7847, Vol II, Source RTA (1978-79)
- 6.6.1.7 DOE/ET/47912-2; NAC-8134 (September 1981)
- 6.6.1.8 General Electric IF-300 Shipping Cask CSAR, NEDO - 10084-3, February, 1985.
- 6.6.1.9 "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel" Summary report by M.N. Baldwin, et al., BAW-1484-7. DOE contract EY-77-C-09-1001.
- 6.6.1.10 "Validation of Calculational Methods for Nuclear Criticality Safety," ANSI N16.9-1976.
- 6.6.1.11 NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" July 1980.

APPENDIX 6.6.2
Input Decks for Calculations
For the NUPAC 140-B Cask
with BWR Fuel

Input Deck 1

NITAWL & XSDRN Homogenous Cross-sections

Creates the homogenous cross-section working library
)

1\$\$ 0 5 0 0 0
3 4 1 126 0
0 0 T

2\$\$ 10001 80004 400000 922351 922381
3**

922381 293.15 2 0.52832 0.19294
61.97 2.2230E-2 1 235.0 4.8332
1 16.0 6.7382 1 1.0

T
/EOR

XSDRNPM - 8X8R FUEL ROD CELL XSEC WEIGHTING

1\$\$ 2 3 12 0 3 3 6 8 3 1
10 25 0 0 0

2\$\$ -2 0 1 E
3\$\$ 1 0 0 0 5.0 0 0 0 0 0

4\$\$ -1 123 0 -2 3 4 126 -1 0
5** 1.0E-04 2.0E-04 1.0 0.0 0.0 1.420892 E

T
13\$\$

1 1 1

2
3

14\$\$ 922351 922381 80004
400000

80004 10001
15** 1.0475-3 2.2230-2 4.6555-2
3.7189-2

3.3368-2 6.6736-2 T
33** 5R1.0 F0.0 T

35** 0.0 0.15 0.30 0.45 0.50 0.52070
0.567055 0.61341

0.65 0.75 0.85 0.90 0.92353

36\$\$ 5R1 2.2 5R3

39\$\$ 1 2 3

40\$\$ F3

51\$\$ 12111.0 123.0 T

/EOR

1\$\$ 0 0 0 10 0 3 4 0 126 0 0 0 T

2\$\$ 922351 -6922351 922381 -6922381

80004 -680004 400000 -6400000

10001 -610001 T

/EOR

1\$\$ 0 11 0 5 0 3 4 0 126 E T

2\$\$

10001 50000 60002 80004 130000

140005 240003 260001 280003 290001

400000 610001 680004 6400000 6922351 6922381

T

/EOR

(Input deck to add lead to the cross-section tape)

1\$\$ 0 1 0 16 0 3 4 0 126 E T

2\$\$

820000

10001 50000 60002 80004 130000

140005 240003 260001 280003 290001

400000 610001 680004 6400000 6922351

6922381

T

/EOR

Input Deck 2
NITAWL Heterogenous

NITAWL -Create heterogenous cross-sections for the
infinite array of assemblies.)

1\$\$

0	11	0	0	0
3	4	2	126	0
0	0			

T

2\$\$

10001	50000	80004	240003	260001
280003	290001	400000	922351	922381
-1922381				

3**

922381	293.15	2	0.52832	0.19294
61.79	2.2230E-2	1	235.0	4.8332
1	16.0	6.7382	1	1.0
1922381	293.15	2	0.52832	0.19294
61.79	2.2230E-2	1	235.0	4.8332
1	16.0	6.7382	1	1.0

T

Input Deck 3
KENO-IV Heterogenous Infinite Array

```

( INPUT DECK 3
KENOIV -heterogenous infinite array of assemblies. )

HET1PO.KI - INF ARRAY, 1.0% N'SORB 08/04 11:51
10.0 55 500 3 123 75 10 6 14 13
2 8 8 1 -10 1 0 2000 00 1
1 1 0 0 0 00 0 0
-1.00 -1.00 -1.00 -1.00 -1.00 -1.00
1 922381 2.22300E-02
1 -922351 1.04750E-03
1 80004 4.65550E-02
2 260001 6.05790E-02
2 240003 1.72300E-02
2 280003 7.43030E-03
3 290001 8.49300E-02
4 260001 5.56448E-02
4 240003 1.70760E-02
4 280003 1.07473E-02
4 50000 4.32292E-03
5 400000 3.71890E-02
6 80004 3.33680E-02
6 10001 6.67360E-02

BOX TYPE 1
CYLINDER 1 0.528320 30.480000 0.000000 1232
CYLINDER 5 0.613410 30.480000 0.000000 1232
CUBOID 6 0.812800 -0.812800 0.812800 -0.812800
30.480000 0.000000 1232

BOX TYPE 2
CYLINDER 6 0.674370 30.480000 0.000000 1232
CYLINDER 5 0.750570 30.480000 0.000000 1232
CUBOID 6 0.812800 -0.812800 0.812800 -0.812800
30.480000 0.000000 1232

CORE BDY 0 14.452600 1.447800 14.452600 1.447800
30.480000 0.000000 1232

CUBOID 6 14.653260 1.247140 14.653260 1.247140
30.480000 0.000000 1232

CUBOID 5 14.907260 0.993140 14.907260 0.993140
30.480000 0.000000 1232

CUBOID 6 15.481300 0.419100 15.481300 0.419100
30.480000 0.000000 1232

CUBOID 2 15.544800 0.355600 15.544800 0.355600
30.480000 0.000000 1232

CUBOID 3 15.621000 0.279400 15.621000 0.279400
30.480000 0.000000 1232

CUBOID 4 15.900400 0.000000 15.900400 0.000000
30.480000 0.000000 1232

1 1 8 1 1 8 1 1 1 1 0
2 4 4 1 4 4 1 1 1 1 0
2 5 5 1 5 5 1 1 1 1 1

END KENO

```


Input Deck 4
KENO-IV Homogenous Infinite Array

```

(          INPUT DECK 4
KENOIV- Homogenous infinite array of fuel assemblies      )

INFHOM.KI - INF ARRAY, 1.0% N'SORB, HOMOGENIZED FA
  9.0  103  500   3  123   75  13   6  16   7
    1   1   1   1  -13   1   0  2000  00   1
    1   1   0   0   0   00   0   0
-1.00 -1.00 -1.00 -1.00 -1.00 -1.00
  1  6922381  7.1461E-03
  1 -6922351  3.3673E-04
  1  6400000  4.2953E-03
  1  680004  3.3753E-02
  1  610001  3.7575E-02
  2  260001  6.05790E-02
  2  240003  1.72300E-02
  2  280003  7.43030E-03
  3  290001  8.49300E-02
  4  260001  5.56448E-02
  4  240003  1.70760E-02
  4  280003  1.07473E-02
  4  50000  4.32292E-03
  5  400000  4.29100E-02
  6  80004  3.33680E-02
  6  10001  6.67360E-02
BOX TYPE  1
CUBOID   1  14.452600  1.447800  14.452600  1.447800
           30.480000  0.000000  1232
CUBOID   6  14.653260  1.247140  14.653260  1.247140
           30.480000  0.000000  1232
CUBOID   5  14.907260  0.993140  14.907260  0.993140
           30.480000  0.000000  1232
CUBOID   6  15.481300  0.419100  15.481300  0.419100
           30.480000  0.000000  1232
CUBOID   2  15.544800  0.355600  15.544800  0.355600
           30.480000  0.000000  1232
CUBOID   3  15.621000  0.279400  15.621000  0.279400
           30.480000  0.000000  1232
CUBOID   4  15.900400  0.000000  15.900400  0.000000
           30.480000  0.000000  1232
END KENO

```

Input Deck 5

KENO-IV Homogenous Cask Model

(INPUT DECK 5

KENOIV- Homogenous cask model, infinite array of casks
with reflected boundaries)

CASK1P0.KI - NOMINAL CASK, 1.0% NISORB

07/31 17:01

10.0	103	500	3	123	75	17	8	23	23
10	13	13	1	-17	1	0	2000	00	0
1	1	0	0	0	00	0	0	0	0

-1.00 -1.00 -1.00 -1.00 -1.00 -1.00

1	610001	3.75750E-02
1	680004	3.37530E-02
1	6400000	4.29530E-03
1	6922351	3.36730E-04
1	6922381	7.14610E-03
2	240003	1.72300E-02
2	260001	6.05790E-02
2	280003	7.43030E-03
3	290001	8.49300E-02
4	50000	5.56448E-02
4	240003	1.70760E-02
4	260001	1.07473E-02
4	280003	4.32292E-03
5	400000	4.29100E-02
6	80004	3.33680E-02
6	10001	6.67360E-02
7	10001	4.49000E-02
7	60002	8.60160E-03
7	80004	2.80920E-02
7	50000	9.38900E-04
7	130000	6.69290E-03
7	140005	5.97980E-03
8	820000	3.29600E-02

BOX TYPE 1

CUBOID 1	6.502400	-6.502400	6.502400	-6.502400
	30.480000	0.000000	1232	
CUBOID 6	6.703060	-6.703060	6.703060	-6.703060
	30.480000	0.000000	1232	
CUBOID 5	6.957060	-6.957060	6.957060	-6.957060
	30.480000	0.000000	1232	
CUBOID 6	7.531100	-7.531100	7.531100	-7.531100
	30.480000	0.000000	1232	

BOX TYPE 2

CUBOID 2	0.063500	0.000000	15.062200	0.000000
	30.480000	0.000000	1232	
CUBOID 3	0.139700	0.000000	15.062200	0.000000
	30.480000	0.000000	1232	
CUBOID 4	0.419100	0.000000	15.062200	0.000000
	30.480000	0.000000	1232	

BOX TYPE 3

CUBOID 4	0.279400	0.000000	15.062200	0.000000
	30.480000	0.000000	1232	
CUBOID 3	0.355600	0.000000	15.062200	0.000000
	30.480000	0.000000	1232	
CUBOID 2	0.419100	0.000000	15.062200	0.000000
	30.480000	0.000000	1232	

BOX TYPE 4

CUBOID 4	15.062200	0.000000	0.279400	0.000000
	30.480000	0.000000	1232	
CUBOID 3	15.062200	0.000000	0.355600	0.000000
	30.480000	0.000000	1232	
CUBOID 2	15.062200	0.000000	0.419100	0.000000
	30.480000	0.000000	1232	

BOX TYPE 5

CUBOID 2	15.062200	0.000000	0.063500	0.000000
	30.480000	0.000000	1232	
CUBOID 3	15.062200	0.000000	0.139700	0.000000
	30.480000	0.000000	1232	
CUBOID 4	15.062200	0.000000	0.419100	0.000000
	30.480000	0.000000	1232	

BOX TYPE 6

CUBOID 2	0.419100	0.000000	0.419100	0.000000
	30.480000	0.000000	1232	

BOX TYPE 7

CUBOID 6	7.531100	-7.531100	7.531100	-7.531100
	30.480000	0.000000	1232	

BOX TYPE 8

CUBOID 6	0.419100	0.000000	15.062200	0.000000
	30.480000	0.000000	1232	
BOX TYPE 9				
CUBOID 6	15.062200	0.000000	0.419100	0.000000
	30.480000	0.000000	1232	
BOX TYPE 10				
CUBOID 6	0.419100	0.000000	0.419100	0.000000
	30.480000	0.000000	1232	
CORE BOX 0	64.020700	0.000000	64.020700	0.000000
	30.480000	0.000000	1232	
CUBOID 2	75.247500	0.000000	75.247500	0.000000
	30.480000	0.000000	1232	
CUBOID 8	84.137500	0.000000	84.137500	0.000000
	30.480000	0.000000	1232	
CUBOID 2	87.947500	0.000000	87.947500	0.000000
	30.480000	0.000000	1232	
CUBOID 7	102.882700	0.000000	102.882700	0.000000
	30.480000	0.000000	1232	
CUBOID 2	104.787700	0.000000	104.787700	0.000000
	30.480000	0.000000	1232	
CUBOID 6	114.947700	0.000000	114.947700	0.000000
	30.480000	0.000000	1232	
6	1 13 1	1 13 1	1 1 1	0
1	2 11 3	2 11 3	1 1 1	0
7	11 11 1	8 8 1	1 1 1	0
7	8 11 3	11 11 1	1 1 1	0
4	2 11 3	1 13 3	1 1 1	0
5	2 11 3	3 12 3	1 1 1	0
3	1 13 3	2 11 3	1 1 1	0
2	3 12 3	2 11 3	1 1 1	0
10	12 13 1	9 10 1	1 1 1	0
10	12 13 1	12 13 1	1 1 1	0
10	9 10 1	12 13 1	1 1 1	0
8	12 13 1	11 11 1	1 1 1	0
9	11 11 1	9 10 1	1 1 1	0
9	11 11 1	12 13 1	1 1 1	1

END KENO

7.0 OPERATING PROCEDURES

7.1 Introduction

- 7.1.1 This section contains a summary of cask operating procedures for loading the NuPac 140-B Cask at a reactor and for unloading the NuPac 140-B Cask at the MRS/Repository using a Hot Cell.
- 7.1.2 The procedure contains only the steps which must be performed in the receipt, cask handling, loading, testing, drying, inerting, and shipping of the NuPac 140-B Cask.
- 7.1.3 The procedures do not include the "how to do" or specific tooling requirements. This information will be included in the Cask Operating Manual.
- 7.1.4 All procedures start with receipt of the cask and end with cask ready for shipment. They do not included steps for more than one cask.
- 7.1.5 The procedure for the reactor site is a generic procedure and such things as prerequisites, will vary for each reactor.
- 7.1.6 The procedure for the Hot Cell unloading is based on the following:
 - 7.1.6.1 Use of remote-automated methods
 - 7.1.6.2 Use of a facility supplied yoke (non-redundant)
 - 7.1.6.3 A separate cask work station
 - 7.1.6.4 Use of a transfer dolly to move cask into the Hot Cell and to raise cask to engage Hot Cell
 - 7.1.6.5 No need for major decontamination

- 7.1.7 These procedures summations do not contain "warning or cautions" but these will be included in the "Cask Operation Procedures Manual" and will be similar to the following:

WARNING

A WARNING is provided where a potentially dangerous condition to personnel could exist.

WARNING RADIATION HAZARD

A WARNING - RADIATION HAZARD is provided where a potential radiation hazard could exist.

CAUTION

A CAUTION is provided where a condition or situation exists with potential for damage to equipment.

7.2 Cask Handling Procedure At A Reactor

NOTE: This procedure is based on a BWR Reactor with a Redundant Lifting System. A procedure for a PWR Reactor and Non-Redundant Crane would be less complex, but similar to the following.

7.2.1 Prerequisites

NOTE: All site specific prerequisites must be completed prior to cask handling and prior to shipment.

- 7.2.1.1 Verify that the Utility is a registered licensee - user (see 10 CFR 71.12 for requirements) to the cask furnished.
- 7.2.1.2 Verify that the cask owner has supplied a current certification stating the cask is in full compliance to all current C of C, SRAP and regulation requirements and with the date the certification expires (see NRC 1E Information Notice No. 83-10 for more details).
- 7.2.1.3 Verify that all fuel assemblies to be shipped meet the contents requirements of the Cask C of C. These requirements will include such things as pin array, maximum active length, burn up, cooling time, and initial enrichment.
- 7.2.1.4 Verify that all fuel assemblies have been inspected for fuel cladding integrity (no cladding defects greater than pin holes or hairline cracks).
- 7.2.1.5 Verify that all cranes used for this procedure have been tested, inspected and are dated for use.

- 7.2.1.6 Verify that all procedures for handling and loading this cask are approved and in place. This includes such procedures as gage calibration, torque wrench calibration and cask handling.
- 7.2.1.7 Verify that all personnel who will be performing cask handling are trained and that training is up to date.
- 7.2.1.8 These prerequisites will vary for each reactor due to differences in Technical Specifications and operating philosophy.

7.2.2 Cask And Rail Car Receipt

- 7.2.2.1 Perform security search of the rail car & cask.
- 7.2.2.2 Perform external radiation survey and radiological smear survey of the rail car, sunshield, and impact limiters.
- 7.2.2.3 Open the sunshield/personnel barrier.
- 7.2.2.4 Complete incoming radiological survey of the shipping cask.
- 7.2.2.5 Inspect the transport vehicle and cask for shipping damage.

7.2.3 Prepare The Cask For Removal From The Rail Car

- 7.2.3.1 Place Cask and Transporter in airlock (or cask unloading area).
- 7.2.3.2 Set up a Radiation work area (if required).
- 7.2.3.3 Remove the impact limiters from the cask body and store on rail car.

- 7.2.3.4 Remove cask hold down device.
- 7.2.3.5 Install cask lifting device to facility crane.
- 7.2.3.6 Install cask turning fixture.
- 7.2.3.7 Engage a device to respot the transporter doing cask rotation if required (required at most BWR sites).
- 7.2.3.8 Spot the transporter so cask lifting yoke can be engaged to cask.

NOTE: Some of the above tasks may require more than one location at any given Reactor Site.

7.2.4 Placing the Cask on the Decon Pad

- 7.2.4.1 Remove any hatchway cover to facilitate cask transfer.
- 7.2.4.2 Engage yoke to cask.
- 7.2.4.3 Raise the cask to the vertical position.
- 7.2.4.4 Install the redundant lifting system.
- 7.2.4.5 Move the cask to the cask work area.
- 7.2.4.6 Remove the lifting system from the cask.
- 7.2.4.7 Install the working scaffolding.
- 7.2.4.8 Store the lifting system.
- 7.2.4.9 Replace any hatchway covers removed to facilitate cask transfer.

7.2.5 Preparing The Cask For Loading On Decon Pad

- 7.2.5.1 Wash off road dirt.
- 7.2.5.2 Remove valve box covers as appropriate.
- 7.2.5.3 Install vent and fill hoses.
- 7.2.5.4 Vent the cask and fill with water.
- 7.2.5.5 Unbolt cask lid and install guide pins.
- 7.2.5.6 Engage the lifting system to the cask.
- 7.2.5.7 Install head removal bolts and engage head restraint system.
- 7.2.5.8 Close valves and remove vent and fill hoses.
- 7.2.5.9 Install a venting device to vent port in head (so cask water pressure will equalize while placing or removing cask from pool).

7.2.6 Loading The Cask

- 7.2.6.1 Disassemble scaffolding as required to move cask.
- 7.2.6.2 Place cask in pool (wet with water as it is being lowered).
- 7.2.6.3 Disengage the cask lifting system and head restraint system from the cask body.
- 7.2.6.4 Remove the cask lid from pool.
- 7.2.6.5 Inspect head seal area and O-rings.

- 7.2.6.6 Install cask seal area protector device.
- 7.2.6.7 Load the Cask with spent fuel.
- 7.2.6.8 Verify the fuel assembly serial numbers are correct for this shipment.
- 7.2.6.9 Inspect the fuel assembly loading to be sure fuel assemblies are completely seated in their cells.
- 7.2.6.10 Perform an underwater (view box or TV) visual inspection for debris on top of fuel or on seal protector device.
- 7.2.6.11 Remove any debris from the cask found in step 7.2.6.10.
- 7.2.6.12 Remove the seal surface protector device.
- 7.2.6.13 Replace cask lid on cask.
- 7.2.6.14 Reengage the cask lifting system and cask restraint system.
- 7.2.6.15 Remove the cask from the pool (wash off the cask as it is being removed).
- 7.2.6.16 Place cask on decon pad.
- 7.2.6.17 Remove cask lifting system and head lifting bolts from cask.

7.2.7 Preparing The Cask For Shipment On The Decon Pad

- 7.2.7.1 Reinstall the scaffolding.
- 7.2.7.2 Disconnect venting device from the cask port.

7.2.7.3 Reattach vent hose and vent the cask.

NOTE: Due to heat load in cask the vent hose may contain water.

7.2.7.4 Decon Cask.

Note: This is a good ALARA step because the water adds shielding and reduces radiation exposure.

7.2.7.5 Remove guide pins.

7.2.7.6 Torque the cask lid to 1,350 ft lbs + 125 ft lbs and safety wire bolts.

NOTE: Use torque sequence stamped on the closure head.

7.2.7.7 Connect a drain hose with method to detect when flow changes from water to gas to the lower fill/drain valve.

7.2.7.8 Disconnect the vent hose and connect a hose to a Helium supply with a pressure regulator set at 25 psig.

7.2.7.9 Open the drain and vent valves.

7.2.7.10 Pressurize the cask with Helium until the drain line indicates gas flowing. Shut off the Helium supply.

7.2.7.11 Shut the drain valve when gas pressure stops flowing or 0 psig gas pressure is in the cask and disconnect hose.

7.2.7.12 Verify that Helium gas exhausts from drain line. This verification is to prove less than 1 ft³ of water remains in the cask and the fuel shipment is blanketed with inert gas.

7.2.7.13 There is no need to vacuum dry the cask with a requirement of less than 1 ft³ of water, however, if vacuum drying is used perform the following steps:

- 7.2.7.13.1 Replace the helium used in step 7.2.7.10 with air.
- 7.2.7.13.2 Disconnect the air hose from the vent valve.
- 7.2.7.13.3 Install a vacuum pump to the vent port.
- NOTE:** Using the vent port reduces the amount of radioactive crud removed from the cask while pulling the vacuum on the cask.
- 7.2.7.13.4 Draw a vacuum on the cask to the pre-established limits.
- 7.2.7.13.5 Shut vent valve and remove vacuum pump.
- 7.2.7.13.6 Connect Helium source to vent port.
- 7.2.7.13.7 Open vent valve and back fill cask with Helium to 0 psig pressure.
- 7.2.7.13.8 Shut vent valve and disconnect Helium source.

7.2.7.14 Assembly Verification Leak Test.

NOTE: The actual acceptance test leak rate will depend on NRC approval of the C of C. An acceptance leak rate of 1×10^{-3} standard cubic centimeter per second is being used for this procedure. This leak test will meet the requirements of ANSI N14.5-1987.

- 7.2.7.14.1 Install the gas pressure rise leak detection equipment to the seal test port.
- 7.2.7.14.2 Evacuate through the seal test port until the vacuum is sufficient to operate the gas pressure rise leak detection equipment per the manufacturer's recommendations.

- 7.2.7.14.3 Perform leak tests and accept leaks equal to or smaller than 1×10^{-3} standard cubic centimeters per second.
- 7.2.7.14.4 Disconnect the leak detection equipment and close the test port.
- 7.2.7.14.5 Install the gas pressure rise leak detection equipment to the vent port.
- 7.2.7.14.6 Evacuate through the vent port until the vacuum is sufficient to operate the gas pressure rise leak detection equipment per the manufacturer's recommendations.
- 7.2.7.14.7 Perform leak tests and accept leaks equal to or smaller than 1×10^{-3} standard cubic centimeters per second.
- 7.2.7.14.8 Upon completion of the vent port plug seal integrity leak test, remove the gas pressure rise leak detection equipment from the vent port and replace the vent port cover (13 - 16 ft-lbs torque).
- 7.2.7.14.9 Install the gas pressure rise leak detection equipment to the drain port.
- 7.2.7.14.10 Evacuate through the drain port until the vacuum is sufficient to operate the gas pressure rise leak detection equipment per the manufacturer's recommendations.
- 7.2.7.14.11 Perform leak tests and accept leaks equal to or smaller than 1×10^{-3} standard cubic centimeters per second.

7.2.7.14.12 Upon completion of the drain port plug seal integrity leak test, remove the gas pressure rise leak detection equipment from the drain port and replace the drain port cover (13 - 16 ft-lbs torque).

7.2.7.14.13 Visually inspect the rupture disk for damage.

Note: If leak testing of the rupture disk installation is required, the test will be done during this step.

7.2.7.14.14 Safety wire shut all valves.

7.2.7.14.15 Engage cask yoke.

7.2.7.14.16 Raise cask and decon underside of cask.

7.2.7.14.17 Remove scaffolding.

7.2.8 Place The Cask On The Transportation Vehicle

7.2.8.1 Conduct a radiological survey of the transportation vehicle and decon as required.

7.2.8.2 Open hatchways if required.

7.2.8.3 Lower the cask to the transport vehicle.

7.2.8.4 Remove the redundant lifting system.

7.2.8.5 Lower the cask to the horizontal position.

7.2.8.6 Store the yoke.

7.2.8.7 Close the hatchway if required.

7.2.9 Preparing The Cask For Shipment On The Rail Car

- 7.2.9.1 Remove the cask turning fixture.
- 7.2.9.2 Resmear the cask and spot decon.
- 7.2.9.3 Install cask hold down devices.
- 7.2.9.4 Install cask impact limiters.
- 7.2.9.5 Remove the device for respotting the car during cask rotation.
- 7.2.9.6 Move the rail car from the airlock.

7.2.10 Shipping The Cask

- 7.2.10.1 Conduct final Radiological Survey.
- 7.2.10.2 Prepare Shipping Papers.
- 7.2.10.3 Install labels, placards, etc. on casks and rail car.
- 7.2.10.4 Close the sunshield/personnel barrier.
- 7.2.10.5 Apply seals as required.
- 7.2.10.6 Inspect the rail car.
- 7.2.10.7 Move the cask off-site.

7.3 Procedure For Unloading At The Repository/MRS Using A Hot Cell

- NOTE:
1. This procedure covers only the major steps affecting the cask. The repository/MRS will use a combination of contact, remote, and remote-automated handling methods. A detail procedure can not be written at this time.
 2. "Miscellaneous" are minor steps such as removal of valve covers, wetting down the cask while placing in pool, or removing and installing of safety wire.
 3. This procedure is based on use of a cask work station where lid unbolting, lid bolting, venting, spot deconning cask leak testing, etc. are performed.
 4. It is also based on use of a transfer dolly for moving the cask to and from the Hot Cell Opening.

7.3.1 Cask Receiving

- 7.3.1.1 Perform incoming inspection.
- 7.3.1.2 Open sunshield/personnel barrier.
- 7.3.1.3 Perform incoming radiation survey.
- 7.3.1.4 Miscellaneous

7.3.2 Cask Preparation on Rail Car

- 7.3.2.1 Place rail car under crane.
- 7.3.2.2 Remove impact limiters.
- 7.3.2.3 Install rotating device.

7.3.2.4 Rotate cask to vertical.

7.3.2.5 Move cask to cask work station.

7.3.2.6 Miscellaneous

7.3.3 Cask Preparation at Work Station

7.3.3.1 Take cask gas samples.

7.3.3.2 If cask cool down is required, install this system and cool the cask to facility limits.

7.3.3.3 Vent cask.

7.3.3.4 Unbolt cask lid.

7.3.3.5 Install cask lid lifting fixture.

7.3.3.6 Place cask on transfer dolly.

7.3.4 Cask Unloading In Hot Cell

7.3.4.1 Place cask under Hot Cell Opening.

7.3.4.2 Engage cask to Hot Cell and seal.

7.3.4.3 Engage crane hook to lid lifting fixture.

7.3.4.4 Remove cask lid.

7.3.4.5 Verify fuel assemblies serial numbers agree with shipping papers.

7.3.4.6 Unload fuel assemblies.

7.3.4.7 Use remote viewing device and inspect O-ring.

7.3.4.8 Install cask and lid disengage crane hook.

7.3.4.9 Disengage cask from Hot Cell.

7.3.4.10 Use dolly and move cask back to cask Work Station.

7.3.4.11 Miscellaneous

7.3.5 Cask Preparation At Work Station

7.3.5.1 Torque cask lid bolts & safety wire.

7.3.5.2 Decon cask if required (Note: casks do weep in transit).

7.3.5.3 Preform assembly verification leak test if residue in cask is greater then Type A quantity (for large cask this may be true after 5 or 6 loads).

7.3.5.4 Miscellaneous

7.3.6 Load Casks on Rail Car

7.3.6.1 Engage yoke to cask.

7.3.6.2 Move cask to rail car.

7.3.6.3 Rotate cask to horizontal.

7.3.6.4 Disengage yoke.

7.3.6.5 Miscellaneous

7.3.7 Preparation Of Cask For Shipment

7.3.7.1 Remove cask turning fixture.

7.3.7.2 Install impact limiters.

7.3.7.3 Move cask to outside (away from any background radiation).

7.3.7.4 Take final survey and label.

7.3.7.5 Install sunshield.

7.3.7.6 Make out shipping papers (Note: Placards are not normally required for empty cask).

7.3.7.7 Miscellaneous

7.4 REFERENCES

- 7.4.1 Title 10 Code of Federal Regulations Part 71 (10 CFR 71)
Packaging and Transportation of Radioactive Material
- 7.4.2 NRC I E Information Notice No. 83-10 Clarification of Several
Aspects Relating to use of NRC Certified Transport Packages,
March 11, 1983
- 7.4.3 ANSI N14.5-1987, American National Standard for Radioactive
Materials Leakage Tests on Packages for Shipment

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

This chapter describes the acceptance tests and the maintenance program which shall be used on the NuPac 140-B Cask in compliance with Subpart G of 10 CFR 71 (Reference 8.3.1.1)

8.1 Acceptance Tests

This section discusses the tests to be performed prior to first use of the NuPac 140-B Cask.

8.1.1 Visual Inspection

All NuPac 140-B Cask materials of construction and welds shall be examined in accordance with the specifications delineated on the Packaging General Arrangement Drawing found in Section 1.4.

8.1.2 Structural and Pressure Tests

8.1.2.1 Lifting Device Load Testing

The maximum total working load of each set of NuPac 140-B Cask lifting trunnions is the loaded weight of the cask with water in the cavity but without impact limiters, or 190,108 pounds, or 95,054 pounds per lifting trunnion. Each set of cask lifting trunnions shall be load tested to 150% of their maximum total working load, or 285,162 pounds. Load testing of the cask lifting trunnions shall be performed after installation of the lead and final welding of the cask body welds.

Per the Packaging General Arrangement Drawing in Section 1.4, accessible base material and welds directly related to the load testing of the lifting trunnions shall be visually inspected for plastic deformation or cracking, and liquid penetrant inspected per Section V, Article 6, and Section III, Division 1, Subsection NB, Article NB-5000 of Reference 8.3.1.2, as called for in Reference 8.3.1.3. Indications of cracking or distortion shall be recorded on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition prior to repair and final acceptance in accordance with the requirements of Section 9.2.15.

8.1.2.2 Containment Vessel Pressure Testing

The cask containment vessel shall be pressure tested to 150% of the design pressure per the requirements of 10 CFR 71.85(b), Reference 8.3.1.1, to verify structural integrity. The maximum design pressure of the cask containment vessel is 400 psig. Thus, the containment vessel shall be pressure tested to 600 psig.

Accessible base material and welds directly related to the pressure testing of the containment vessel shall be visually inspected for plastic deformation or cracking, and liquid penetrant inspected per Section V, Article 6, and Section III, Division 1, Subsection NB, Article NB-5000 of Reference 8.3.1.2, as called for in Reference 8.3.1.3. Indications of cracking or distortion shall be recorded on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition prior to repair and final acceptance in accordance with the requirements of Section 9.2.15.

8.1.3 Leakage Tests

Fabrication Verification Leak Testing shall conform to the requirements of Section 6.3, Reference 8.3.1.4.

The Fabrication Verification Leak Test shall be performed after the structural and/or pressure test(s) delineated in Section 8.1.2, above, to verify package integration and performance to design criteria. For testing, the containment vessel shall be thoroughly cleaned and leak tested prior to installation of a basket assembly.

Fabrication Verification Leak Tests shall be performed on the cask containment vessel. Five (5) separate tests comprise the series. Each test shall meet the acceptance criteria delineated in Section 8.1.3.6, below.

8.1.3.1 Testing the Cask Containment Vessel Structure Integrity

8.1.3.1.1 The Fabrication Verification Leak Test of the cask containment vessel structure shall be performed per Section A3.10.2, Helium Mass Spectrometer Envelope, Pressurized Envelope, Reference 8.3.1.4.

8.1.3.1.2 Remove the vent port cover and open the plug to allow free passage of gas through the vent port.

8.1.3.1.3 Remove the cask lid. Install a large sealed plug into the cask cavity to reduce the cavity void volume.

8.1.3.1.4 Install the cask lid with both O-ring seals installed in the lid. Assembly is as shown on the Packaging General Arrangement Drawings in Section 1.4.

8.1.3.1.5 Install the helium mass spectrometer leak detector to the vent port.

8.1.3.1.6 Evacuate the cask cavity through the vent port until the vacuum is sufficient to operate the helium mass spectrometer leak detector per the manufacturer's recommendations.

8.1.3.1.7 Provide a helium atmosphere about the exterior of the cask.

8.1.3.1.8 Perform the leak test to the requirements of Section 8.1.3.6. If after repeated attempts, the cask containment vessel structure fails to pass the leak test, isolate the leak path and, prior to repairing the leak path and repeating the cask containment vessel structure leak test, record on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition in accordance with the requirements of Section 9.2.15.

8.1.3.2 Testing the Body O-ring Seal Integrity

8.1.3.2.1 The Fabrication Verification Leak Test of the body O-ring seal shall be performed per Section A3.10.1, Helium Mass Spectrometer Envelope, Evacuated Envelope, Reference 8.3.1.4.

8.1.3.2.2 The cask shall be assembled with both main O-ring seals installed in the cask lid. Assembly is as shown on the Packaging General Arrangement Drawing in Section 1.4.

8.1.3.2.3 Remove the vent port cover and open the plug to allow free passage of gas through the vent port.

8.1.3.2.4 Connect a vacuum pump to the vent port.

8.1.3.2.5 Evacuate the cask cavity through the vent port to less than 1 psia.

8.1.3.2.6 Install the helium mass spectrometer leak detector to the seal test port.

8.1.3.2.7 Evacuate the seal test port until the vacuum is sufficient to operate the helium mass spectrometer leak detector per the manufacturer's recommendations.

8.1.3.2.8 Provide a helium atmosphere within the interior of the cask by backfilling with helium gas to one atmosphere (± 1 psig).

8.1.3.2.9 Perform the leak test to the requirements of Section 8.1.3.6. If after repeated attempts, the body O-ring seal fails to pass the leak test, isolate the leak path and, prior to repairing the leak path and repeating the body O-ring seal leak test, record on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition in accordance with the requirements of Section 9.2.15.

8.1.3.3 Testing the Vent Port Plug Seal Integrity

8.1.3.3.1 The Fabrication Verification Leak Test of the vent port plug seal shall be performed per Section A3.10.1, Helium Mass Spectrometer Envelope, Evacuated Envelope, Reference 8.3.1.4.

8.1.3.3.2 The cask shall be assembled with both body main O-ring seals installed in the cask lid. Assembly is as shown on the Packaging General Arrangement Drawing in Section 1.4.

8.1.3.3.3 Remove the vent port cover and open the plug to allow free passage of gas through the vent port.

8.1.3.3.4 Connect a vacuum pump to the vent port.

8.1.3.3.5 Evacuate the cask cavity through the vent port to less than 1 psia.

8.1.3.3.6 Prior to closing of the vent port plug, provide a helium atmosphere in the cask cavity by backfilling with helium to one atmosphere (± 1 psig).

8.1.3.3.7 Close the vent port plug.

8.1.3.3.8 Install the helium mass spectrometer leak detector to the vent port.

8.1.3.3.9 Evacuate the vent port until the vacuum is sufficient to operate the helium mass spectrometer leak detector per the manufacturer's recommendations.

8.1.3.3.10 Perform the leak test to the requirements of Section 8.1.3.6. If after repeated attempts, the vent port O-ring seal fails to pass the leak test, isolate the leak path and, prior to repairing the leak path and repeating the vent port O-ring seal leak test, record on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition in accordance with the requirements of Section 9.2.15.

8.1.3.4 Testing the Rupture Disc Sealing Integrity

8.1.3.4.1 Provide a support structure for the cask lid to allow access to the inner and outer face at the same time.

8.1.3.4.2 Utilizing appropriate fittings, attach a vacuum pump and a source of helium gas, in parallel, to the rupture disc port on the inner face of the cask lid. Install a valve into each of the two lines to allow independent isolation of the vacuum pump and the source of the helium gas to the rupture disc port.

8.1.3.4.3 Utilizing appropriate fittings, attach the leak detector to rupture disc port on the outer face of the cask lid.

8.1.3.4.4 Evacuate the cavity above the rupture disc (outer face of the cask lid) until the vacuum is sufficient to operate the leak detector as per the manufacturer's recommendations.

8.1.3.4.5 Use the vacuum pump attached to the inner face of the cask lid to evacuate the cavity below the rupture disc to less than 1 psia. Isolate the vacuum pump from the system.

8.1.3.4.6 Open the valve allowing free passage of helium gas into the evacuated cavity below the rupture disc. When the pressure in the cavity reaches one atmosphere (± 1 psig), close the valve to isolate the source of helium gas from the system.

8.1.3.4.7 Perform the leak test to the requirements of Section 8.1.3.6. If after repeated attempts, the rupture disc fails to pass the leak test, isolate the leak path and, prior to repairing the leak path and repeating the rupture disc leak test, record on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition in accordance with the requirements of Section 9.2.15.

8.1.3.5 Testing the Drain Port Plug Seal Integrity

8.1.3.5.1 The Fabrication Verification Leak Test of the vent port plug seal shall be performed per Section A3.10.1, Helium Mass Spectrometer Envelope, Evacuated Envelope, Reference 8.3.1.4.

8.1.3.5.2 The cask shall be assembled with both body O-ring seals installed in the cask lid. Assembly is as shown on the Packaging General Arrangement Drawing in Section 1.4.

8.1.3.5.3 Remove the vent port cover and open the plug to allow free passage of gas through the vent port.

8.1.3.5.4 Connect a vacuum pump to the vent port.

8.1.3.5.5 Evacuate the cask cavity through the vent port to less than 1 psia.

8.1.3.5.6 Prior to closing of the vent port, provide a helium atmosphere in the cask cavity by backfilling with helium to one atmosphere (± 1 psig).

8.1.3.5.7 Close the vent port.

8.1.3.5.8 Install the helium mass spectrometer leak detector to the drain port.

8.1.3.5.9 Evacuate the drain port until the vacuum is sufficient to operate the helium mass spectrometer leak detector per the manufacturer's recommendations.

8.1.3.5.10 Perform the leak test to the requirements of Section 8.1.3.6. If after repeated attempts, the drain port O-ring seal fails to pass the leak test, isolate the leak path and, prior to repairing the leak path and repeating the drain port O-ring seal leak test, record on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition in accordance with the requirements of Section 9.2.15.

8.1.3.6 Acceptance Criteria

8.1.3.6.1 To be acceptable, per Section 5.4(3), Reference Air Leakage Rate, of Reference 8.3.1.4, the containment vessel shall have a leakage rate of 1×10^{-7} standard cubic centimeters per second (leaktight), or less.

8.1.3.6.2 In order to demonstrate that a cask is leaktight, per Section 7.3.2, Leakage Tests, of Reference 8.3.1.4, the sensitivity of the leak test equipment shall be 5×10^{-8} standard cubic centimeters per second, or less.

8.1.4 Polyurethane Foam Test

Prior to foam installation, the internal cavity bounded by the stainless steel shell of the impact limiters shall be cleaned of scale, oils, grease, debris and be coated with a foam release agent. These surfaces shall then be washed with a solvent or cleaner compatible with the chemicals present in the foam. The liquid foam material shall be poured directly into the impact limiter cavity. Each pour shall be carefully controlled so that the liquid components will react to form the rigid foam, and will rise in such a way that the entire volume of the single foamed impact limiter will be filled with expanded foam with a density of approximately fifteen pounds per cubic foot (15 pcf). The direction of foam rise shall be parallel to the centerline of the NuPac 140-B Cask. The temperature of the mixed foam components shall be maintained at a minimum of 60 °F. The surrounding walls of the impact limiter being foamed shall be maintained at a temperature between 55 °F and 85 °F.

The foam shall be poured in small batches to ensure uniformity of the final foam impact limiter. The time delay between multiple pours into a single impact limiter shall be no less than 20 hours, per the foam manufacturer's recommendations, and the level of each batch pour shall be recorded.

Bracing and shoring for all surrounding impact limiter walls shall be provided as necessary to prevent distortion due to internal foam pressures. The method used for bracing must allow the impact limiter to meet the required dimensions and dimensional tolerances as specified on the Packaging General Arrangement Drawing in Section 1.4.

Production records for each foam pouring operation shall be compiled during the operation. At a minimum, the record shall include pour dates, operator name, impact limiter part number and serial number, Quality Assurance buy-off, and material traceability identified by batch number. This record shall be retained for the life of the NuPac 140-B Cask.

On completion of production, a certification referencing the production record data and all testing data pertaining to each set of impact limiters for a NuPac 140-B Cask shall be issued to Nuclear Packaging by the foam supplier. Test data relevant to the pouring operation shall be included with the certification. All QA data submittals shall be dated and signed by the foam supplier's designated QA representative.

Each production pour made into an impact limiter, and each sample pour made during each production pour for test purposes shall be completely recorded. Test sample pours shall be foamed in test boxes or containers at the same time as the actual production pour they represent. A portion of the actual production pour stream shall be utilized for the corresponding sample. Test foam material shall be taken from each test sample box prepared during production. Test foam material from each pour shall be tested prior to installation of the next pour and the level of each batch pour shall be recorded.

The test foam material shall be tested for compressive strength within the parameters presented in Section 2.3, Mechanical Properties of Materials, in accordance with the requirements of Reference 8.3.1.6. Stress-strain plots, similar to those shown in Figures 2.3-1 and 2.3-2 shall be prepared for both the parallel-to-rise and perpendicular-to-rise orientations. The test data shall be recorded and reviewed to ensure compliance with the foam structural acceptance criteria outlined in Section 2.3.

The test foam material shall be tested for fire retardancy within the parameters presented in Section 3.2, Thermal Properties of Materials, in accordance with the requirements of Section 853(c) of Reference 8.3.1.7. The test data shall be recorded and reviewed to ensure compliance with the foam thermal acceptance criteria outlined in Section 3.2.

Polyurethane foam not in compliance with the above requirements shall be recorded on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition prior to repair and final acceptance in accordance with the requirements of Section 9.2.15.

8.1.5 Lead Inspection and Installation

8.1.5.1 Lead Inspection

Poured lead shielding integrity shall be confirmed via gamma scanning. There are two gamma scan techniques utilized. The main difference is in the method utilized to determine acceptance criteria. Both gamma scan techniques are exactly the same in all other respects and are conducted as follows:

An Eberline El20 probe, or equivalent, is used to scan the outer cask surface while an Iridium 192 or Cobalt 60 source of sufficient strength is present at the center of the cask inner cavity. The source is first placed on the bottom of the cask cavity while the surface is scanned around its circumference, parallel to the source. The source is then moved up a predetermined distance and the circumference scanned again. This sequence is repeated until the entire cask surface is scanned.

The cask outer surface is gridded and a chart is made to reflect the gridded surface. Dose rates are recorded from each grid square by scanning every point in the grid and recording the maximum dose rates in the corresponding grid on the chart. This data then serves as the raw gamma scan results. All dose rates are in milliroentgens per hour (mR/hr).

The dose rates are evaluated by comparing them to predetermined dose rate values for nominal (as designed) lead thickness and nominal less 10% lead thickness. The two different methods utilized to determine acceptance criteria are discussed below:

The first, the Laboratory Calibration Method (NuPac Procedure GS-001), utilizes test blocks of the cask wall made up of lead and steel sheets. The test blocks simulate nominal, or as designed, and nominal less 10% lead thicknesses. The source is placed behind the nominal test block assembly at a distance equal to the inside radius of the cask. The probe is then placed on the outside of the test block assembly and dose readings are recorded.

This test sequence is repeated on the nominal less 10% test block assembly and the resultant dose value is the maximum acceptable dose value for the inspected cask. Additionally, the expected dose rate values for the nominal and reduced thickness shielding are calculated utilizing attenuation values for air, steel and lead as a correlation verification.

The second, the Field Calibration Method (NuPac Procedure GS-002), utilizes a specially fabricated test lid which incorporates a holder for various lead and steel sheet thicknesses. This fixture is installed onto the cask to be scanned, with the test lid set up to simulate the nominal lead thickness. The source is placed below the test lid, inside the cask, at a distance equal to the inside radius of the cask. A dose reading is then recorded. The test lid is adjusted to establish the nominal less 10% lead thickness configuration. Again, the source is placed below the test lid at a distance equal to the inside radius of the cask and a dose reading is recorded. Additional dose readings are taken in 1/8 inch lead thickness increments between and beyond the two base readings until four to eight dose readings are obtained. The resultant data is then plotted on a chart for dose values versus lead thickness. The value for nominal less 10% lead thickness is utilized as the maximum acceptable dose value for the inspected cask.

Lead shielding installation requires measurements before and after the lead pour process of the diameter and straightness of the cask inner shell. Also, cask temperatures during the preheat, lead pour and cool-down process are required to be closely monitored and controlled. A detailed description of the lead installation process and required measurements are described below.

8.1.5.2 Lead Installation Procedure

The following delineates the procedure for installation of poured lead into the NuPac 140-B cask. The procedure presents pre-pour, cask inner shell dimensional inspection requirements, lead pour, gamma scan inspection of the lead shielding, and post-pour, cask inner shell dimensional inspection requirements. Note: The cask thermal shield is attached after lead pour and cool down.

8.1.5.2.1 Record the cask, inner shell, inside diameter at five locations approximately four equal spaces along the inside diameter length, and at four points 45° apart along the circumference at each of the five locations. Using a straightedge, or equivalent, which runs from one point (A) to another point (B) obtain surface variation, if any, at three locations approximately equally spaced along the inside diameter length and at four points, 90° apart, along the circumference. Utilize a permanent marker, or equivalent, to record the locations where measurements are taken to ensure repeatability for the post-pour measurements. Record the ambient temperature at the time the roundness and straightness measurements are taken.

8.1.5.2.2 Interior cavities and surfaces of the cask shells which are to be in contact with the lead shielding will be swabbed with solvent, wiped down, sandblasted, or otherwise cleaned of loose mill scale, weld slag, and all carbonaceous materials prior to lead pour.

8.1.5.2.3 All cask inner and outer surfaces shall be supported or braced, as necessary, to prevent distortion during the lead pouring and cool-down operations.

8.1.5.2.4 The lead pouring operation is performed with the cask inverted (i.e., cask base plate is up, lid end is down)..

Fill pipes shall be provided through the outer shell of the cask, adjacent to the lid-end forging, sized as required to allow for rapid and even lead fill.

Standpipes shall be provided through the outer shell of the cask, adjacent to the base plate, sized as required to allow for easy access of lead during the cooling operation.

Vent risers shall be provided through the cask base plate to allow for escape of air and impurities.

8.1.5.2.5 The cask will be instrumented with thermocouple wires at twelve (12) locations on the inside and outside to monitor temperature differentials between the two shells. A total of twenty four (24) thermocouples will be required. Internal and/or external bracing may be temporarily removed to facilitate thermocouple installation. The thermocouples will be mounted opposite each other on the inside and outside shells to assure proper temperature differential measurements. The thermocouple sets shall be mounted at equal intervals over the full length of the cask. The sets shall also be located around the cask circumference so that all areas of the cask are instrumented.

In addition, a calibrated strip recorder with the thermocouples capable of measuring temperatures to 1000° F and plotting time/temperature data every 60 seconds shall be utilized.

Adequate precautions must be taken to ensure that the thermocouples are protected from impact or flame impingement during the preheat, lead pour, and cool-down phases.

8.1.5.2.6 All cask components (i.e., outer shell, inner shell, top end forging and bottom end closure) will be uniformly preheated to a temperature of 600° F to 700° F over the entire surface prior to lead pouring. Preheating shall be done in a suitable enclosure to assure uniform preheating. During preheat, the maximum temperature change for any thermocouple shall not exceed 100° F in any thirty (30) minute period.

8.1.5.2.7 The temperature of the lead at the time of lead pour shall range from 750° F to 850° F.

8.1.5.2.8 The rate of lead flow will be controlled to fill the cask as rapidly and evenly as possible. The lead shall enter the cavity in such a way as to minimize impingement on the cask shells.

8.1.5.2.9 Heat sources shall be controlled during the cool-down period so that the cask is cooled to ensure that only one solidifying front exists in the lead. During cool-down, the maximum temperature change for any thermocouple shall not exceed 100° F in any thirty (30) minute period, and the maximum difference in temperatures between any pair of thermocouples mounted directly opposite each other (one on the OD and the corresponding one on the ID) shall not exceed 300° F at any instant in time.

8.1.5.2.10 Molten lead at a minimum temperature of 750° F shall be added to the standpipes, as necessary, at a rate consistent with normal shrinkage.

8.1.5.2.11 After all the thermocouples indicate a temperature less than 100° F, the thermocouples and any internal and/or external bracing shall be removed and the cask gamma scanned in accordance with the tests delineated in Section 8.1.5.

8.1.5.2.12 Upon a successful gamma scan inspection, the inside diameter and straightness measurements, described in Step 8.1.5.2.1, shall be retaken. To be acceptable, the following must be confirmed (Note: the significance of these criteria is discussed in Appendix 2.10.3).

1. All straightness and all diameter measurements taken before and after lead pour must meet the requirements specified on the Packaging General Arrangement Drawing in Section 1.4 (i.e., the 0.125 inch straightness requirement and the tolerance requirements of ASME, Section III, Subsection NE, Article NE-4220).
2. No individual diameter measurement shall be allowed to change by more than 0.125 inches in going from pre-pour to post-pour conditions.

8.1.5.2.13 Remove all fill, vent, and standpipes, clean the holes thoroughly of all lead, and install plugs of identical material as the base material.

8.1.6 Thermal Acceptance Tests

Material properties established in Section 3.0, THERMAL EVALUATION, are consistently conservative for the analyses performed. As such, with the exception of the polyurethane foam (addressed in Section 8.1.4, above), acceptance tests for material thermal properties are not performed.

8.1.7 Criticality Control Materials Acceptance Tests

Criticality control materials are installed in a NuPac 140-B Cask basket assemblies (PWR and BWR). A sample of the criticality control material shall be retained from each production batch for testing to verify elemental composition.

The acceptance criterion shall be that the actual boron and boron-10 weight percents within the criticality control materials meet or exceed the corresponding values used in the criticality evaluations in Chapter 6.0.

8.2 Maintenance Program

This section describes the maintenance program used to ensure continued performance of the NuPac 140-B Cask.

8.2.1 Structural and Pressure Tests

Other than the tests required prior to first use, no structural or pressure tests are necessary to ensure continued performance of the cask.

8.2.2 Leak Tests

Maintenance Verification Leak Testing shall conform to the requirements of Section 6.4 of Reference 8.3.1.4.

Appropriate sections of the Maintenance Verification Leak Test shall be performed during routine maintenance to verify cask configuration and performance to design criteria. Maintenance Verification Leak Tests of the main (containment) O-ring seal, the vent and drain port plug seals and the rupture disc are to be performed upon replacement, but seals and rupture disc need not necessarily be replaced on an annual basis or after completion of seal surface repair.

Maintenance Verification Leak Tests shall be performed on the containment boundary seals of the cask containment vessel and on the rupture disc in the cask lid. Four (4) separate tests comprise the series. Each test shall meet the acceptance criteria delineated in Section 8.2.2.5, below.

8.2.2.1 Testing the Body O-ring Seal Integrity

8.2.2.1.1 The Maintenance Verification Leak Test of the body O-ring seal shall be performed per Section A3.10.1, Helium Mass Spectrometer Envelope, Evacuated Envelope, Reference 8.3.1.4.

8.2.2.1.2 The cask shall be assembled with both body O-ring seals installed in the cask lid. Assembly is as shown on the Packaging General Arrangement Drawing in Section 1.4.

8.2.2.1.3 Remove the vent port cover and open the plug to allow free passage of gas through the vent port.

8.2.2.1.4 Connect a vacuum pump to the vent port.

8.2.2.1.5 Evacuate the cask cavity through the vent port to less than 1 psia.

8.2.2.1.6 Install the helium mass spectrometer leak detector to the seal test port.

8.2.2.1.7 Evacuate the seal test port until the vacuum is sufficient to operate the helium mass spectrometer leak detector per the manufacturer's recommendations.

8.2.2.1.8 Provide a helium atmosphere within the interior of the cask by backfilling with helium gas to one atmosphere (± 1 psig).

8.2.2.1.9 Perform the leak test to the requirements of Section 8.2.2.5. If after repeated attempts, the cask body O-ring seal fails to pass the leak test, isolate the leak path and, prior to repairing the leak path and repeating the cask body O-ring seal leak test, record on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition in accordance with the requirements of Section 9.2.15.

8.2.2.2 Testing the Vent Port Plug Seal Integrity

8.2.2.2.1 The Fabrication Verification Leak Test of the vent port plug seal shall be performed per Section A3.10.1, Helium Mass Spectrometer Envelope, Evacuated Envelope, Reference 8.3.1.4.

8.2.2.2.2 The cask shall be assembled with both body O-ring seals installed in the cask lid. Assembly is as shown on the Packaging General Arrangement Drawing in Section 1.4.

8.2.2.2.3 Remove the vent port cover and open the plug to allow free passage of gas through the vent port.

8.2.2.2.4 Connect a vacuum pump to the vent port.

8.2.2.2.5 Evacuate the cask cavity through the vent port to less than 1 psia.

8.2.2.2.6 Prior to installation of the vent port plug, provide a helium atmosphere in the cask cavity by backfilling with helium to one atmosphere (± 1 psig).

8.2.2.2.7 Close the vent port plug.

8.2.2.2.8 Install the helium mass spectrometer leak detector to the vent port.

8.2.2.2.9 Evacuate the vent port until the vacuum is sufficient to operate the helium mass spectrometer leak detector per the manufacturer's recommendations.

8.2.2.2.10 Perform the leak test to the requirements of Section 8.2.2.5. If after repeated attempts, the vent port O-ring seal fails to pass the leak test, isolate the leak path and, prior to repairing the leak path and repeating the vent port O-ring seal leak test, record on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition in accordance with the requirements of Section 9.2.15.

8.2.2.3 Testing the Rupture Disc Sealing Integrity

8.2.2.3.1 Provide a support structure for the cask lid to allow access to the inner and outer face at the same time.

8.2.2.3.2 Utilizing appropriate fittings, attach a vacuum pump and a source of helium gas, in parallel, to the rupture disc port on the inner face of the cask lid. Install a valve into each of the two lines to allow independent isolation of the vacuum pump and the source of the helium gas to the rupture disc port.

8.2.2.3.3 Utilizing appropriate fittings, attach the leak detector to the rupture disc port on the outer face of the cask lid.

8.2.2.3.4 Evacuate the cavity above the rupture disc (outer face of the cask lid) until the vacuum is sufficient to operate the leak detector as per the manufacturer's recommendations.

8.2.2.3.5 Use the vacuum pump attached to the inner face of the cask lid to evacuate the cavity below the rupture disc to less than 1 psia. Isolate the vacuum pump from the system.

8.2.2.3.6 Open the valve allowing free passage of helium gas into the evacuated cavity below the rupture disc. When the pressure in the cavity reaches one atmosphere (± 1 psig), close the valve to isolate the source of helium gas from the system.

8.2.2.3.7 Perform the leak test to the requirements of Section 8.2.2.5. If after repeated attempts, the rupture disc port fails to pass the leak test, isolate the leak path and, prior to repairing the leak path and repeating the rupture disc port leak test, record on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition in accordance with the requirements of Section 9.2.15.

8.2.2.4 Testing the Drain Port Plug Seal Integrity

8.2.2.4.1 The Fabrication Verification Leak Test of the drain port plug seal shall be performed per Section A3.10.1, Helium Mass Spectrometer Envelope, Evacuated Envelope, Reference 8.3.1.4.

8.2.2.4.2 The cask shall be assembled with both body O-ring seals installed in the cask lid. Assembly is as shown on the Packaging General Arrangement Drawing in Section 1.4.

8.2.2.4.3 Remove the drain port cover and open the plug to allow free passage of gas through the drain port.

8.2.2.4.4 Connect a vacuum pump to the drain port.

8.2.2.4.5 Evacuate the cask cavity through the drain port to less than 1 psia.

8.2.2.4.6 Prior to closing of the drain port plug, provide a helium atmosphere in the cask cavity by backfilling with helium to one atmosphere (± 1 psig).

8.2.2.4.7 Close the drain port plug.

8.2.2.4.8 Install the helium mass spectrometer leak detector to the drain port.

8.2.2.4.9 Evacuate the drain port until the vacuum is sufficient to operate the helium mass spectrometer leak detector per the manufacturer's recommendations.

8.2.2.4.10 Perform the leak test to the requirements of Section 8.2.2.5. If after repeated attempts, the drain port O-ring seal fails to pass the leak test, isolate the leak path and, prior to repairing the leak path and repeating the drain port O-ring seal leak test, record on a Quality Discrepancy Report/Supplier Disposition Request for Material Review Board disposition in accordance with the requirements of Section 9.2.15.

8.2.2.5 Acceptance Criteria

8.2.2.5.1 To be acceptable, per Section 5.4(3), Reference Air Leakage Rate, of Reference 8.3.1.4, the containment vessel shall have a leakage rate of 1×10^{-7} standard cubic centimeters per second (leaktight), or less.

8.2.2.5.2 In order to demonstrate that a package is leaktight, per Section 7.3.2, Leakage Tests, of Reference 8.3.1.4, the sensitivity of the leak test equipment shall be 5×10^{-8} standard cubic centimeters per second, or less.

8.2.3 Subsystems Maintenance

8.2.3.1 Fasteners

All threaded parts shall be inspected prior to each use for deformed, excessively worn or stripped threads. Any damaged parts shall be replaced prior to further use. At a minimum, cask threaded fasteners shall be replaced every five (5) years.

In addition, before each use, visually inspect the bolts and the test port and vent port covers. Any damaged parts shall be replaced prior to further use. Covers are not required to be replaced except if damaged.

8.2.3.2 Impact Limiters

An inspection program shall be followed to ensure the integrity of the impact limiters. The impact limiters attachment bolts threads shall be inspected annually and prior to each use for damaged or stripped threads. Any damaged parts shall be repaired or replaced prior to further use. Additionally, the plastic pipe plugs in the impact limiters shall be inspected for damage and replaced prior to further use if damage is present. The solar reflective white painted on external surfaces shall be inspected, periodically cleaned and/or repainted if more than 10% of the coverage is lost.

8.2.3.3 Trunnions

The trunnions shall be inspected annually and before each use for excessive wear, galling or distortion. Trunnions exhibiting such damage shall be corrected prior to further use.

8.2.3.4 Seal Areas and Grooves

8.2.3.4.1 Seal Area Routine Inspection and Repair

Before each use and at the time of seal replacement, cask sealing surfaces and O-ring seal grooves shall be visually inspected for damage that could impair the sealing capabilities of the NuPac 140-B Cask. Damage shall be corrected prior to further use (e.g., using 400-600 grit emery cloth to polish the sealing surfaces) to the surface finish specified in Section 8.2.3.4.2.

Upon completion of seal area and/or groove repairs, perform a seal area inspection per Section 8.2.3.4.2, and leak test per the applicable section of the Maintenance Verification Leak Test per Section 8.2.2.

8.2.3.4.2 Annual Seal Area Inspection

In order to maintain an effective seal between the cask body and lid, an annual inspection of the seal area surface finishes shall be performed. The surface finish in the sealing areas shall be maintained at a maximum 125 micro-inch finish. The following surfaces shall be measured annually:

- 1) Bottom surface of each seal O-ring groove, and
- 2) Surface of the cask body O-ring sealing area.

If the surface condition of the above sealing areas is determined to exceed 125 micro-inches, repair the sealing area(s) per the requirements of Section 8.2.3.4.1.

8.2.4 Valves, Rupture Discs, and Gaskets on the Containment Vessel

This section describes the inspection and replacement schedule for these components.

8.2.4.1 Valves

The NuPac 140-B Cask does not contain any valves.

8.2.4.2 Rupture Discs

The rupture disc in the cask lid shall be replaced annually or when damaged. Upon replacement, perform the Maintenance Verification Leak Test upon the rupture disc.

8.2.4.3 Gaskets

All O-ring seals shall be replaced annually or when damaged, per size and material specifications as delineated on the Packaging General Arrangement Drawing in Section 1.4. Following O-ring seal replacement for containment boundary seal(s), and prior to a loaded shipment, the new seal(s) shall be leak tested to the requirements of Section 7.4.2 Assembly Verification Leak Test.

8.2.5 Shielding

Other than the tests required prior to first use, no shielding tests are necessary to ensure continued performance of the cask.

8.2.6 Thermal

Thermal inspections or tests necessary to ensure continued performance of the NuPac 140-B Cask shall be performed as determined during the final design phase of the project.

8.2.7 Criticality Control Materials

Tests to be required during use are to be determined by the criticality control material supplier(s).

8.3 Appendix

8.3.1 References

Appendix 8.3.1

References

8.3.1 References

8.3.1.1 Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), Packaging and Transportation of Radioactive Materials, August 24, 1983.

8.3.1.2 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

8.3.1.3 ANSI N14.6-1986, American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More.

8.3.1.4 ANSI N14.5-1987, American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment.

8.3.1.5 ASTM D1622-63, Method of Test for Apparent Density of Rigid Cellular Plastics.

8.3.1.6 ASTM D1623-73, Method of Test for Tensile Properties of Rigid Cellular Plastics.

8.3.1.7 Title 14, Code of Federal Regulations, Part 25 (14 CFR 25), Airworthiness Standards: Transport Category Airplanes, July 21, 1986.

9.0 QUALITY ASSURANCE

The NuPac 140-B Cask is being designed and will be fabricated by Nuclear Packaging, Inc., (NuPac) Federal Way, Washington. The Quality Assurance Program used for the design, fabrication, assembly, testing, use and maintenance of the NuPac 140-B cask satisfies the eighteen (18) criteria of 10 CFR 71, Subpart H in its entirety. Pacific Nuclear Systems, Inc. (PNSI) Quality Assurance Program meeting these criteria has been submitted to the United States Nuclear Regulatory Commission and has been awarded Approval Number 0192, Revision 2.

The Quality Program is implemented throughout the company and its subsidiaries. The Subsidiaries include: Nuclear Packaging, Inc., NuPac Services, Inc., and PN Services, Inc.

A synopsis of the Pacific Nuclear Systems, Inc./Nuclear Packaging, Inc. Quality Assurance Program follows:

9.1 Introduction

PNSI has developed a quality program to (1) assure traceability, and (2) control the quality of all materials and processes utilized in the production of radioactive shielding, casks, containers, and other equipment pertaining to shipping packages for irradiated fuel, high level waste, and plutonium.

A Quality Manual delineates requirements and procedures necessary to exercise control over design, documentation, procurement, material, fabrication, inspection, operational testing, equipment operation and use, maintenance, repair, modification, inventory, shipment and quality data retention.

The PNSI Quality Program is implemented by Quality Procedures which are designed and administered to meet the 18 criteria of 10 CFR 71, Subpart H.

9.2 Description of the PNSI, 10 CFR 71, Subpart H Quality Program

9.2.1 Organization

Full responsibility for the Quality Assurance (QA) Program adherence to 10 CFR 71, Subpart H criteria rests with PNSI. Quality Program activities include calibration of measuring equipment, non-destructive examination (NDE), and materials testing. PNSI surveys and qualifies all organizations performing these services to assure adherence to the applicable sections of the 18 criteria prior to their use. All other quality activities are performed by PNSI quality personnel. However, the responsibility of the control of quality in the other organizations continues to rest with PNSI.

PNSI's President has full authority over all functions of the company, and delegates authority and responsibility for selected functions to other personnel within the company.

The administrative function includes financial, legal, and marketing activities.

Procurement department personnel perform purchasing activities and maintain supplier performance records. The Engineering Department is responsible for research and development of shipping container technology, design of casks for licensing and fabrication, and design documentation.

The NuPac Quality Department has sufficient authority and organizational freedom to identify quality programs, implement corrective action, and verify corrective action effectiveness.

Additionally, the Quality Department is independent from other organizations within NuPac and reports directly to the PNSI Corporate Director of Quality Assurance. The Quality Department is headed by the NuPac Quality Assurance Manager who is responsible for the development, implementation, and administration of the entire NuPac Quality Program. He must have sufficient expertise in the entire field of Quality to enable him to direct the entire quality function in close adherence to the 18 criteria of 10 CFR 71 and the PNSI Quality

Manual. Responsibility for development of quality acceptance requirements, inspections, and NDE activities rests with the NuPac Quality Assurance Manager. It is his responsibility to delegate and evaluate the performance of all quality related tasks for NuPac through the authority of the Corporate Director of Quality Assurance.

It is delineated in writing through the Corporate Director of Quality Assurance that QA personnel have the authority to prevent the continued processing, fabrication, installation, or delivery of unsatisfactory work.

Production responsibilities include scheduling or administration of all fabrication activities, both within NuPac and at qualified suppliers. The shipping and receiving function is also the responsibility of the Production Department.

The NuPac Quality Assurance Manager and all other quality personnel and/or organizations within, or utilized by NuPac, are fully qualified for their quality responsibilities. Qualification records are maintained in the NuPac Quality Record File.

9.2.2 Quality Assurance Program

NuPac has established and implemented a QA Program for the control of quality in the design, fabrication, operation, and maintenance of shipping containers for nuclear products. Training and/or evaluation of personnel qualifications are required for all QA functions in accordance with written procedures and are approved by the Quality Assurance Manager. The QA Program assures that all quality requirements, engineering specifications, and specific provisions of any package design approval are met. Those characteristics critical to safety are emphasized.

The Corporate Director of Quality Assurance regularly evaluates the NuPac QA program for adherence to the 18 criteria in scope, implementation, and effectiveness.

Further, the Quality Assurance Director requires that the Quality System, including the QA Manual Policies and Procedures, be implemented and enforced on all applicable programs at NuPac.

A Material Review Board, consisting of Engineering, Production, and Quality Personnel has been established to resolve all discrepancies or disagreements pertaining to the acceptability of material, hardware, or safety related operations. Their dispositions are final and binding.

9.2.3 Design Control

PNSI Quality Procedures (QP's) have been developed, approved, and implemented to control design review in such a manner to assure that the following occur:

- 9.2.3.1 Design activity is planned, controlled, and documented.
- 9.2.3.2 Regulatory and design requirements are correctly translated into specification, drawings, and procedures.
- 9.2.3.3 Design documents contain quality requirements.
- 9.2.3.4 Deviations from quality requirements are controlled.
- 9.2.3.5 Design verification is performed by qualified personnel independent of the design activity. These verifications may include tolerance studies, alternate calculations or tests. Qualification tests are conducted in accordance with approved test programs and procedures.
- 9.2.3.6 Interface control is established and adequate.
- 9.2.3.7 Design and specification changes are reviewed and approved by the same organization(s) as the original issue.
- 9.2.3.8 Design errors and deficiencies are documented and corrective action is taken to prevent recurrence.

9.2.3.9 Design organization(s) and their responsibilities and authorities are delineated and controlled via written procedure.

9.2.4 Procurement Document Control

The PNSI QA Program assures that all purchased material, components, equipment, and services adhere to design specifications.

Supplier evaluation and selection, objective evidence of supplier quality, assignment of quality requirements to procurement documents and related design documents, and source, in-process, and receiving inspections are all administered and controlled in accordance with approved PNSI QA procedures.

All procurement activities are performed in accordance with written procedures delineating requirements for preparation, review, approval, and control of procurement documentation. Revisions to procurement documentation are reviewed and approved by the same cognizant groups as the original.

Quality Assurance clause sheets are included with request for quotes and purchase orders in accordance with PNSI Quality Assurance procedures. Quality Assurance personnel assign clauses from the sheets to the procurement document referencing 10 CFR 71, Subpart H requirements appropriate to the contract. In addition, material information including grade, type, size, and special physical or chemical data requirements is included on the procurement documents. Other documentation and information such as drawings, procedures, inspection and test requirements, hold points, welding and other process qualification requirements are delineated on the procurement documents by the Quality Assurance personnel as appropriate to the contract.

Quality Assurance personnel assure that requirements for acceptance of hardware and documentation appropriate to the contract are included in procurement documentation.

NuPac Quality Assurance personnel maintain the right of access to all supplier facilities and documentation for source inspection and/or audit activities. A statement to this effect is included on procurement documentation when it is appropriate to the contract.

9.2.5 Instruction, Procedures and Drawings

Quality planning is developed by qualified Quality Engineers (QE's) for all activities requiring quality participation in accordance with approved PNSI QA procedures and is approved by the NuPac Quality Assurance Manager.

All design documents (i.e., drawings, specifications, special processes, etc.) affecting quality are reviewed by the Quality Department and referenced in quality planning as necessary to assure adherence to package design approvals and the applicable criteria of 10 CFR 71, Subpart H.

All instructions, procedures, and drawings are developed, reviewed, approved, utilized, and controlled in accordance with the requirements of written quality assurance procedures.

9.2.6 Document Control

Policy and procedure for review, approval, release and change control of all controlled, quality related documents are delineated in approved PNSI QA Procedures. Provisions are provided in the QA Procedures for identification of individuals/organizations responsible for review, approval, and issuance of documents. Document control responsibilities, facilities, and distribution requirements are also addressed.

Controlled documents include, but are not limited to:

- (a) Design specifications
- (b) Design manufacturing drawings

- (c) Special process specifications and procedures
- (d) Procurement documents
- (e) QA Procedures and manuals
- (f) Quality Planning for receiving, in-process, source and in-service inspections
- (g) Source surveillance and evaluation reports
- (h) Test procedures
- (i) Audit reports
- (j) Operational test procedures and data.

When revised documents appear in other documents as references, supplements, or exhibits, appropriate revisions are made to those documents prior to the release of the basic approved change.

Documentation listings are maintained delineating the title, number and current revision for all drawings, procedures, specifications, and purchase orders.

The Quality Personnel assure that all required support documentation is available at the work area prior to the initiation of the work effort.

9.2.7 Control of Purchased Materials, Parts and Components

Procurement documents are reviewed for acceptability of suggested suppliers based on the NuPac approved supplier lists.

In addition, and as required, supplier surveys are conducted by qualified NuPac personnel to further assure supplier acceptability. These evaluations are based on one or all of the following criteria:

- (a) The supplier's capability to comply with the requirements of 10 CFR 71, Subpart H, that are applicable to the contract.
- (b) A review of previous records and performance of the supplier.
- (c) A survey of the supplier's facilities and QA program to determine his capability to supply a product which meets the design, manufacturing, and quality requirements.

Results of all supplier evaluations are recorded on Supplier Evaluation forms and are retained in the Quality Data File.

Quality requirements and standard clauses are added to procurement documents to require suppliers to identify material, provide test reports, control special processes, certify equipment and personnel, etc. As a minimum, requirements are imposed on suppliers to identify materials, specific codes, specifications and/or design not adhered to during fabrication. Justifications for 'accept-as is' or 'repair' dispositions are also required to be submitted to the Material Review Board for review and acceptance.

Quality planning is prepared and approved by the Quality Department for performance of all source, test, shipping and/or receiving inspections in accordance with approved design requirements, applicable 10 CFR 71 criteria, procurement document requirements, and contract specifications.

Receiving inspection is performed to determine that the following, as appropriate to the contract, are assured:

- (a) The material, component, or equipment is properly identified and corresponds with the identification on receiving documentation.
- (b) Material, components, equipment, and acceptance records are inspected and are acceptable in accordance with inspection instructions, prior to installation or use.
- (c) Inspection records and/or certificates of conformance attesting to the acceptance of material and components are available prior to installation or use.
- (d) Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for further work.

All described activities are delineated in approved PNSI QA procedures.

9.2.8 Identification and Control of Materials, Parts, and Components

The identification and control of materials, parts, components, and completed and in-process assemblies is administered by the Quality Department in accordance with approved PNSI QA Procedures. These procedures address quality status tags, maintenance of material identification and traceability, part identification, and related documentation. Some of the details of these procedures follow:

- (a) Material identification procedures included in inspection planning and fabrication drawings require that identification of material, components, and/or hardware be maintained on the item or in traceable records to prevent use of incorrect or defective items.
- (b) When appropriate, due to contractual or safety related concerns requiring specific identification and Material Review Board action, Quality Assurance personnel assure that identification of materials, components, specifications, procurement documentations, manufacturing, and inspection records, discrepancy reports, and material test data is provided and is complete.

- (c) Quality Assurance personnel assure, via drawings and inspection planning requirements, that identification locations do not affect the fit-up, interfacing capability, performance or overall quality of the finished product. Identification, in accordance with drawings and inspection planning requirements, is verified prior to releasing the item for further processing or delivery.

9.2.9 Control of Special Processes

PNSI approved QA Procedures delineate the policies and procedures established to control such special processes as: welding, heat treating, lead pouring, non-destructive examination, waste processing, etc. in accordance with applicable codes, standards, specifications, 10 CFR 71 criteria, and other requirements. Special processes developed by NuPac suppliers and by NuPac are documented.

All procedures for special processes and the personnel required to perform them are qualified under the cognizance of the Quality Department in accordance with applicable codes, standards, specifications, and contract requirements.

All qualification records and support data are retained in the Quality Data file, and are maintained in a current status by Quality Assurance personnel.

These documents are controlled as delineated in Section 9.2.6 of this Quality System description.

9.2.10 Inspection

All receiving, source, in-process, and in-service inspection activities are performed in accordance with approved PNSI QA procedures. All inspection personnel and/or organization qualifications are reviewed and accepted by the Quality Assurance Manager prior to inspection activity.

The inspection activity is performed in strict accordance with approved quality planning prepared by qualified QA personnel (See also Section 9.2.5 discussion).

Quality Inspection personnel are independent from all other organizations within NuPac and report directly to the Quality Assurance Manager.

Inspection personnel qualifications are based on their capability to perform the required inspection functions in accordance with applicable codes, standards, professional society programs such as the American Society for Quality Control quality technician certification, and PNSI training programs. Qualification reviews are performed periodically to maintain personnel proficiency and assure current qualification.

Mandatory hold points, inspection equipment requirements, accept reject criteria, personnel requirements, characteristics to inspect, variable/attributes recording instructions, reference documentation, and other requirements are included in the inspection planning.

The Quality Assurance department assures that any replacements, modifications, or repairs performed after final acceptance of material, components or hardware are inspected in accordance with the original inspection planning or new planning prepared as appropriate.

9.2.11 Test Control

A test control program, as it applies to quality, is addressed in approved PNSI QA Procedures and assures, via required planning, that all required testing, such as proof and acceptance tests, are identified and performed in accordance with test procedures, design requirements, and limitations. Prerequisites, accept/reject criteria, data recording criteria, instrumentation calibration, environmental conditions, documentation and evaluation requirements, etc. are delineated in the test procedures. Changes to the test procedures are required to be reviewed/approved by the same organization(s) as the original issue.

Whenever equipment, components, and/or assemblies require modification, repairs, or replacement which could result in requirements for re-test or additional testing, Quality Assurance personnel assure, as appropriate, that original or new test inspection planning is prepared and adhered to.

In any case, test results are documented, evaluated, and accepted by qualified personnel as required by the test inspection plan prepared for the test under the cognizance of Quality Assurance personnel.

9.2.12 Control of Measuring and Testing Equipment

Administration of the calibration of measuring equipment and instrumentation is performed by the Quality Department in accordance with approved PNSI QA Procedures. The calibration system assures that all measuring and testing equipment (MTE) used in the acceptance of material, equipment, and assemblies are calibrated and properly adjusted at specified intervals to maintain accuracy within pre-determined limits. Calibration is performed using equipment traceable to national standards. All calibrated equipment is identified and is traceable to the calibration test data.

Whenever MTE are found to be out of calibration during or immediately after use, all items inspected during that period are rejected by inspection and are submitted to review action for possible re-inspection or other appropriate corrective action.

9.2.13 Handling, Storage, and Shipping

PNSI approved QA Procedures require that handling, storage, and shipping requirements adherence verification criteria be included in quality planning. These requirements are designed to prevent damage or deterioration of material and equipment. Information pertaining to shelf life, environment, packaging, temperature, cleaning, handling, preservation, etc., is included as required to meet design, NRC package approval and/or U.S. Department of Transportation shipping requirements.

Shipping documentation preparation, departure and arrival time, and destination data recording are also addressed in the planning, when applicable. Shipping requirements in quality planning must be met prior to release for shipment.

9.2.14 Inspection, Test and Operating Status

The use of inspection status tags, quality inspection stamps, and other means to indicate inspection and test status at, or for, NuPac are delineated in approved PNSI QA Procedures.

The clarity of the status indication, prevention of inspection, and/or test step by-passing, and prohibition of removal or modification of status indications,

except with Quality Department approval/Material Review disposition, is assured via these procedures. The Quality Assurance Department assures via Quality Procedure, interoffice memoranda, training sessions, and audit that all NuPac personnel are aware of and understand the meaning and uses of status tags on all hardware, material, and test set-ups (see also Section 9.2.15 discussion).

9.2.15 Non-conforming Material, Parts, or Components

PNSI approved QA Procedures require that material, components, and equipment that do not conform to requirements are controlled to prevent their inadvertent use. Identification, segregation, discrepancy reporting, disposition of non-conformances by authorized individuals, and re-inspection activities are performed and controlled in strict accordance with these procedures.

Quality Discrepancy Reports (QDR) are utilized by the PNSI quality department to identify discrepant items, describe the discrepancy, and provide disposition and reinspection requirements. The signatures of authorized cognizant personnel are placed on the QDR to signify approval of the disposition. These personnel must be approved by the Quality Assurance Manager and must be from the same groups approving the original design. In conjunction with repair or re-work dispositions, quality assurance personnel provide supplemental inspection planning to verify proper implementation of the QDR disposition. This assures that the item is retested and/or reinspected to a degree at least equal to the original acceptance activity.

9.2.16 Corrective Action

Failures, malfunctions, and deficiencies in material, components, equipment, and services are identified and reported to the Quality Assurance Manager and the Corporate Director of Quality Assurance. The cause of the condition and corrective action necessary to prevent recurrence is identified, implemented and then followed up to verify corrective action effectiveness. All reporting requirements of applicable contractual and regulatory specifications and regulations are adhered to as part of any corrective action activity. Detail requirements for this activity are delineated in approved PNSI QA Procedures.

9.2.17 Quality Assurance Records

A quality records system is in effect at NuPac and is administered in accordance with approved PNSI QA procedures. The purpose of the quality record system is to assure that documented evidence pertaining to quality related activities is maintained and available for use by NuPac, its customers, and/or regulatory agencies as applicable. Quality Records include, but are not limited to, inspection and test records, audit reports, quality personnel qualifications, design reviews, quality related procurement data, supplier evaluation reports, etc.

All records are identified by work order number, part number, contract number, or drawing number as appropriate to the record type. A complete list of

all quality records is maintained and provides cross reference between the different identity methods described above and pinpoints the record location.

Design related records such as calculations, drawings, research and development test reports, etc., are retained in the Quality Assurance records system for the life of the shipping package. All other quality related records are retained for the life of the shipping package in accordance with 10 CFR 71.91(c) unless otherwise specified in related contractual or regulatory requirements.

Inspection records retained in the Quality Assurance records system provide the following data when applicable:

- (a) Inspection type, i.e., in-process, in-service, testing, receiving, and shipping.
- (b) Evidence of completion and verification of manufacturing, inspection, or test operation.
- (c) The date and results of the inspection or test.
- (d) Information related to noted discrepancies.
- (e) Inspector or data recorder identification.
- (f) Evidence of acceptance.

9.2.18 Audits

Quality program audits are performed on a periodic, scheduled basis by personnel without direct responsibilities in the areas being audited. Audit personnel are certified quality assurance lead auditors who have met all requirements of ANSI N 45.2.23. Written planning sheets and check lists are utilized. Audit results and corrective action activity are reported to management, in writing, and are retained in the quality assurance record file. Responsible management personnel are required to respond to audit findings with the necessary action to correct the noted deficiencies. Current PNSI practice is to audit all quality functions on an annual basis. Areas found deficient during audits are reaudited on a first priority basis to verify corrective action implementation and effectiveness. Details of the PNSI Audit System are delineated in approved PNSI QA Procedures.

9.3 References

- 9.3.1 Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), Subpart H, Criteria 1-18 dated August 24, 1983, Quality Assurance Criteria for Shipping Packages for Radioactive Material.
- 9.3.2 PNSI Corporate Quality Manual, dated February 1, 1988
- 9.3.3 ANSI/ASME N45.2.23-1978, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants.