NLI 10/24

Safety Analysis Report

November 1991

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Section I

SECTION I

INTRODUCTION

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SECTION I

INTRODUCTION

This report contains a detailed description and safety analysis of the N L Industries, Inc. NLI 10/24 Universal Spent Fuel Shipping Cask. This cask is designed for rail transportation of either ten (10) PWR type or twenty-four (24) BWR type light water reactor spent fuel assemblies.

The cask design contains several unique features which permit operation as a "zero release" system under normal conditions of transport and hypothetical accident conditions. The design provides double containment of the fuel assemblies in a dry environment. The internal pressure in the inner containment will not exceed 86 psig even under hypothetical accident conditions with no external mechanical cooling of the package and assuming all fuel rods release fission product gas to the inner containment environment. This pressure is easily contained. There are no active mechanical connections such as pressure relief valves between the inner containment and the environment. The inner containment has drain and vent penetrations which terminate at valves located in the secondary containment space between the cask inner and outer closure heads. Thus the inner containment functions exactly as a failed fuel can and no distinction need be made with regard to fuel cladding integrity prior to cask loading.

Decay heat is removed from the fuel to the cask by thermal radiation and conduction through an aluminum fuel basket. Heat is then transferred through the cask sides and ends by a combination of conduction, natural convection in the water filled neutron shield, and natural convection and radiation from the surfaces of the cask. Being entirely passive this means of heat dissipation is highly reliable. A self-contained redundant auxiliary cooling system, mounted on the rail car, is used to maintain cask and fuel temperatures as low as possible solely to facilitate cask cooldown and unloading operations.

The criticality analysis was based on fresh fuel assemblies with zero burnup and shows that the NLI 10/24 rail cask meets all the requirements of a Fissile Class III package in accordance with the requirements of 10 CFR 71.

The NLI 10/24 cask has been designed to meet all applicable requirements of 10 CFR, Chapter 1, Part 71 and 49 CFR, Chapter 1, Parts 170-189. The design function was carried out under N L Industries, Inc. Quality Assurance Program for Design and Engineering which is comparable to 10 CFR, Chapter 1, Part 50, Appendix B and ASME, Section III, Design Control. Manufacturing and Quality Assurance Program will be carried out in accordance with N L Industries, Inc. Commercial Nuclear Quality Control Manual as applicable by design requirements.

For the purpose of package evaluation, the shipping cask mounted on the rail car with impact structures attached is to be considered as the package presented for shipment. The auxiliary cooling system is not considered as part of the package since the safety and integrity of the package is not dependent upon the operation of the cooling system.

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Section II

SECTION II

DESIGN SUMMARY

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SECTION II

DESIGN SUMMARY

1.0 Design Philosophy

The basic philosophy followed in the development of the NLI 10/24 cask design was to approach as near as possible to a "zero release" system. Moreover, it was decided that this system should be passive in nature and not be dependent upon post accident operation of active components to achieve containment. For this reason, dry shipment with double containment was selected as the preferred mode for large rail shipments of spent fuel. With the containment filled with helium, the cavity pressure under normal conditions of transport is 16.45psig. If the assumption is made that all the fuel pins rupture and there is a total release of fission gas product to the containment vessel environment, the maximum transport pressure would be 72.9 psig. Any increase in cask cavity temperature due to hypothetical accident conditions will be accompanied by a very small increase in cavity pressure. Maximum possible cavity pressure is only 85.8 psig which occurs at the end of the fire accident only if all the BWR fuel pins rupture. With a wet shipment, normal operating pressure would be on the order of 400 to 500 psig and loss of coolant would occur due to loss of auxiliary cooling and neutron shield

water as a result of the hypothetical accident sequence. The NLI 10/24 cask dry concept does not require a safety valve. The inner and outer containments have a maximum allowable working pressure several times greater than the maximum possible internal pressure of 85.6 psig. No release of the helium filler gas is predicted under normal conditions of transport or hypothetical accident conditions. The filler gas is considered to be part of the package contents.

2.0 Cask System Description

The NLI 10/24 shipping cask has been designed to provide double containment of either intact or failed fuel assemblies. The package as presented for shipment consists of the cask body, fuel basket, cask cavity closure head (inner closure head), the cask body closure head (outer closure head), and impact structures attached to each end of the cask. (See NLI Drawing 70650). The cask body consists of inner and outer stainless steel shells which are joined by stainless steel forgings at each end to make a continuous weldment. The annulus between the inner and outer cylinders contain a lead gamma shield. Neutron shielding is provided by a water jacket which surrounds the outer stainless steel cylinder and axially blankets the active fuel region of the fuel assembly. The internal cask cavity, which is 45.0 inches in diameter, has been sized to accommodate the removable aluminum fuel baskets. Separate baskets are designed to carry either 10 PWR type or 24 BWR type fuel assemblies. The closure head end or open end of the cask cavity is machined from a stainless steel forging. The length of the forging is such that it forms part of the cask cavity. This design provides the necessary material from which two closure head flange seating surfaces can be machined. The design also provides the opportunity to have the cask body penetration pass through the solid stainless steel forging, eliminating the need for

a penetration through the shielding materials and the inner and outer shells compromising the integrity of the cask structure. The head end forging also provides maximum protection of the inner and outer closure heads and closure head seals since the closure head is located within the outer dimensions of the forging. The cask lifting trunnions are located in the head end forging eliminating the concern of a puncture of the shielding containment shell by an impact on a trunnion.

The neutron shield water jacket is constructed of 3/4 inch stainless steel. At each end of the 3/4 inch shell is a short reinforced section of stainless steel which provides an adequate transition section from the 3/4 inch shell material to the upper and lower forgings. The water jacket design was developed to provide structural integrity and resistance to penetration under normal conditions of transport as well as to contribute to the overall strength of the cask in the hypothetical accident conditions. Expansion tanks are provided in order to maintain a solid water shield. The expansion tanks are located on the rail car and are connected to the water jacket by a flexible metal hose. The water jacket is equipped with a fill and drain valve and a relief valve and a plugged penetration at the top end which is used to vent the jacket during initial filling. The relief valve is set to relieve at a pressure of 220 psig.

There is one penetration through the head end forging of the cask body. This penetration exits into the cavity area between the inner and outer closure heads and is outside the primary containment seal area. This penetration is used to drain the area between the inner and outer closure heads as the cask is removed from the spent storage pool. Prior to shipment of the loaded cask, the penetration is used to pressure test the secondary containment system. The penetration terminates in a threaded connection located within a recess in the head end forging. A bolted and gasketed closure plate isolates the penetration from the ambient. The cask body closure head (outer closure head) is a flat stainless steel plate 2-1/2 inches thick. There are no penetrations through the outer closure head since the single necessary service penetration is made through the cask body end forging. The outer closure head, which is the secondary containment boundary, also serves as a rugged valve box cover for the two service penetrations in the inner closure head cask flange. Viton or silicone may be used for the outer closure head seal.

The cask cavity closure head (inner closure head) consists of a stainless steel forging, the center section of which is filled with uranium which is covered by a stainless steel plate welded to the forging. The bulk of the required gamma shielding is contained in the inner head to provide operator protection during the handling operations involving a loaded cask. The closure head is held in

place by 16 high strength bolts. The inner closure head seal is a metallic "O" ring. The seal is more than adequate to withstand the temperatures expected as a result of the hypothetical fire accident. There are two penetrations in the inner closure head cask flange which are used for venting and draining the cask cavity. These penetrations are equipped with 1-1/2 valved quick disconnect fittings. These fittings are housed in a valve box which is bolted to the cask flange. The metal "O" ring seals in the valve box arrangement provide the pressure boundary for the primary containment system.

Separate fuel baskets are provided for the 10 PWR and 24 BWR cask loading. Each fuel basket is constructed from aluminum castings 45 inches in diameter. The basket covers the active length of the PWR and BWR fuel. Axial channels are located in the casting to accept individual fuel assemblies. Criticality control is achieved by means of a box structure which lines the fuel channel openings in the aluminum basket. The box is constructed of plate material consisting of Ag-In-Cd in a 304 stainless steel sandwich. The box structures are supported at both ends by means of a stainless steel plate to form a rigid structure independent of the aluminum basket.

The aluminum basket serves as a heat conduction path from each individual fuel assembly to the cask cavity wall. Decay

heat is removed from the cavity wall by means of cooling water circulated through channels welded to the outside surface of the inner shell. When the cooling system is not operating, heat dissipation from the inner shell occurs by conduction through the lead and outer shell, natural convection in the shield tank and convection and radiation to the environment from the finned outer surface of the water shield jacket.

Impact limiting structures are integrated into the tie-down structure at each end of the cask. The impact limiter system is a cylinder of balsa wood encased between aluminum shells. The impact limiter is mojnted on the ends of the cask and provide protection for angles of impact. The impact limiters are designed such that the kinetic energy of the system upon impact is spent in the progressive crushing of the impact structure yet at the end of the failure mode, there has been no contact made between any portion of the cask body and the impact surface.

The NLI 10/24 Spent Fuel Shipping Cask has a fully loaded weight of 194,000 pounds. The transport system consists of a specially designed rail car with the cask tie-down arrangement built into the car frame. The rail car can be equipped with either a single or redundant cask cooling system which can be used to maintain lower in transit cask and fuel temperatures to facilitate cask cooldown and unloading operations. The rail car mounted cooling system is a closed circulating water loop consisting

of a pump and a water to air heat exchanger in series with cooling channels located on the cask inner shell. The system is normally run on forced convection with power supplied to the pump and heat exchanger fan by means of a diesel generator. In the case of a redundant system loss of power or loss of an active component, automatic transfer to the redundant system takes place. Should power be lost in both systems, decay heat removal will be accomplished by boiling natural convection in both systems. In the ambient condition of still air at 130°F, the average coolant temperature is 181°F with forced convection and 250°F in the boiling natural convection mode.

Personnel barriers completely cover both cooling system packages as well as the cask under normal transport conditions. The personnel barrier is basically an aluminum cage made from expanded aluminum sheet on an aluminum frame. A removable panel at the center of the car provides access to the connection between the water jacket and expansion tanks. A specially designed lift rig is provided for cask handling which engages the trunnions on the cask body.

The safety analysis report also covers an alternate configuration of the rail cask. The alternate configuration as shown on Dr. 70708F affects the bottom end of the cask only. The features affected are as follows:

1) The ring structure located in the void space, outboard of the uranium shielding is deleted.
2) Auxiliary cooling system headers in the bottom forging have been deleted.
3) The twenty cooling channels which run axially down the shell are now inter connected on the shell by a return bend between pairs of channels.
4) The weld joint which joins the outer ring forging to the outer bottom head is located in board from the original location.

The alternate configuration does not change the essential features on which structural integrity, shielding, criticality or thermal performance are based.

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NLI 10/24 SPENT FUEL SHIPPING CASK DATA

Capacity	10 PWR Type Fuel Assembly	or	24 BWR Type Fuel Assembly
Fuel Data	10 PWR		24 BWR
Envelope, inches	8.60 sq. x 171.5		5.44 sq. x 176.25
Enrichment w/o U-235	3.50		2.80
Weight of Uranium, kg	4,750		4,800
Average burnup, MWD/MTU	35,500		29,700
Assembly average specific power, kw/kg	36.3		25
Total Fixed Neutron Source, n/sec	48.5×10^8		46.6×10^8
Total Gamma Source, mev/sec	27.9×10^{16}		25.1×10^{16}
Total Decay Heat, kw	70		70
Cask weight, loaded	Calculated Design weight		194,000 lbs. 200,000 lbs.
Cask Assembly Envelope Dimensions	96 in diameter 204.5" long		
Internal Cavity Dimensions	45.0" in diameter 179.5" long		
*Maximum normal operating pressures			
Cask cavity (primary containment)	72.9 psig		
Water Jacket	153 psig*		

*Water jacket relief is set for 220 psig.

* Maximum normal operating pressures are based on operation without auxiliary cooling and total fission gas release.

3.0 Thermal Analysis

The following cask operating conditions were analyzed using a two dimensional model of the cask axial cross section:

- Normal Conditions without Auxiliary Cooling. Under these conditions a rupture of both cooling systems is assumed. Decay heat is transferred through the cask gamma shield and the neutron shield water jacket, and is dissipated by means of natural convection and radiation from the finned exterior surface of the package.
- Hypothetical Accident Conditions. The hypothetical accident conditions are assumed to result in loss of all auxiliary coolant and loss of shield water. This is the worst thermal condition (post fire steady state) since decay heat must be transferred through the empty neutron shield. The increased temperature drop across the empty shield tank causes higher cask and fuel operating temperatures.

A detailed description of the thermal analyses performed is provided in Section VIII. Results of these analyses are summarized in Table II-1. The design heat load used in the thermal analysis is 70 kw.

TABLE II-1

CASK TEMPERATURE DISTRIBUTIONS

FUEL MIDPLANE

	Normal Conditions of Transport Without Cooling		Post Fire Steady State	
Decay Heat = 70				
Ambient	130°F		130°F	
Outer Surface	323		327	(1230**)
Outer Shell	344		490	
Outer Lead	359		514	
Inner Lead	420		566	
Duct Fluid	420	(dry)	556	(dry)
Inner Shell	420		561	
Outer Seal	180		270	(1190**)
Inner Seal	270		350	

** Maximum temperature, occurs at end of fire

The analysis for 70 kw is shown in appendix E of Section VIII. This represents a 2-D axial analysis including the top and bottom sections of the cask. Calculated temperatures are shown in Table II-1 for the cask body at the cask midplane.

The temperatures obtained during actual operation will be lower than the temperatures shown in Table II-1 since the cooling system will be operating. If power is lost in both cooling systems, decay heat will be removed by natural convection boiling heat transfer in the coolant channels with the steam at a temperature of about 250°F.

Under normal conditions of transport without auxiliary cooling, no heat is removed from the cooling channels. Decay heat is dissipated to the ambient statically by conduction, radiation and convection. The package surface temperature reaches a maximum of 323°F under these conditions with the neutron shield water and ethylene glycol

reaching a maximum average temperature of 341°F with a corresponding saturation pressure of 85.3 psig. The design pressure of the neutron shield tank is 235 psig.

Under hypothetical accident conditions all coolant and shield water is lost. The increased ΔT across the neutron shield results in the hottest cask temperatures. Any fission gas released from the fuel pins will be trapped in the cask cavity. Since there is no blowdown of cavity fluid and the closure head seals are not affected structurally by the accident sequence, there is no predicted release of activity to the ambient as a result of the hypothetical accident conditions of 10 CFR 71.

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4.0 Criticality Analysis

The NLI 10/24 cask meets all requirements of a fissile Class III package. The fresh fuel assumption was used in performing the required analyses. A preliminary analysis was used to determine the most reactive pitch of the individual fuel assemblies assuming water filled cavity conditions. This most reactive pitch was used in performing the criticality analysis for the array. Results are summarized in Table II-2. A detailed description of the analysis is provided in Section X.

TABLE II-2

CRITICALITY ANALYSIS RESULTS

Fuel Type	Number of Elements	Weight Percent U-235 in Fuel	K_{eff}
PWR	10	3.50	0.946 ± 0.005
BWR	24	2.80	0.953 ± 0.005

5.0 Structural Analysis

A detailed structural analysis of the NLI 10/24 rail cask is presented in Section XI. This analysis shows that the NLI 10/24 cask is designed to withstand all the loads associated with the normal conditions of transport with a minimum factor of safety of 1.5 based on yield strength of the material. Under the dynamic loading of the hypothetical accident sequence, stresses in the inner and outer pressure boundary of the cask are below the allowable dynamic strength of the material. Thus, all pressure boundary materials retain their integrity and there is no breach of the system. Integrity of the double containment system is maintained.

The impact limiting structures on the cask ends are sacrificial members which are designed to absorb the energy of the free drop tests by plastic deformation. The shock absorber system prevents direct impact of any part of the cask body including the neutron shield tank. For the purposes of package evaluation, the cask with impact limiters attached is to be considered as the package presented for shipment. Loss of shield water is assumed. This has been considered in the thermal analyses. Peak acceleration loads for the free drop are as follows:

Side Impact	80.75 g
Corner Impact (45°)	30.95 g
Top & Bottom Impact	29.4 g

The fuel basket and associated internal structures are designed to maintain basket integrity and fuel bundle spacing under impact conditions.

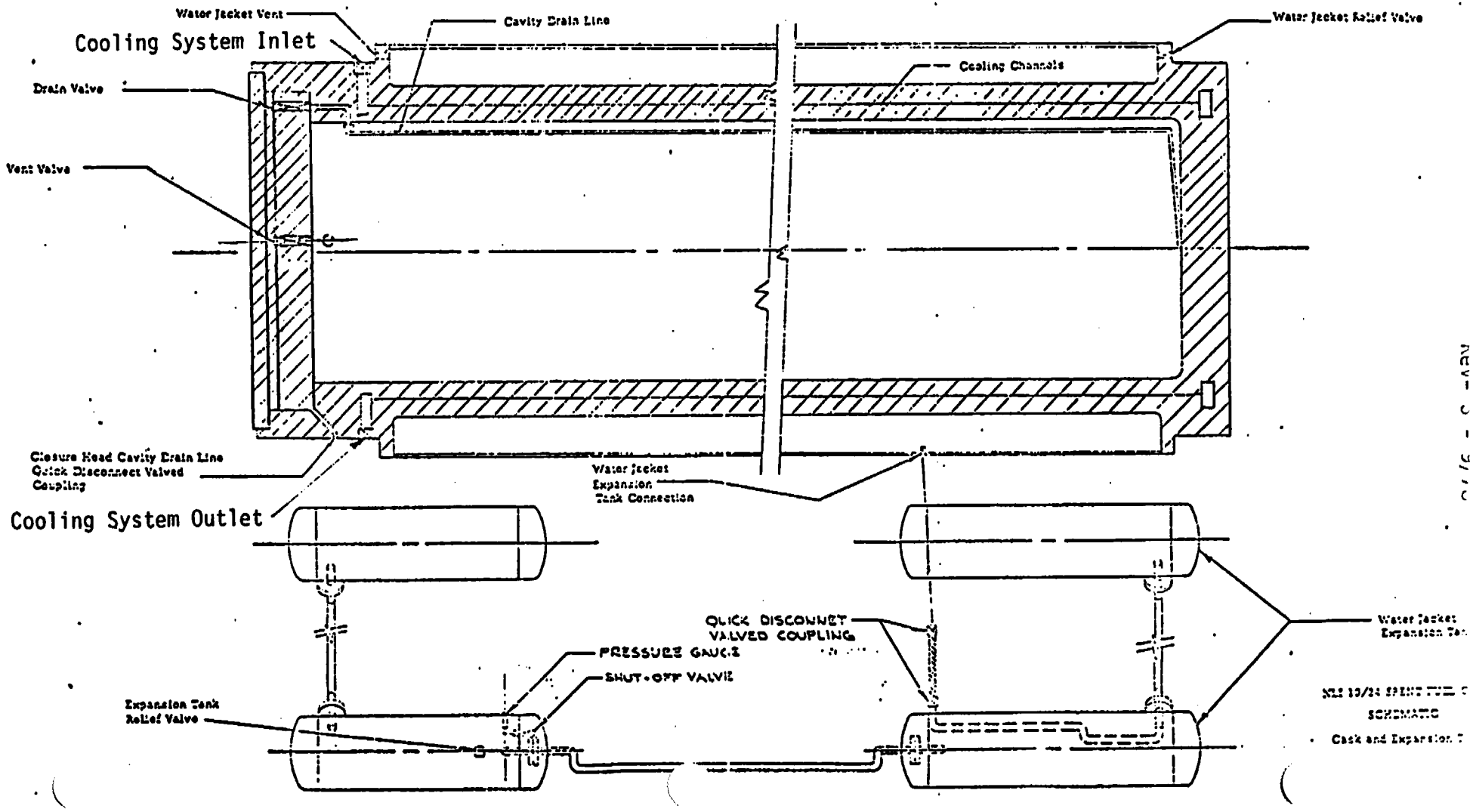
6.0 Shielding Analysis

The neutron and gamma shield has been designed to limit the combined dose rate to less than 10 mrem/hour at a point six feet off the cask personnel barrier. Under hypothetical accident conditions, loss of all shield water is assumed to occur.

The accident dose rates at a point 3 feet off the surface of the cask are:

	mrem/hour
Gamma	25
Neutron	625
Total	650

The combined accident dose rates are less than the allowable value of 1000 mrem/hour. Please refer to Section IX for a detailed discussion of the shielding design.


Water Jacket Vent
Cooling System Inlet
Cavity Drain Line
Water Jacket Relief Valve
Cooling Channels
Drain Valve
Vent Valve
Closure Head Cavity Drain Line
Quick Disconnect Valved
Coupling
Cooling System Outlet
Water Jacket
Expansion
Tank Connection
QUICK DISCONNECT
VALVED COUPLING
PRESSURE GAUGE
SHUT-OFF VALVE
Expansion Tank
Relief Valve
SCHEMATIC

Section III

SECTION III

FUEL DESCRIPTION AND SOURCE DATA

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Section III

FUEL DESCRIPTION AND SOURCE DATA

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PREFACE

Section III establishes the design basis fuel parameters and calculates the resulting neutron, gamma and decay heat source values. Subsequent determinations required the decay heat load to be limited to 70 kw/hr. This reduction in decay heat load is not accommodate by a reduction in operating values of fuel exposure, specific power and initial enrichments established in this section but rather by an increased in fuel cool-down time prior to shipment.

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Section III

FUEL DESCRIPTION AND SOURCE DATA

1.0 INTRODUCTION AND SUMMARY

The NLI 10/24 spent fuel cask is a dual purpose shipping container designed to transport ten PWR or twenty four BWR spent fuel assemblies. The cask is capable of accommodating the fuel designs of General Electric, Westinghouse, Babcock and Wilcox, and Combustion Engineering (typical descriptions in Table III - 1). The reference design cask loadings and the corresponding sources for the NLI 10/24 cask are given in Table III-2. The fuel heat and radiation source terms which were derived from consideration of operating values of fuel exposure, specific power, initial enrichment and axial peaking factor are described in the following paragraphs. The design basis PWR fuel will have been cooled for 150 days following discharge from the reactor and prior to loading in the cask. The design basis cooling time for BWR fuel is 120 days following discharge. The mechanical design parameters of BWR and PWR fuel shown in Table III-1 were developed from a review of PSAR and FSAR data and are considered representative of the designs which will be shipped in the NLI 10/24 cask.

2.0 NEUTRON SOURCE

2.1 Source Per Assembly

With regard to operating parameters the generation of neutrons in spent fuel from spontaneous fission and α, n reactions in oxygen is primarily dependent on burnup and only weakly dependent on specific power. The specific fixed neutron source (i.e., neutrons/sec/gram uranium without subcritical multiplication) as a function of burnup is

TABLE III-1

FUEL MECHANICAL DESIGN PARAMETERS

Reactor Type	Westinghouse PWR	Combustion Engineering PWR	Babcock & Wilcox PWR	General Electric BWR
Pellet Dia., in.	0.3649	0.3795	0.370	0.487
Pin Array	15 x 15	14 x 14	15 x 15	7 x 7
Pin Pitch, in.	0.563	0.580	0.568	0.738
Pin Dia., in.	0.422	0.440	0.430	0.563
KgU/Assembly	448	433	454	197
Clad Material	Zr-4	Zr-4	Zr-4	Zr-2
Clad Thickness, in.	0.0243	0.026	0.0265	0.032
Fuel Pin/Fuel Assy.	204	176	208	49
Overall Fuel Assy. Cross Section, in.	8.426 x 8.426	8.13 x 8.13	8.536 x 8.536	5.438 x 5.438
Active Fuel Length, in.	144	137	144	144
Overall Shipping Length, in.	159.8	157.2	165.6	175.9
Enrichment w/o U-235	3.35	2.90	3.09	2.65
Specific Power kw/KgU	38.5	35.7	39.8	22.6
Average Burnup MWD/MTU	33,000	32,000	32,400	30,000
UO_2 Density, % TD	93	93	91	95

TABLE III-2

REFERENCE DESIGN CASK LOADINGS AND SOURCES

	10 PWR	24 BWR
Total Uranium Loading, KgU	4,540	4,728
Initial Enrichment, w/o U-235	3.35	2.65
Average Burnup, MWD/MTU	35,500	29,700
Average Specific Power, Kw/KgU	36.3	25
Total Fixed Neutron Source, n/sec	48.5×10^8	46.6×10^8
Axial Maximum to Average Neutron Source	1.25	1.38
Total Gamma Source, MeV/sec	27.9×10^{16}	25.1×10^{16}
Total Decay Heat, Kw	97.2 *	88.8 *
Axial Maximum to Average Decay Heat	1.2	1.2

* Used for certain conditions, balance of cases are for a 70kw thermal load.

shown in Figure III-1. These curves are based on LEOPARD-FLYASH calculations as described in Reference 1. In each case they are for a constant specific power(38 kw/Kg PWR, 27 kw/Kg BWR) with various continuous operating times to give the desired burnup. Other calculations have shown that a 50% reduction in specific power would yield only a 5% increase in neutron source.

Because the burnup of a given fuel assembly is not uniform and because the specific neutron source does not increase linearly with burnup, the total neutron source must be obtained from a summation of the source from various portions of the fuel assembly. This will lead to a source different from that obtained from Figure III-1 at the average assembly burnup.

The most significant spatial variation of burnup is the axial distribution. Typical axial burnup distributions are shown in Figure III-2.[2, 3] The total assembly fixed neutron sources using these axial distributions and the specific neutron sources from Figure III-1 are given versus assembly average burnup in Figure III-3.

The rapid increase in specific neutron source with burnup leads to an axial neutron source distribution which has a larger maximum-to-average ratio than the burnup distribution. The maximum-to-average neutron source ratio based on the burnup distributions in Figure III-2 are given in Table III-2.

2.2 Total Cask Source

In order to determine a total neutron source for the cask available information on the actual expected distribition of average assembly burnup in discharge fuel was reviewed. This review led to a typical cask loading

FIGURE III-1

TRANSURANIUM ELEMENT SPECIFIC NEUTRON SOURCE PRODUCTION AS A FUNCTION OF FUEL BURNUP

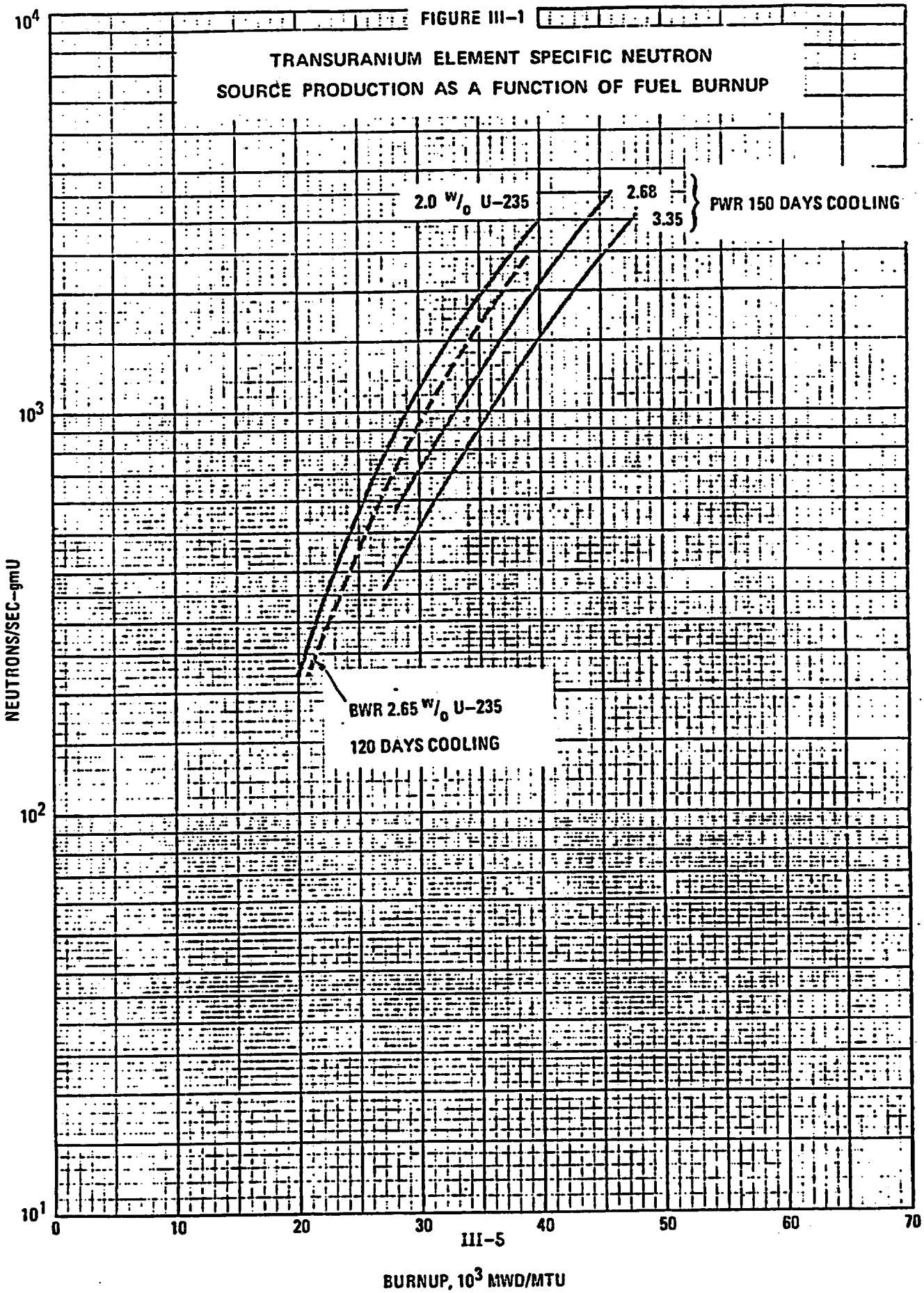
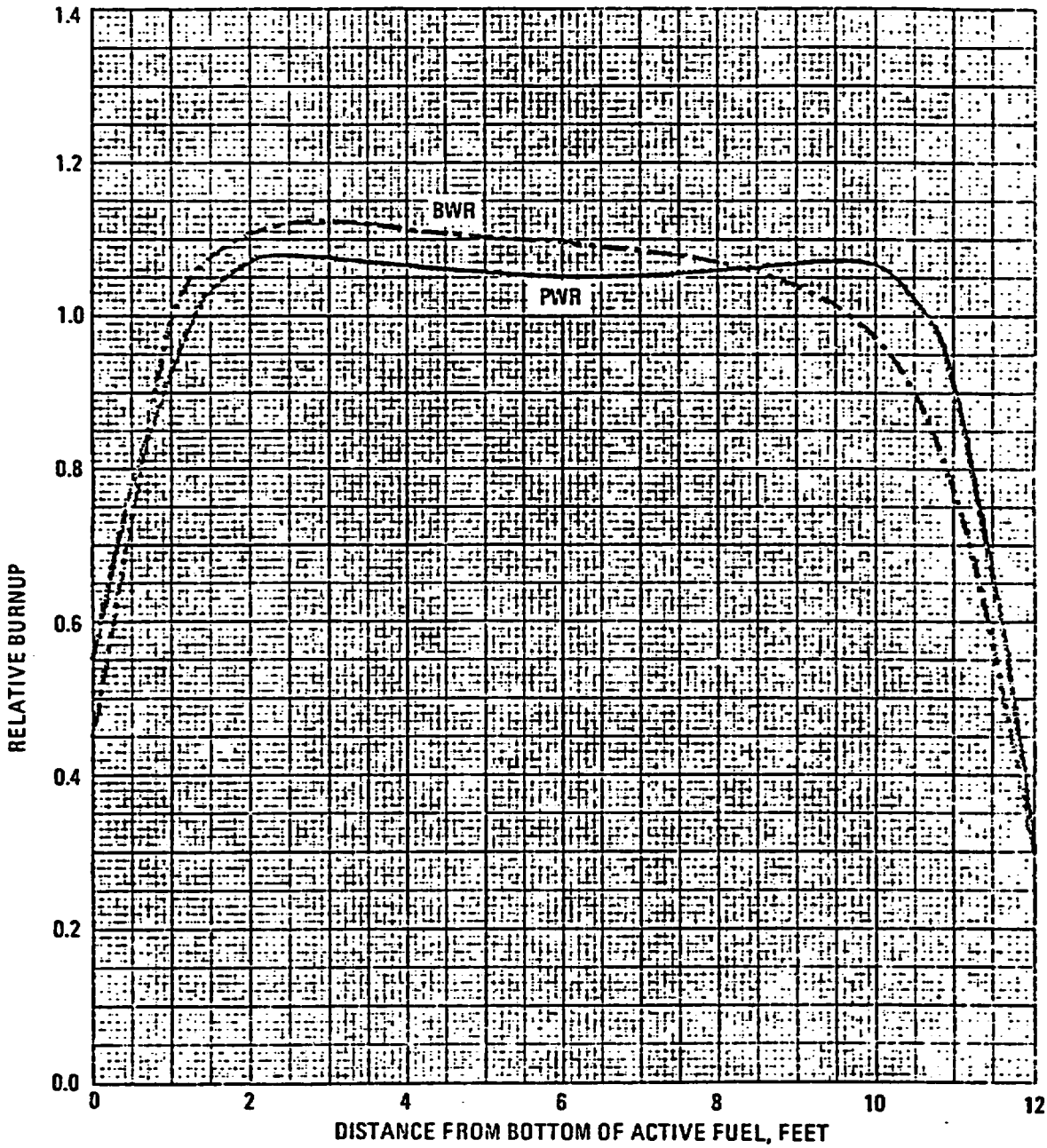


FIGURE III–2

AXIAL BURNUP DISTRIBUTION



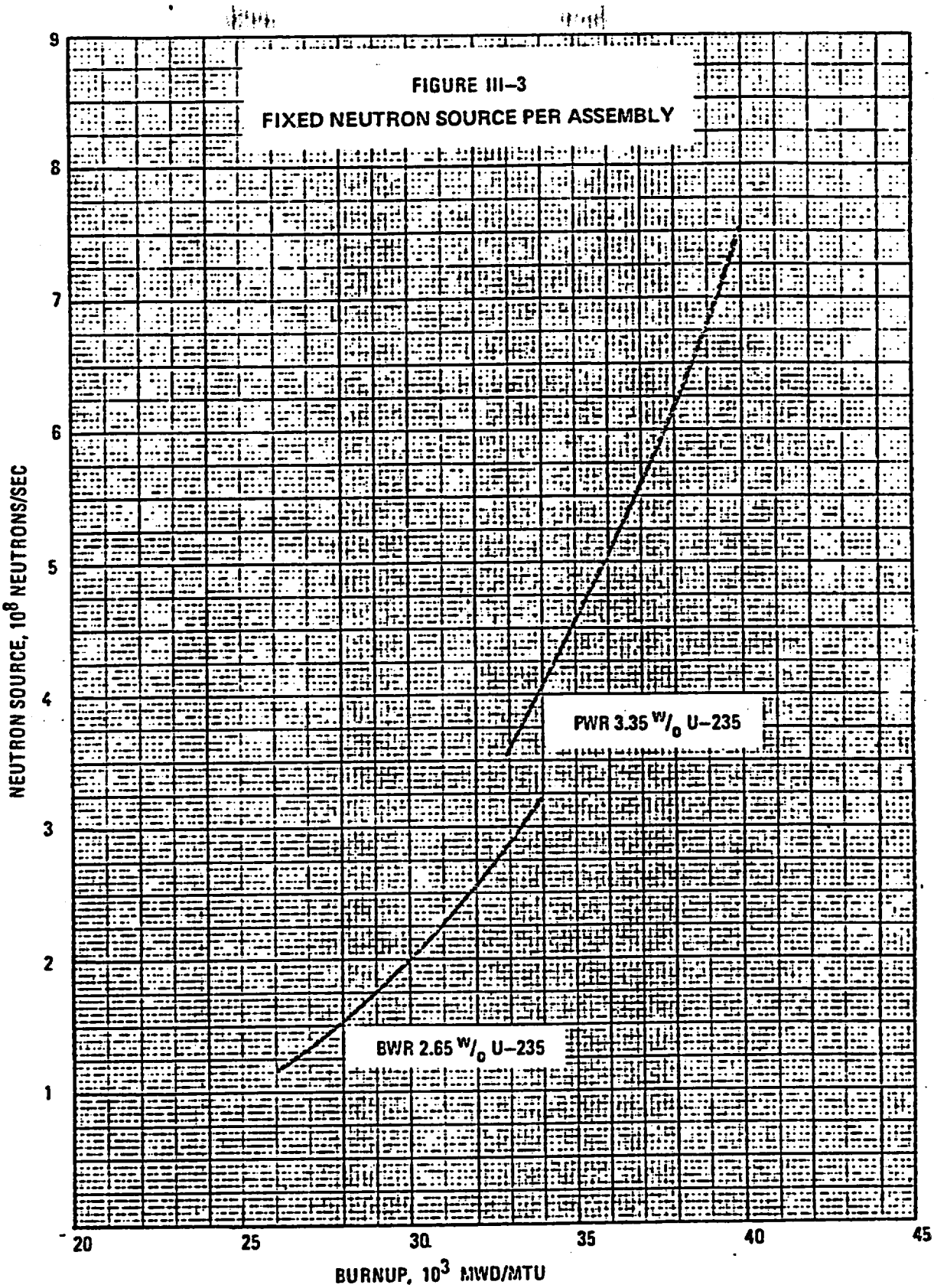


FIGURE III-3
FIXED NEUTRON SOURCE PER ASSEMBLY
PWR 3.35 w/o U-235
BWR 2.65 w/o U-235
NEUTRON SOURCE, 10^8 NEUTRONS/SEC
BURNUP, 10^3 MWD/MTU
1
2
3
4
5
6
7
8
9
20
25
30
35
40
45

for each fuel type (PWR and BWR). The source for each assembly was then obtained from Figure III-3 and summed to give the total source in Table III-2. The total source is seen to be only slightly larger than that obtained for the average burnup in the cask.

2.3 Energy Spectrum

The energy spectra for neutrons coming from spontaneous fission and from (α, n) reactions are given in Reference 4 for both Cm-242 and Cm-244. For both isotopes the spectra were weighted by the relative contributions of spontaneous fission and (α, n) and put into the energy structure used in the shielding studies. The resulting combined spectra are shown in Table III-3. For a given fuel condition the spectra from the separate isotopes Cm-242 and Cm-244 can be combined based on the fraction of neutrons emitted from the two isotopes. For the PWR fuel at 3.35 w/o and 35,000 MWD/MTU, FLYASH-II results gave 52.3% of the neutrons emitted by Cm-242 and 47.7 % by Cm-244. Using these weighting factors the combined spectrum for Curium neutrons is shown in Table III-3.

3.0 DECAY HEAT

As described in more detail in Reference 1, the fission product decay heat was calculated using the method given in the American Nuclear Society Proposed Standard, "Energy Release Following Shutdown of Uranium-Fueled Thermal Reactors".[5]

For the conditions of interest for spent fuel shipment (and for a given cooling time) the fission product decay heat is primarily dependent on specific power during operation with only a secondary dependence on

TABLE III-3

NEUTRON SPECTRUM FROM CURIUM

Group	Energy, Mev	Cm-242	Cm-244	Combined Cm Spectrum (E=3.35%) 35,000 MWD/MTU
1	15.0 - 12.2	--	--	--
2	12.2 - 10.0	0.0004	0.0008	0.0006
3	10.0 - 8.18	0.0020	0.0040	0.0030
4	8.18 - 6.36	0.0073	0.0138	0.0104
5	6.36 - 4.96	0.0229	0.0454	0.0336
6	4.96 - 4.06	0.0472	0.0473	0.0472
7	4.06 - 3.01	0.2345	0.1163	0.1781
8	3.01 - 2.46	0.2068	0.1072	0.1592
9	2.46 - 2.35	--	--	--
10	2.35 - 1.83	0.1702	0.1432	0.1573
11	1.83 - 1.11	0.1309	0.1887	0.1585
12	1.11 - 0.55	0.1173	0.2188	0.1657
13	0.55 - 0	0.0606	0.1145	0.0864

burnup (or operating time). For a cooling time of 150 days increasing the operating time from 1000 to 1200 days gives only a 5% increase in fission product decay heat for the same specific power during operation. Moreover, the decay heat is more than that during the earlier stages of operation. For example, for constant specific power the last 1/3 of operation contributes 65% of the decay heat, the middle 1/3 of operation contributes 22% of the decay heat while the first 1/3 contributes 13% of the decay heat.

Based on a review of typical operating specific powers, it was concluded for a 10 PWR loading that the fission product decay heat could be adequately based on average specific power of 36.3 kw/KgU. For 1000 days of operation and 150 days cooling this yields a fission product decay heat of 8.94 Kw per PWR assembly.

In addition to the fission product decay heat, heat is generated from the decay of the transuranium nuclides produced from non-fission capture in the fuel. This has been calculated to be 0.78 Kw for a 40,000 MWD/MTU burnup assembly.[1] The total decay heat load is therefore 97.2 Kw for the 10 PWR cask loading. *

For BWR fuel it was concluded that the decay heat load could be based on a specific power of 25 Kw/KgU and an operating time of 1260 days. For 120 days cooling this yields a fission product decay heat of 3.35 Kw per assembly and a transuranium decay heat of 0.35 Kw per assembly. The total decay heat is therefore 3.70 Kw per assembly and 88.8 Kw for a cask loading of 24 BWR assemblies. *

The operating power distribution in the reactor leads to an axial decay heat distribution. At cooling times of interest for spent fuel shipment the axial distribution of decay heat is better represented by the average

* Decay heat load has been reduced to 70kw.

axial power distribution over the last several months of operation than by the reactor core design axial power distribution. The decay heat distribution is therefore considerably less peaked than the core design distribution, and will approach the end-of-life burnup distribution. An axial maximum-to-average decay heat peaking factor of 1.2 was therefore selected for use in the thermal design of the cask.

4.0 GAMMA SOURCE

The gamma source for shielding purposes can be adequately taken as 50% of the fission product decay heat. Using the previously defined fission product decay heats the 50% assumption yields a 4.47 Kw gamma source per PWR assembly and 1.68 Kw gamma source per BWR assembly. Converting units the total gamma source strengths are 27.9×10^{16} MeV/sec for the 10 PWR loading and 25.07×10^{16} MeV/sec for the 24 BWR loading.

For shielding calculations the total gamma source defined above is broken down into discrete energy groups. The Shure data in Reference 6 was used to obtain the relative gamma energy distribution at 150 days after shutdown as shown in Table III-4. The previously calculated total gamma source was then assigned to the different energy groups according to this distribution. The resulting energy dependent source distribution is given in Table III-4. As indicated in Section IX this energy distribution was further subdivided to be consistent with the energy distribution used in the shielding calculation. As can be seen from the table, almost 99% of the gamma ray energy is emitted below 1 MeV.

TABLE III-4

REFERENCE DESIGN 10 PWR LOADING
GAMMA ENERGY DISTRIBUTION

Shure[6] Energy Group MeV	Relative Distribution 150 Days After Shutdown	Gamma Decay Source* MeV/sec
0.1 - 0.4	1.2×10^{-2}	3.35×10^{15}
0.4 - 0.9	0.97526	27.21×10^{16}
0.9 - 1.35	1.8×10^{-3}	5.02×10^{14}
1.35 - 1.8	9.4×10^{-4}	2.62×10^{14}
1.8 - 2.2	1.0×10^{-2}	2.79×10^{15}
> 2.2	$< 2 \times 10^{-4}$	--
	Total	27.9×10^{16}

*Source for 10 PWR assemblies each with 36.3 kw/kg specific power, 150 days cooling time, and 0.454 MTU.

5.0 REFERENCES

1. "NLI 1/2 LWT Cask Safety Analysis Report", Amendment 1, Docket No. 70-1318, December 1972.

2. North Anna Units 3 & 4, VEPCO, DKT 50-404.

3. "Plutonium Utilization in Boiling Water Reactors, Phase II Semi-Annual Report, July-December 1970", NEDC-12198, May 1971.

4. Rinshaw, S. J., and Ketchen, E.E., "Curium Data Sheets", ORNL-4357, January 1969.

5. ANS Standard ANS-5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors (DRAFT)," American Nuclear Society, Hinsdale, Illinois, October 1971.

6. Shure, K., "Fission Product Decay Energy", WAPD-BT-24, December 1961.

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Section IV

SECTION IV

COMPLIANCE WITH CFR, TITLE 10, CHAPTER 1, PART 71

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SECTION IV

COMPLIANCE WITH 10 CFR 71, CHAPTER 1

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.22	Package Description	
71.22 (a) (1)	Gross Weight	The maximum gross weight of the cask is 200,000 lbs.
71.22 (a) (2)	Model Number	The model number of the cask is NLI-10/24.
71.22 (a) (3)	Specific materials, weights, dimensions, and fabrication methods.	The weights of the cask components are given in Section VII, "Weight Calculations". The materials of construction are given in Section VI, "Material Specifications". Fabrication methods are described in Section XIII, "Manufacturing and Quality Control".
71.22 (a)(3)(i)	Receptacles, identifying the one which is considered to be the containment vessel.	The cask structure consists of inner and outer stainless steel cylinders which are joined by stainless steel forgings at each end. The annulus between the inner and outer cylinders contains a lead gamma shield. Neutron shielding is provided by a water jacket which surrounds the outer stainless steel cylinder and all of the region containing active fuel. The fuel is carried in the cask cavity which has its own closure head, and is fitted with an aluminum basket which acts as a heat conductor. The cask cavity and its closure head is considered the primary containment vessel. The cask body is also fitted with an outer closure head. The volume between the inner and outer closure heads is considered as the secondary containment vessel.

Section of Part	Requirement or Subject of Provision	Assessment of Compliance
71.22(a) (3) (ii)	Non-fissile Neutron Absorbers or moderators.	Criticality control is achieved by means of a box structure which lines each fuel slot opening in the aluminum fuel basket and surrounds each fuel assembly. The box is constructed of plate material consisting of Ag-In-Cd absorber material in a 304 SS sandwich. The box structures are supported at both ends by means of a stainless steel plate to form a rigid structure independent of the aluminum basket. For further discussion see Section X "Criticality Analysis".
71.22 (a) (3) (iii)	Internal and external structures supporting or protecting receptacles.	The fuel is supported radially within the primary containment by means of the neutron absorber structure which is backed up by a solid aluminum basket. The basket is positioned axially within the cask by means of internal top and bottom support structures. The fuel is supported top and bottom axially by a separate support structure. The inner and outer closure heads are supported by and bolted to steps machined in the cask end forging. The ends of the cask body are fitted with impact structures consisting of a built up balsa wood disc encapsulated in aluminum shells.

Section of Part	Requirement or Subject of Provision	Assessment of Compliance
71.22 (a) (3) (iv)	Valves, sampling ports, lifting devices and tie down devices	There is one penetration through the body end forging to drain the volume between the inner and outer closure heads. There are two penetrations in the inner closure head flange to allow venting and draining of the cask cavity. These penetrations are between the primary and secondary containments. There are no penetrations between the primary containment and the ambient. The penetration thru the inner closure head flange are fitted with valved quick disconnect fittings. These fittings are housed in stainless steel valve boxes which are bolted to the cask flange. The metal "O" ring seals in the valve box arrangement provide the pressure boundary for the primary containment system. The single penetration through the body end forging terminates in a threaded connection located in a recess in the body end forging. A bolted and gasketed closure plate isolates the penetration from the ambient.

Section of Part	Requirement or Subject of Provision	Assessment of Compliance
71.22 (a) (3) (iv) Cont.		The neutron shield jacket and expansion tanks are fitted with NUPRO-#1/2-8CPAZ-150 relief valve. The valves are made of Type 316 stainless steel with VITON "O" Ring seals. Service temperature range is 100°F to 400°F. Water jacket is provided with a fill quick-disconnect valve type Snaptite EA. Series with VITON "O" Ring seals. Service temperature range is 100°F to 400°F.
		Lifting and positioning of the cask is accomplished with the use of a lifting yoke which engages a set of trunnions welded to the top forging of the cask body. Each trunnion is machined from a solid bar of stainless steel. The top forging of the cask body provides a more than adequate foundation for the trunnion.
		The tie-down of the cask to the rail car is accomplished by a set of saddle arrangements which support the cask at its ends. At each end of the cask is a large lug which is welded to the cask end forgings. These lugs mesh with sets of lugs which are welded to the car floor. With the cask in the horizontal shipping position and the lugs on the cask meshed with the lugs on the rail car, pins are fitted through the holes in the engaged lugs. This arrangement at the bottom end of the cask is designed to take the entire 10g longitudinal loading. The arrangements at the top end and the bottom end of the cask are designed to take their share of the 5g lateral and 2g vertical loadings.

Section of Part	Requirement or Subject of Provision	Assessment of Compliance
71.22 (a) (3) (iv) Cont.		This arrangement is capable of withstanding the specified acceleration components of 2 g vertical, 5 g transverse and 10 g in the direction of travel without generating stresses in excess of the yield stress in any material of the package.
71.22 (a) (3) (v)	Structural and mechanical means for the transfer and dissipation of heat.	Decay heat is removed from the fuel to the cask first by thermal radiation to the basket and then by conduction through the aluminum basket to the cask inner shell. Heat is then removed through the cask sides by a combination of conduction, natural convection in the water filled neutron shield, and natural convection and radiation from the surfaces of the cask. Thermal radiation augments the conduction heat transfer across well established gaps within the cask. Being entirely passive, this means of heat dissipation is highly reliable. A rail car mounted cooling system is provided to maintain cask and fuel temperatures as low as possible during transportation in order to facilitate cooldown and cask unloading operations at the fuel reprocessing plant.

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.22(a)(4)	Identification and volumes of any coolants and of receptacles containing coolant	Being a dry shipment with passive heat removal, there is no coolant associated with the package.
71.22(b)(1)	Identification and maximum radioactivity of radioactive constituents;	See Section III, "Fuel Description and Source Data."
71.22(b)(2)	Identification and maximum quantities of fissile constituents;	Section Section III, "Fuel Description and Source Data."
71.22(b)(3)	Chemical and physical form;	See Section III, "Fuel Description and Source Data."
71.22(b) (4)	Extent of reflection, the amount and identity of non-fissile neutron absorbers;	See Section Section X, "Criticality Analysis".
71.22(b)(5)	Maximum weight of contents	The maximum weight of fuel carried in the cask is 18,000 pounds.
71.22(b)(6)	Maximum amount of Decay Heat;	The maximum decay heat generated by fuel assemblies in the cask will not exceed 70 kw.
71.23	Package Evaluation	
71.23(a)	Demonstrate that the package satisfies the standards specified in Subpart C.	The cask satisfies the standards in Subpart C. See subpart "C" below.
71.23(b)	Fissile Class II	Not Applicable.

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.23(c)	Fissile Class III	There are no special controls or precautions to be exercised during transport. Operating procedures and check-off lists will be used to control the loading, unloading and handling of the cask. Accident control and recovery plans will be established in accordance with local and federal regulations.
71.24	Procedural Controls	Procedural Control adequate to satisfy the requirements of 71.51(b) have been prepared. See 71.51 below.
SUBPART C		
71.31	General Standards for all Packaging	
71.31(a)	Packaging shall be of such materials and construction that there will be no significant chemical, galvanic, or other reaction among the packaging components.	There will be no significant chemical, galvanic, or other reaction among the packaging components, or between the packaging components and the package contents. The fuel is carried in an aluminum basket which is housed in the stainless steel weldment. Reaction between local uranium shielding and stainless steel is prevented by a layer of flame-sprayed copper applied to those areas of the stainless steel shell where maximum temperatures are predicted.
71.31(b)	Packaging shall be equipped with a positive closure which will prevent inadvertent opening.	The double containment system utilizes two separate closure heads. The inner head is held in place by 16 stud bolts, 1 3/4" diameter. The outer closure head is held in place by 28 bolts, 1¼ inch diameter. Removal of the closure head requires deliberate action and the use of tools.

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.31(c)	Lifting Devices	
71.31(c)(1)		The cask lifting structure is capable of supporting 3 times the weight of the loaded cask without exceeding the yield strength of the materials involved. See Section XI, "Structural Analysis".
71.31(c)(2)		The closure head lifting structure is capable of supporting 3 times the weight of the closure head without exceeding the yield strength of the materials involved. See Section XI, "Structural Analysis".
71.31(c)(3)		The eyebolts used for closure head lifting and handling are removed prior to shipment. Hence, the only structure available for lifting the entire cask is the structure intended for that purpose.
71.31(c)(4)		The failure of any lifting device which is a part of the cask would not impair the containment or shielding properties of the cask.
71.31(d)	Tie-Down Devices	
71.31(d)(1)		When the acceleration defined by this section are applied to the cask the resulting stress in any material of the cask does not exceed the yield point of the material involved. See Section XI, "Structural Analysis".

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.31(d)(2)		There are no parts of the cask structure which can be used as tie-down devices, other than those specifically designed for that purpose.
71.31(d)(3)		The failure of any of the cask tie-down structures would not impair the ability of the cask to meet the other requirements of this subpart.
71.32	Structural Standards, Large Quantity Packaging	
71.32(a)	Load Resistance	Regarded as a simple beam, supported at its ends along the major axis, the cask can withstand a static load, normal to and uniformly distributed along its length, equal to 5 times its fully loaded weight without generating stress in any material of the cask in excess of its yield strength.
71.32(b)	External Pressure	The containment vessel can withstand an external pressure of 25 pounds per square inch without loss of contents. See Section XI, "Structural Analysis".
71.33	Criticality Standards	The cask remains sub-critical under all the conditions defined in this section. See Section X, "Criticality Analysis".
71.34	Evaluation of a Single Package	
71.34(a)(1)	Normal Transport	See 71.35 below.

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.34(a)(2)	Hypothetical Accident	Model testing was not done. The mechanical properties of the materials of construction of the cask and the shock absorber are well known. Analytical methods for predicting cask and shock absorber response to impact loads have been developed and correlated to actual data. Considering this, and the fact that conservative assumptions have been used throughout the analysis, mathematical analysis has been used to satisfy this subpart.
71.34 (b)		For the purpose of the hypothetical accidents, the cask has been analyzed independently of the vehicle.
71.34 (c)		The normal and accident conditions for which the cask has been analyzed are the same as those defined in 71.35 and 71.36.
71.35	Standards for normal conditions of transport for a single package.	
71.35 (a) (1)	There will be no release of radioactive material from the containment vessel.	Maximum integrity is achieved by double containment. Both the inner and outer closure heads maintain their integrity under the conditions specified in Appendix "A". See Section XI, "Structural Analysis".

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.35(a)(2)	Packaging Effectiveness	The effectiveness of the cask is not reduced by the conditions specified in Appendix "A". See Section XI, "Structural Analysis".
71.35(a)(3)	Explosive Mixtures	There is no mixture of gases or vapors in the package which could, through any credible increase in pressure or explosion, significantly reduce the effectiveness of the cask.
71.35(a)(4)	Coolant Contamination	There is no coolant associated with the package. The dry double containment system isolates the low pressure cavity filler gas from the environment under normal conditions of transport and hypothetical accident conditions. The coolant activity limits of this section do not apply.
71.35(a)(5)	Loss of Coolant	There will be no loss of the helium filler gas during a normal shipment. The dry double containment system provides maximum integrity at low operating pressure.
71.35(b)(1)	Criticality	The cask is sub-critical under the normal conditions of transport. See Section X, "Criticality Analysis".
71.35(b)(2)	Geometric Form	The geometric form of the cask contents are not altered under the normal conditions of transport.
71.35(b)(3)	No leakage of water into the containment vessel.	There can be no leakage of water into the cask during normal shipment. The cask is equipped with two separately sealed closure heads.

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.35(b)(4)(i)	Effective Volume	Under normal conditions of transport, there is no reduction in the volume of the cask or the containment vessel.
71.35(b)(4)(ii)	Effective Spacing	Under normal conditions of transport there is no reduction in the space between the center of the containment vessel and the outer surface of the cask.
71.35(b)(4)(iii)	Apertures	Under normal conditions of transport no apertures of any size can occur in the outer surface of the cask.
71.35(c)	Venting	The cask containment vessel does not vent to the atmosphere under the normal conditions of transport. See 71.35(a)(5) above.
71.36	Hypothetical Accident	
71.36(a)(1)	Radiation Dose Rate	Following the hypothetical accident condition the highest dose rate at point 3 feet from the external surface cask, would be 650 mrem/hr. See Section IX, "Shielding Analysis".
71.36(a)(2)	Radioactive Material Release	No radioactive material would be released from the package. The package is designed such that two individually sealed closure heads prevent loss of the low pressure filler gas trapped in the dry system. The impact structures on each end of the cask effectively form a protective envelope around the cask. There are no valves in the external surface of the cask which provide a leakage path directly into the primary containment system. All valves which penetrate the primary container are located on the inner head and are protected by a rugged outer stainless steel closure head.

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.36(b)	Criticality	The cask would remain sub-critical following the hypothetical accident sequence specified in Appendix "B" of this part. See Section X, "Criticality Analysis".
71.37	Evaluation of Fissile Package Array	
71.37(a)	Model Testing	Model testing has not been done to evaluate the cask by the criteria specified in 71.39. Damage to the package, following the hypothetical accident described in Appendix "B" of this part, has been evaluated by analytical methods. See response to 71.34(a)(2) above.
71.37(b)	Criticality Assumptions	The assumptions made in determining compliance with 71.39(a)(2) comply with this subpart. See Section X, "Criticality Analysis".
71.38	Standards for Fissile Class I	Not Applicable.
71.39	Standards for Fissile Class II	Not Applicable.
71.40	Standards For Fissile Class III	There will be only one cask per transport vehicle. The shipment remains sub-critical under the conditions of 71.40 (a) and (b). See Section X, "Criticality Analysis".
71.41	Previously Constructed Package	Not Applicable.

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.51	Procedures	
71.51(a)	Operating Procedures	See Section XVI, "Operating Procedures".
71.51(b)	Inspection Procedures	All loading, unloading and handling operations with the cask will be supervised by a qualified representative of the licensee who will ensure that the operating procedures are properly executed.
71.52	Unknown Properties	Conservative values have been assigned to variables; such as isotopic abundance, degree or irradiation, degree of moderation, etc., in all criticality calculations. The value of K_{eff} calculated are the maximum credible values. See Section X, "Criticality Analysis".
71.53	Preliminary Determinations	
71.53(a)	Defects	All materials of construction will be procured to ASTM, ASME, or AISI specifications. A thorough program of quality assurance including non-destructive testing will be conducted during fabrication to insure that there are no cracks, pinholes, uncontrolled voids, or other defects which could significantly reduce the effectiveness of this part. See Section XIII, "Manufacturing and Quality Control".
71.53(b)	Pressure Test	Prior to its first use the inner container and cask cavity will be given a helium leak test. See Section XIV, "Functional Test Procedures".

Section of Part	Requirement or Subject of Provision	Assessment of Compliance
71.53 (c)	Marking	The cask model number will be engraved in a stainless steel plate which will in turn be welded to the cask outer shell.
71.54	Routine Determination	
71.54 (a)	Damage	Prior to each use, a visual inspection will be made of all accessible surfaces of the cask. See Section XVI "Operating Procedures" and Section XVII, "Maintenance Program".
71.54 (b)	Neutron Absorbers	The neutron absorber is a permanent part of the fuel basket. The basket must be in place to support the fuel.
71.54 (c)	Closure	See 71.54 (a) above. In addition, a routine pressure test will be performed prior to each use.
71.54 (d)	Valves	The valves which penetrate the primary system are located on the inner closure head flange. The inner closure is protected by an outer closure head which is structurally adequate to survive the hypothetical impact condition. Unauthorized access to these valves would be extremely difficult if not impossible.

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.54(e)	Internal pressure will not exceed normal operating pressure during anticipated period of transport	After securing the inner closure head and prior to installing the outer closure head the inner container pressure will be checked. Internal pressures under normal shipping conditions are not critical in a dry system since the containment is designed to hold the pressures generated as a result of the hypothetical fire accident condition.
71.54(f)	Coolant Contamination	The cask design precludes release of the radioactive contents including the gaseous coolant. The coolant is considered as being part of the package contents. Therefore, the restrictions on coolant activity levels are not applicable.
71.62	Records	A record of each spent fuel shipment shall be prepared and maintained as required by Paragraph 71.62

Section V

SECTION V

COMPLIANCE WITH CFR, TITLE 49, CHAPTER 1, PART 170-189

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SECTION V

Compliance with CFR 49, Chapter 1

Section of Part	Requirement or Subject of Provision	Assessment of Compliance
173.393	General Packaging Requirements	Compliance with this part, except as specifically noted below, is described in Section IV, "Compliance with CFR, Title 10, Chapter 1, Part 71".
173.393 (b)	Evidence that package has not been illicitly opened	Prior to each shipment, two adjacent top impact structure bolts will be lockwired and a lead seal installed on the lockwire. The impact structure must be removed to gain access to outer head bolts.
173.393 (e) (2)	Accessible Surface Temperature	Fully loaded, in shade, assuming still air an ambient temperature of 130°F., the external surface of the cask body will be a temperature of 324°F without auxiliary cooling. However, the cask will be totally enclosed in an expanded metal personnel barrier. The personnel barrier is then the accessible surface of the package. The temperature of the personnel barrier will be less than the maximum permissible.
173.393 (h)	Surface Contamination	See response to 173.397 below.
173.393 (j)	Radiation Dose Rates	The radiation dose rate 6 feet from the accessible surface of the vehicle will not exceed 10 mrem/hr. See Section IX, "Shielding Analysis".

Section	Requirement or Subject of Provision	Assessment of Compliance
173.397	Surface Contamination	Prior to shipment, the exposed surfaces of the cask will be decontaminated to conform with the limits of this subpart. See Section XVI, "Operating Procedures".
173.399 (a) (3)	Labeling	The cask will be labeled Radioactive Yellow III.

Section VI

SECTION VI

MATERIAL SPECIFICATION

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SECTION VI

MATERIAL SPECIFICATIONS

Top Forging - ASME SA 336, Grade F8

Uranium - NLI Spec. No. 7065, Issue No. 1

Inner Shell - ASME SA 240, Type 304

Outer Shell - ASME SA 240, Type 304

Inner Bottom Forging - ASME SA 336, Grade F8

Outer Bottom Forging - ASME SA 336, Grade F8

Bottom Outer Ring Forging - ASME SA 336, Grade F8

Cooling Pipes - ASME SA 312, Type 304

Lead - ASTM B-29, Chemical Grade

Water Jacket Shell - ASME SA 240, Type 304

Fins - ASME SA 240, Type 304

Trunnions - ASME SA 336, Grade F8

Trunnion Sleeves - 17 -4 Ph conditioned per drawing requirements

Outer Closure Head Bolts - Republic A-286, Condition STA, ASTM 453, Grade 660

Inner Closure Head Bolts- Republic A-286, Condition STA, ASTM 453, Grade 660

Miscellaneous Fasteners - ASTM 193, Grade B7.

Neutron Shielding - Ricorad PPC-V

Fuel Basket - Aluminum 1180

Spacer Plug - Aluminum 5052-H32

Containment Vessel Valve Bolts - ASTM A193 Class 2 Grades B8, B8C, B8T.

Neutron Absorber Material -	Silver based alloy 80% Ag., 15 In. and 5% Cd. Clad in Type 304 Stainless Steel. Silver based alloy properties taken from Neutron Absorber Materials for Reactor Control, by W.K. Anderson and J.S. Theilacker, Naval Reactors, Division of Reactor Development, USAEC.
Accessories	
Pipe	ASTM-A-312, Type 304
Pipe Fittings	ASTM-A-182, Type 304
Tubing	ASTM-A-269, Type 304

Outer Closure Seal, "O" Ring - Viton

Mechanical and Physical Properties of Viton

Mechanical properties of Viton

Tensile strength 1735 psi

Elongation 190%

100% modulus, 880 psi

Hardness, Durometer A 79

Viton mechanical properties exceeds mechanical properties of AMS-7278D, MIL-R-83248, AMS-7280

Physical properties of Viton

Specific gravity 1.80 to 1.86

Coefficient of linear expansion 88×10^{-6} in./in./^{o}F.

WATER JACKET EXPANSION TANK FLEXIBLE HOSE

1. Flexible hose type: ANACONDA 1/2" BW-21-1H
2. Flexible hose end fitting type: PMW4
3. Threads are sealed with SILVER GOOP or equal.

See Pages VI-4, VI-5 and VI-6 for Manufacturers Specifications.

Corrugated Hose By Size

SIZES ½" THROUGH 1" I.D.

ANACONDA METAL HOSE

EXPLANATION OF COLUMN HEADINGS:

1. Bulletin Designations:

S	—Corrugated Bronze Hose
BW21, LW21, BWL21, BWA21, BWC21, BWM21	—Corrugated Stainless Steel Hose
BW10	—Corrugated Steel Hose
BW73	—Corrugated Monel Hose

2. Metal:
   B = Bronze CS = Steel
   SS = Stainless Steel ML = Monel
3. Minimum centerline bend radius specifications are for Constant Flexure. A smaller radius can be used for permanent bends. See specification tables.
4. Method of attaching fittings is not necessarily the same for each type of hose. Consult referenced page. For ordinary services, allow minimum safety factor of four to one. Where elevated temperatures are involved, consult "Temperature Factors for Elevated Temperatures" chart, Bulletin EN-1, page EN5.
5. Complete specifications for each item listed in this chart can be found on page listed under "Refer to Page No." column.

Nominal I.D. (in.)	Bulletin Designation	Metal	Braided (Brd) Unbraided (Unbrd)	Min. C/L Bend Radius (in.)	Rated Burst Pressure (psi) at room temp. with fittings attached	Refer to Page No.
[illegible]	S-0	B	Unbrd	4	—	CR2
	S-1	B	Brd	4	6,500	CR2
	S-2	B	2 Brd	4	10,000	CR2
3/16	S-0	B	Unbrd	4	—	CR2
	S-1	B	Brd	4	5,000	CR2
	S-2	B	2 Brd	4	9,000	CR2
	S-0-H	B	Unbrd	3	—	CR3
	S-1-H	B	Brd	3	5,000	CR3
	S-2-H	B	2 Brd	3	7,500	CR3
¼	S-0	B	Unbrd	4½	—	CR2
	S-1	B	Brd	4½	5,000	CR2
	S-2	B	2 Brd	4½	7,500	CR2
	S-0-H	B	Unbrd	3½	—	CR3
	S-1-H	B	Brd	3½	5,000	CR3
	S-2-H	B	2 Brd	3½	7,500	CR3
	BW21-0	SS	Unbrd	4	—	CR12
	LW21-1	SS	Brd	4	5,175	CR12
	BW21-1H	SS	Brd	4	10,000	CR14
	BW21-2H	SS	2 Brd	4	15,000	CR14
	BW10-0	CS	Unbrd	6	—	CR20
	BW10-1	CS	Brd	6	6,000	CR20
	BW10-2	CS	2 Brd	6	9,000	CR20
	BW73-0	ML	Unbrd	4	—	CR22
	BW73-1	ML	Brd	4	7,000	CR22
	BW73-2	ML	2 Brd	4	10,500	CR22
5/16	S-0	B	Unbrd	4½	—	CR2
	S-1	B	Brd	4½	4,500	CR2
	S-2	B	2 Brd	4½	6,500	CR2
	S-0-H	B	Unbrd	4½	—	CR3
	S-1-H	B	Brd	4½	4,500	CR3
	S-2-H	B	2 Brd	4½	6,500	CR3
	S-0	B	Unbrd	5	—	CR2
	S-1	B	Brd	5	4,000	CR2
	S-2	B	2 Brd	5	6,000	CR2
	S-0-H	B	Unbrd	4½	—	CR3
	S-1-H	B	Brd	4½	4,000	CR3
	S-2-H	B	2 Brd	4½	6,000	CR3
	BW21-0	SS	Unbrd	6	—	CR12
	BW21-1	SS	Brd	6	4,025	CR12
	BW21-1H	SS	Brd	6	10,000	CR14
	BW21-2H	SS	2 Brd	6	15,000	CR14
⅜	BW10-0	CS	Unbrd	7½	—	CR20
	BW10-1	CS	Brd	7½	4,500	CR20
	BW10-2	CS	2 Brd	7½	6,750	CR20
	BW73-0	ML	Unbrd	4½	—	CR22
	BW73-1	ML	Brd	4½	6,000	CR22
	BW73-2	ML	2 Brd	4½	9,000	CR22
½	S-0	B	Unbrd	6	—	CR2
	S-1	B	Brd	6	3,000	CR2
	S-2	B	2 Brd	6	5,000	CR2
	S-0-H	B	Unbrd	6½	—	CR3
	S-1-H	B	Brd	6½	3,000	CR3
	S-2-H	B	2 Brd	6½	5,000	CR3
	BW21-0	SS	Unbrd	8	—	CR12
	BW21-1	SS	Brd	8	4,025	CR12
	BW21-1H	SS	Brd	8	9,000	CR14
	BW21-2H	SS	2 Brd	8	13,500	CR14
	BW10-0	CS	Unbrd	8½	—	CR20
	BW10-1	CS	Brd	8½	3,500	CR20
	BW10-2	CS	2 Brd	8½	5,250	CR20
	BW73-0	ML	Unbrd	7½	—	CR22
	BW73-1	ML	Brd	7½	5,000	CR22
	BW73-2	ML	2 Brd	7½	7,500	CR22
¾	S-0	B	Unbrd	7	—	CR2
	S-1	B	Brd	7	2,500	CR2
	S-2	B	2 Brd	7	4,500	CR2
	S-0-H	B	Unbrd	7	—	CR3
	S-1-H	B	Brd	7	2,500	CR3
	S-2-H	B	2 Brd	7	4,500	CR3
	LW21-0	SS	Unbrd	7½	—	CR12
	LW21-1	SS	Brd	7½	2,800	CR12
	BWL21-0	SS	Unbrd	7½	—	CR13
	BWL21-1	SS	Brd	7½	2,800	CR13
	BWL21-1H	SS	Brd	7½	4,200	CR13
	BW21-0	SS	Unbrd	8	—	CR14
	BW21-1H	SS	Brd	8	6,000	CR14
	BW21-2H	SS	2 Brd	8	9,000	CR14
	BW10-0	CS	Unbrd	10	—	CR20
	BW10-1	CS	Brd	10	3,000	CR20
	BW10-2	CS	2 Brd	10	4,500	CR20
	BW73-0	ML	Unbrd	8	—	CR22
	BW73-1	ML	Brd	8	3,700	CR22
	BW73-2	ML	2 Brd	8	5,550	CR22
1	S-0	B	Unbrd	8	—	CR2
	S-1	B	Brd	8	2,200	CR2
	S-2	B	2 Brd	8	3,500	CR2
	S-0-H	B	Unbrd	8	—	CR3
	S-1-H	B	Brd	8	2,200	CR3
	S-2-H	B	2 Brd	8	3,500	CR3
	LW21-0	SS	Unbrd	8½	—	CR12
	LW21-1	SS	Brd	8½	2,300	CR12
	BWL21-0	SS	Unbrd	8½	—	CR13
	BWL21-1	SS	Brd	8½	2,300	CR13
	BWL21-1H	SS	Brd	8½	3,450	CR13
	BW21-0	SS	Unbrd	9	—	CR14
	BW21-1H	SS	Brd	9	5,200	CR14
	BW21-2H	SS	2 Brd	9	7,750	CR14
	BW10-0	CS	Unbrd	12	—	CR20
	BW10-1	CS	Brd	12	2,500	CR20
	BW10-2	CS	2 Brd	12	3,750	CR20
	BW73-0	ML	Unbrd	8	—	CR22
	BW73-1	ML	Brd	8	3,000	CR22
	BW73-2	ML	2 Brd	8	4,500	CR22

STAINLESS STEEL CORRUGATED HOSE

Heavy Weight—For High Pressures

SIZES ¼" THROUGH 14" I.D.

TYPES

BW21-0 BWM21-0	Corrugated Stainless Steel, unbraided	BW21-1H-C	Corrugated Stainless Steel, one heavy wire braid, overall stainless steel protective casing
BWM21-1	Corrugated Stainless Steel, one wire braid	BW21-2H	Corrugated Stainless Steel, two heavy wire braids
BW21-1H	Corrugated Stainless Steel, one heavy wire braid	BW21-2H-C	Corrugated Stainless Steel, two heavy wire braids, overall stainless steel protective casing

USES Because of its construction it is able to withstand higher pressures than standard or medium weight hose. This heavier construction also makes it more suitable for some corrosive applications. Type BWM21 has the same properties as type BW21 and is available in sizes 5"-14" I.D.

CONSTRUCTION Types BW21 and BWM21 — Made with annular corrugations from butt welded Type 321 stainless steel tubing. Covered with Type 321 stainless steel wire braid. Both BW and BWM available in Type 316 stainless steel. Specify BW16 (—0, —1) and BWM16 (—0, —1) when ordering.

FITTINGS See pages CR15 through CR19 for specifications on standard fittings.

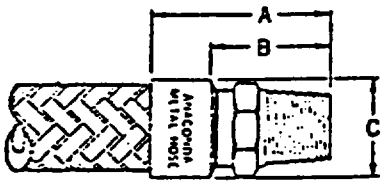
NOMINAL I.D. (in.)	DESIGNATION	MAXIMUM O.D. (in.)	MINIMUM C/L BEND RADIUS FOR: Constant Flexure (in.)	MINIMUM C/L BEND RADIUS FOR: Permanent Bend (in.)	MINIMUM EXPOSED LENGTH FOR NORMAL VIBRATION (in.)	PRESSURE DATA: Max. Pressure Unbraided and Unrestrained (psi)	PRESSURE DATA: RATED BURST PRESSURE (psi): Welded Fittings	PRESSURE DATA: RATED BURST PRESSURE (psi): Silver Brazed Fittings	PRESSURE DATA: Maximum Test Pressure (psi)	APPROX. WEIGHT PER FOOT (lbs.)
¼	BW21-0	.455	4	1	—	255	—	—	4200	.09
	BW21-1H	.535			4½	—	9000	10000		.19
	BW21-2H	.615			4½	—	10000	15000		.28
	BW21-0	.645	6	1½	—	255	—	—	4000	.18
	BW21-1H	.755			5	—	8500	10000		.35
	BW21-2H	.865			5	—	10000	15000		.52
½	BW21-0	.790	8	1½	—	255	—	—	4000	.23
	BW21-1H	.900			5½	—	8000	9000		.46
	BW21-2H	1.010			5½	—	10000	13500		.69
¾	BW21-0	1.125	8	2	—	80	—	—	2600	.50
	BW21-1H	1.235			6	—	5200	6000		.81
	BW21-2H	1.365			6	—	6000	9000		1.12
1	BW21-0	1.465	9	3	—	60	—	—	2200	.69
	BW21-1H	1.575			7	—	4500	5200		1.12
	BW21-2H	1.705			7	—	6000	7750		1.55
1¼	BW21-0	1.775	9	3¼	—	35	—	—	1750	.95
	BW21-1H	1.885			8	—	3500	4000		1.46
	BW21-2H	1.995			8	—	5000	6000		1.97
1½	BW21-0	2.090	10	3¼	—	25	—	—	1750	1.35
	BW21-1H	2.260			8½	—	3500	4000		2.11
	BW21-2H	2.450			8½	—	5000	6000		2.87
2	BW21-0	2.715	11½	5⅜	—	12	—	—	1050	1.99
	BW21-1H	2.885			9½	—	2100	2400		2.97
	BW21-2H	3.075			9½	—	3000	3600		3.95
2½	BW21-0	3.340	13	6	—	6	—	—	850	2.91
	BW21-1H	3.510			10½	—	1500	1725		4.05
	BW21-2H	3.700			10½	—	2000	2600		5.19
3	BW21-0	3.965	18	7½	—	6	—	—	600	3.11
	BW21-1H	4.135			11	—	1400	1600		4.51
	BW21-2H	4.325			11	—	2000	2400		5.91
5	BWM21-0	5.825	87	22	—	8	—	—	650	5.24
	BWM21-1	6.045			13		1100	1250		7.32
6	BWM21-0	6.950	95	24	—	6	—	—	600	6.28
	BWM21-1	7.170			14½		800	900		8.48
8	BWM21-0	9.125	105	28	—	3.5	—	—	500	9.68
	BWM21-1	9.385			16½		800	900		13.58
	BWM21-0	11.245	110	32	—	3	—	—	450	10.23
	BWM21-1	11.555			17½		600	675		14.46
12	BWM21-0	13.420	114	35	—	2.25	—	—	350	13.76
	BWM21-1	13.830			19½		600	675		18.32
14	BWM21-0	14.725	116	37	—	1.5	—	—	275	16.68
	BWM21-1	15.135			21		500	575		22.00

Notes: 1. In order that assemblies made from above listed hose, (BW21 2H), attain the same burst capabilities as the hose, careful consideration must be given to fitting design and method of attaching. Unless complete assemblies are ordered, consult Anaconda Metal Hose Engineering Department.
2. Sizes larger than 14" available on application.

See "Explanation of Specification Column Headings and Terminology", page CR1.

FITTINGS FOR TYPES LW, BWL and BW STAINLESS STEEL HOSE

HEX MALE (¼"-¾")

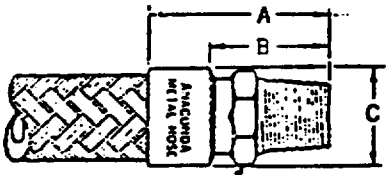


HMW4	304 Stainless Steel fitting—welded
HMS4	304 Stainless Steel fitting—silver brazed
HMW	321 or 316 Stainless Steel fitting—welded
HMS	321 or 316 Stainless Steel fitting—silver brazed
SHMW	Carbon Steel fitting—welded
SHMS	Carbon Steel fitting—silver brazed

NPT THD. AND HOSE I.D. (in.)	A (in.)	B (in.)	C (in.)
¼	1.500	1.125	.650
⅜	1.625	1.125	.794
½	1.937	1.437	1.013
¾	2.187	1.437	1.372

HEX MALE (1"-4")

SCH 40-or-40S PIPE



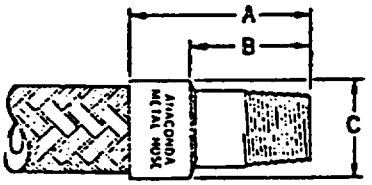
MALLEABLE IRON LOCK NUT

HMW4	304 Stainless Steel fitting—welded
HMS4	304 Stainless Steel fitting—silver brazed
HMW	321 or 316 Stainless Steel fitting—welded
HMS	321 or 316 Stainless Steel hose—silver brazed
SHMW	Carbon Steel fitting—welded
SHMS	Carbon Steel fitting—silver brazed

NPT THD. AND HOSE I.D. (in.)	A (in.)	B (in.)	C (in.)
1	3.000	2.000	1.791
1¼	3.500	2.500	2.196
1½	3.750	2.500	2.522
2	3.750	2.500	3.010
2½	4.250	3.000	3.635
3	4.250	3.000	4.360
3½	5.250	4.000	5.037
4	5.250	4.000	5.720

PLAIN MALE (¼"-4")

SCH 40 or 40S PIPE

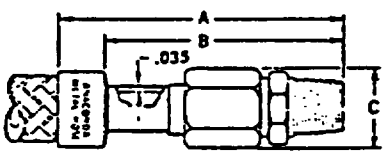


PMW4	304 Stainless Steel fitting—welded
PMS4	304 Stainless Steel fitting—silver brazed
PMW	321 or 316 Stainless Steel fitting—welded
PMS	321 or 316 Stainless Steel fitting—silver brazed
SPMW	Carbon Steel fitting—welded
SPMS	Carbon Steel fitting—silver brazed

NOTE: Sizes ¼" through ¾" available only in stainless steel.

NPT THD. AND HOSE I.D. (in.)	A (in.)	B (in.)	C (in.)
¼	1.875	1.500	.639
⅜	2.125	1.625	.859
½	2.250	1.750	1.018
¾	2.500	1.750	1.353
1	3.000	2.000	1.693
1¼	3.500	2.500	2.147
1½	3.750	2.500	2.52.
2	3.750	2.500	3.010
2½	4.250	3.000	3.635
3	4.250	3.000	4.260
3½	5.250	4.000	4.920
4	5.250	4.000	5.720

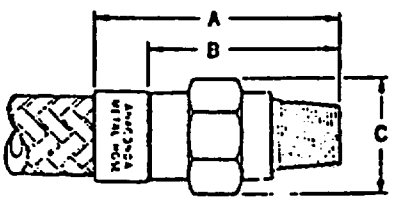
MALE UNION (¼"-1")



MUW	321 Tube End, with 18-8 stainless steel nut, sleeve and adapter—welded
	316 Tube End and Adapter, with 18-8 stainless steel nut and sleeve—welded
MUS	321 Tube End, with 18-8 stainless steel nut, sleeve and adapter—silver brazed
	316 Tube End and Adapter, with 18-8 stainless steel nut and sleeve—silver brazed

NPT THD. AND HOSE I.D. (in.)	A (in.)	B (in.)	C (in.)
¼	2.625	2.250	.796
⅜	2.843	2.343	1.015
½	3.390	2.890	1.156
¾	3.734	2.984	1.453
1	4.250	3.250	1.734

MALE UNION (¼"-3")



SMUWD	3000 lb. Forged Steel union—welded
SMUSD	3000 lb. Forged Steel union—silver brazed

NPT THD. AND HOSE I.D. (in.)	A (in.)	B (in.)	C (in.)
¼	3.125	2.750	1.531
⅜	3.325	2.825	1.813
½	3.719	3.219	1.938
¾	4.156	3.406	2.375
1	4.781	3.781	2.688
1¼	5.031	4.031	3.125
1½	5.563	4.313	3.438
2	6.125	4.875	4.250
2½	6.813	5.563	4.813
3	7.281	6.031	6.125

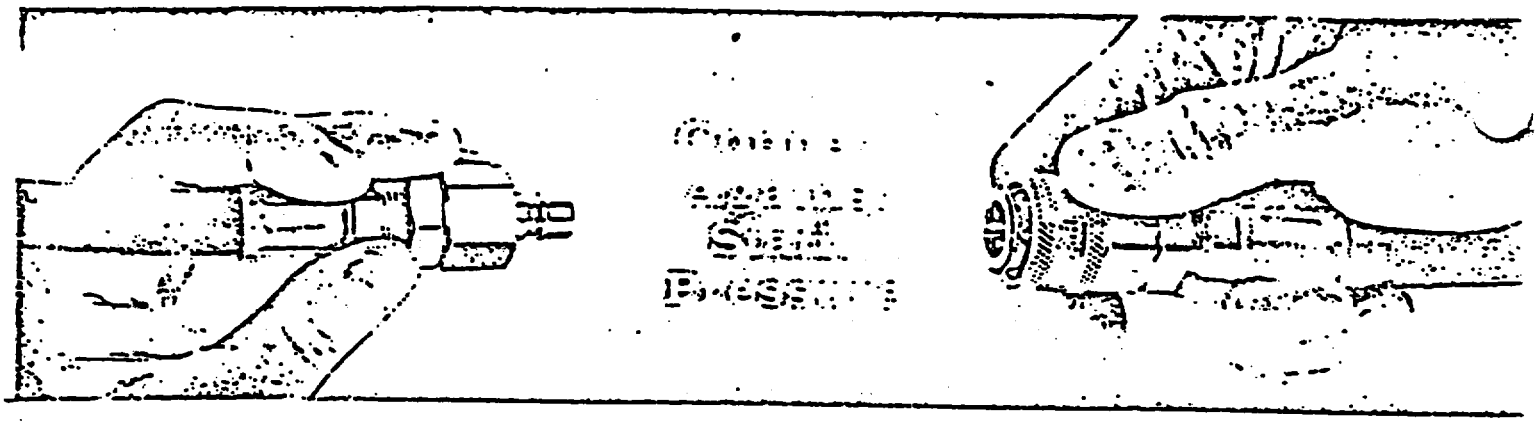
Notes:
1. All of the above fittings can be used on Types 321 or 316 stainless steel hose. Type 321 and 304 fittings are normally used on 321 hose. Type 316 fittings are normally used on 316 hose.
2. Maximum temperature limit for stainless steel hose with stainless steel fittings attached by welding is 1500°F. — by silver brazing 600°F.
3. Maximum temperature limit for stainless steel hose with steel fittings attached by welding is 850°F. — by silver brazing 600°F.
4. To make up an assembly of a required overall length figure length to which hose should be cut by adding fitting dimension "B" for both end fittings and subtracting the total from overall length of assembly.
5. "C" represents the maximum diametral envelope dimension.

CR15

WATER JACKET AND EXPANSION TANK QUICK DISCONNECT COUPLING

1. Valved Coupler type: Snap-Tite Cat. No. VHC8-8F W/SL
2. Valved Nipple type: Snap-Tite Cat. No. VPHN8-8M
3. "O" Ring Seal Material: Viton
4. Threads are sealed with SILVER GOOP or equal.

See Pages VI-8 and VI-9 for Manufacturers Specifications.



New Capability for Hydraulic Applications

Now . . . a quick-disconnect you can connect against trapped pressure. Snap-Tite PH Series quick-disconnect couplings* give all the time-proved performance features of Snap-Tite H Series quick-disconnects. In addition, they can be connected against static pressure . . . for fast, easy connection of hydraulic lines, and for other applications that call for periodic connection against trapped pressure.

The pressure connector feature can be supplied in either the coupler half or nipple half. PH Series couplers and nipples are completely interchangeable with H Series valved couplers and nipples of the same size.

Thus, PH Series units can be readily incorporated as required in new fluid systems that are designed for H Series quick-disconnects. And they can easily be substituted for H Series valved units in existing field installations.

Snap-Tite's PH Series couplings have been successfully field-tested in connecting against static pressures of 3000 psi. They are especially suited for such applications as hydraulic systems on farm implements, construction, utility and materials handling equipment, and on any fluid power system in which removable attachments are supplied through lines from a hydraulic power unit.

Pressure Data	3/8"	1/2"	3/4"	1"
Working Pressures (psig)	4500	4000	3500	2000

Proof Pressures: 1½ times working pressures.
Burst Pressures: 2 times working pressures.

NOTE: *For detailed information on other features plus complete dimensions, weights and technical data, please refer to Snap-Tite H Series Catalog.*

TWO TYPES

1. PH SERIES VALVED COUPLER that mates with H Series valved nipple.

2. PH SERIES VALVED NIPPLE th mates with H Series valved coupler.

All couplers or nipples of the same size are interchangeable regardless of end fitting.

FOUR SIZES. 3/8", 1/2", 3/4", 1".

EIGHT END FITTINGS. Female NPT, Male NPT, Hose Shank, AND 10050 (female), SAE Female, MS33656 (male), SAE (JIC) 37° Male Flare, MS33657 (bulkhead). Other fittings on request.

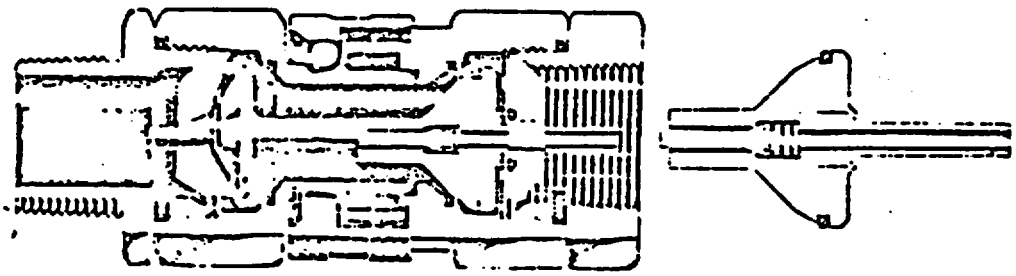
METAL AND FINISH. Heat-treated alloy steel with cadmium-chromate plating.

CAPS AND PLUGS. Dust caps, pressure caps or dust plugs protect disconnected couplers or nipples from damage, dirt and other contaminants.

SLEEVE LOCK. Prevents accidental disconnection of coupling. Sleeve cannot be moved to retracted position until pin is in line with slot in sleeve.

Instant, Automatic Bleed-Off

Easy connection against static pressure is assured by the special poppet valve. When the coupling halves are being connected, the small bleeder valve inside the PH Series shut-off valve is depressed to the open position. Trapped pressure that bears upon the larger surfaces of the shut-off valve is instantly and automatically reduced, permitting effortless engagement.



Coupling shown is H Series valved nipple and PH Series valved coupler.

*Patents Applied For

PH SERIES COUPLINGS WITH END FITTINGS

Ordering Information

Select the proper coupling—consisting of PH or H Series coupler and PH or H Series nipple—from the table.

ANTI-EXTRUSION SEAL SET. This seal set can be supplied as standard instead of the U-Packer seal. See H Series Catalog for details. If it is desired, please suffix the part number with -9. Example: VPHC6-6F-9.

SLEEVE LOCK. To order, add W/SL to part number. Example: VPHC6-6F W/SL.

STEEL COUPLINGS WITH END FITTINGS

Part Numbers by Size

PH Series Valved Couplers

End Fitting	3/8"	1/2"	3/4"	1"
Female NPT	VPHC6-6F	VPHC8-8F	VPHC12-12F	VPHC16-16F
Male NPT	VPHC6-6M	VPHC8-8M	VPHC12-12M	VPHC16-16M
Hose Shank	VPHC6-6H	VPHC8-8H	VPHC12-12H	VPHC16-16H
AND10050	VPHC6-6-S0	VPHC8-8-S0	VPHC12-12-S0	VPHC16-16-S0
MS33656	VPHC6-6-S6	VPHC8-8-S6	VPHC12-12-S6	VPHC16-16-S6
MS33657	VPHC6-6-S7	VPHC8-8-S7	VPHC12-12-S7	VPHC16-16-S7
SAE Female	VPHC6-6-EF	VPHC8-8-EF	VPHC12-12-EF	VPHC16-16-EF
SAE Male	VPHC6-6-EM	VPHC8-8-EM	VPHC12-12-EM	VPHC16-16-EM

H Series Valved Couplers

End Fitting	3/8"	1/2"	3/4"	1"
Female NPT	VHC6-6F	VHC8-8F	VHC12-12F	VHC16-16F
Male NPT	VHC6-6M	VHC8-8M	VHC12-12M	VHC16-16M
Hose Shank	VHC6-6H	VHC8-8H	VHC12-12H	VHC16-16H
AND10050	VHC6-6-S0	VHC8-8-S0	VHC12-12-S0	VHC16-16-S0
MS33656	VHC6-6-S6	VHC8-8-S6	VHC12-12-S6	VHC16-16-S6
MS33657	VHC6-6-S7	VHC8-8-S7	VHC12-12-S7	VHC16-16-S7
SAE Female	VHC6-6-EF	VHC8-8-EF	VHC12-12-EF	VHC16-16-EF
SAE Male	VHC6-6-EM	VHC8-8-EM	VHC12-12-EM	VHC16-16-EM

PH Series Valved Nipples

End Fitting	3/8"	1/2"	3/4"	1"
Female NPT	VPHN6-6F	VPHN8-8F	VPHN12-12F	VPHN16-16F
Male NPT	VPHN6-6M	VPHN8-8M	VPHN12-12M	VPHN16-16M
Hose Shank	VPHN6-6H	VPHN8-8H	VPHN12-12H	VPHN16-16H
AND10050	VPHN6-6-S0	VPHN8-8-S0	VPHN12-12-S0	VPHN16-16-S0
MS33656	VPHN6-6-S6	VPHN8-8-S6	VPHN12-12-S6	VPHN16-16-S6
MS33657	VPHN6-6-S7	VPHN8-8-S7	VPHN12-12-S7	VPHN16-16-S7
SAE Female	VPHN6-6-EF	VPHN8-8-EF	VPHN12-12-EF	VPHN16-16-EF
SAE Male	VPHN6-6-EM	VPHN8-8-EM	VPHN12-12-EM	VPHN16-16-EM

H Series Valved Nipples

End Fitting	3/8"	1/2"	3/4"	1"
Female NPT	VHN6-6F	VHN8-8F	VHN12-12F	VHN16-16F
Male NPT	VHN6-6M	VHN8-8M	VHN12-12M	VHN16-16M
Hose Shank	VHN6-6H	VHN8-8H	VHN12-12H	VHN16-16H
AND10050	VHN6-6-S0	VHN8-8-S0	VHN12-12-S0	VHN16-16-S0
MS33656	VHN6-6-S6	VHN8-8-S6	VHN12-12-S6	VHN16-16-S6
MS33657	VHN6-6-S7	VHN8-8-S7	VHN12-12-S7	VHN16-16-S7
SAE Female	VHN6-6-EF	VHN8-8-EF	VHN12-12-EF	VHN16-16-EF
SAE Male	VHN6-6-EM	VHN8-8-EM	VHN12-12-EM	VHN16-16-EM

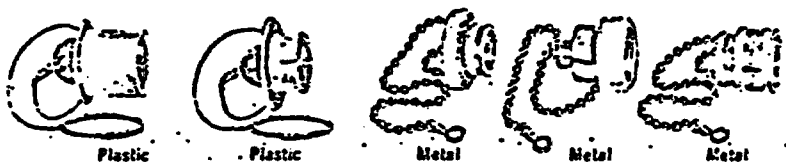
Dust Caps, Pressure Caps, Dust Plugs

PLASTIC. Dust Caps for nipples and Dust Plugs for couplers are supplied complete with corrosion-resistant steel cable.

METAL: Pressure-tight Pressure Caps for nipples and Dust Plugs for couplers are available in cadmium chromate plated steel and clear anodized aluminum. Dust Caps for nipples are also available in aluminum. On sizes through 3/4", a 10-inch long brass chrome-plated bead chain is supplied. On 1" size, a 10-inch long cadmium-plated steel sash chain is supplied.

Special lengths of chains are available on request. Ordering designations are: W/BC for bead chain, W/SC for sash chain.

TO ORDER: For standard caps and plugs, use the part numbers given in table. For special lengths, add chain designation and length in inches to the part number. Example: MCH-6 W/C-8.



Plastic Dust Cap · Plastic Dust Plug · Metal Dust Plug · Metal Dust Cap · Metal Pressure Cap

	3/8"	1/2"	3/4"	1"
PLASTIC				
DUST CAP	PDCH-6	PDCH-8	PDCH-12	PDCH-16
DUST PLUG	PDPH-6	PDPH-8	PDPH-12	PDPH-16
STEEL	3/8"	1/2"	3/4"	1"
PRESSURE CAP	MCH-6	MCH-8	MCH-12	MCH-16
DUST PLUG	MPH-6	MPH-8	MPH-12	MPH-16
ALUMINUM	3/8"	1/2"	3/4"	1"
PRESSURE CAP	AMCH-6	AMCH-8	AMCH-12	AMCH-16
DUST CAP	ADCH-6	ADCH-8	ADCH-12	ADCH-16
DUST PLUG	AMPH-6	AMPH-8	AMPH-12	AMPH-16

Spare Parts

Valve assembly, valved coupler body and valved nipple body for PH Series are shown. To order, specify (1) coupling series, (2) size of coupling and (3) spare part number.

Example: You wish to order a valve assembly for a 1/2" PH Series coupler or nipple. Your order would read: PH8-42.

All other PH Series spare parts are the same as for H Series quick-disconnects. Please refer to H Series Catalog for ordering information.

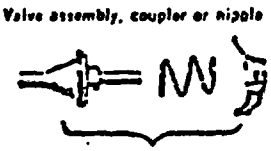
Valved coupler body

05

Valved nipple body

06

Valve assembly, coupler or nipple



42

INC.
Union City, Pa. 16438

Printed i

WATER JACKET EXPANSION TANK DRAIN AND FILL VALVE

1. Valve type: Hex. Eng. HGV-48S, with Stellited Ball on a 316 S/S Stem.
2. Valve body material: 304 S/S.
3. Valve inlet: 1/2" Male NPT

See Pages VI-11 and VI-12 for Manufacturers Specifications.

GAUGE AND INSTRUMENTATION VALVE

TYPE
HGV 48
HGV 48S

PURPOSE

Hex Instrument Valves HGV-48 and HGV-48S are designed with multiple outlets to meet the valve-users needs. Each Hex valve is hand-assembled by skilled craftsmen and individually tested to provide valves of the highest quality.

FEATURES

- Three Seat Types Available
- Replaceable Soft Seat HGV-48S
- Bubble-tight Shut-off
- Backseat Stem — protects packing during operation
- Three outlets
- Inlet connection Schedule 160 or heavier
- John Crane 187-I 1200° F Packing Rings
- Straight through passage (Roddable) on Soft Seat

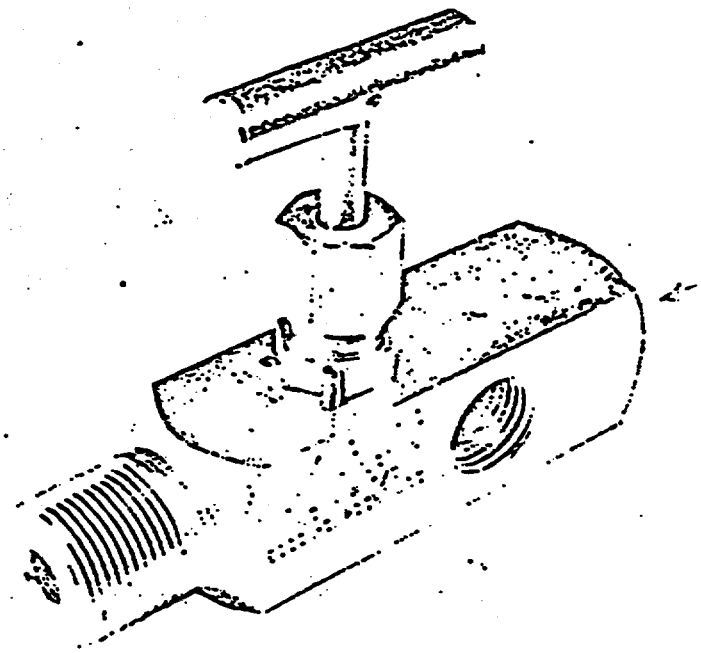
OPTIONS

Directional Safety Tube Bleed Valve — See Bulletin h-200
Teflon Chevron Packing Rings
Kel-F Replaceable Soft Seat Insert
316 Stainless Steel Ball Stem

CONNECTIONS

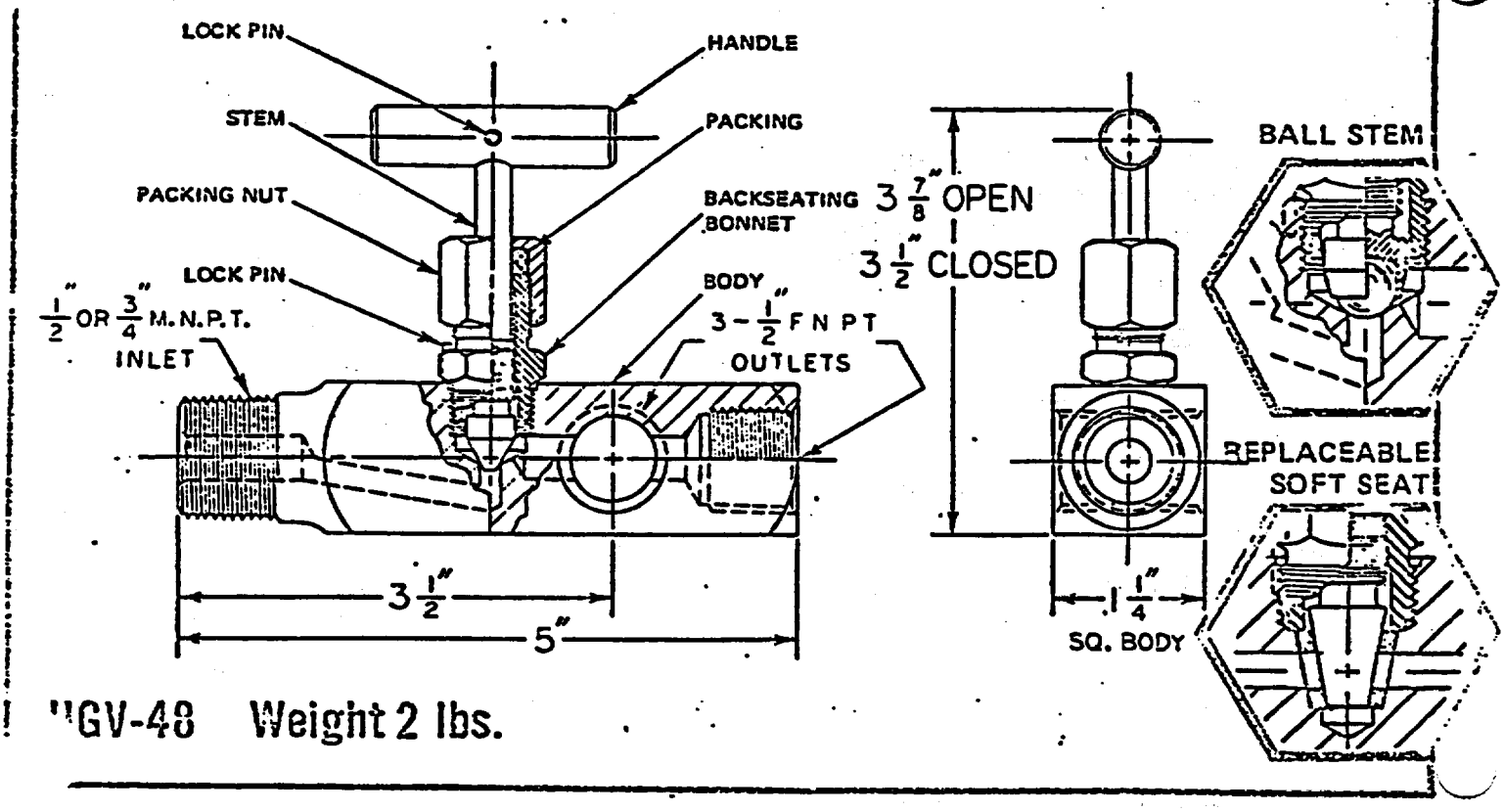
One Inlet
½" or ¾" Male NPT or Socketweld

Three Outlets
¼" or ½" Female NPT



Backseated
Screwed Packing Gland

HEX VALVES ARE USED WORLDWIDE FOR SAFETY AND PERFORMANCE.



HGV-48 Weight 2 lbs.

MATERIALS

Standard Hex Valve body materials are plated carbon steel, 304 and 316 stainless steel. Special materials such as Inconel, Incoloy, etc. are available. 303 stainless steel trim is furnished in both carbon steel and 304 stainless steel valves. 316 stainless steel valves have 316 stainless steel trim. Special trim materials are optional. John Crane 187-I 1200° F Packing Rings are furnished, with other packing materials optional.

PRESSURE-TEMPERATURE RATING

6000 PSI @ 200°F

WHEN ORDERING PLEASE SPECIFY (EXAMPLE)							
QUANTITY	VALVE TYPE	MATERIAL				ONE INLET CONNECTION	THREE OUTLET CONNECTION
		BODY	TRIM	PACKING	SEAT		
10	HGV-48S	C/S	303	J.C. 187-I	PENTON	¾" MALE NPT	½" x ½" x ½" FNPT

Represented by:

HEX VALVES

HEX ENGINEERING

(714) 870-7474

A DIVISION OF CIRCLE SEAL CORPORATION • BOX 3666 • ANAHEIM, CALIFORNIA 92803

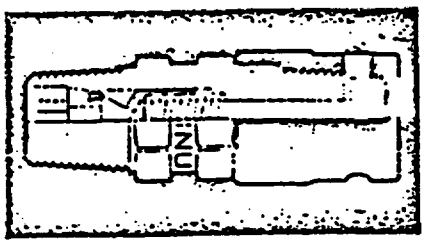
WATER JACKET AND WATER JACKET EXPANSION TANK RELIEF VALVE

1. Valve type: NUPRO No. 8CPA2-150, with Viton "O" Ring Seal.
2. Valve body material: 316 S/S
3. Threads are sealed with SILVER GOOP or equal.

See Pages VI-14, VI-15 and VI-16 for Manufacturers Specifications.

NUPRO
"CP" & "CPA" SERIES PIPE ENDED CHECK, ADJUSTABLE RELIEF & POPOFF RELIEF VALVES

DEFLECTOR CAP



The green polyethylene Deflector Cap is available for valves used as popoff-to-atmosphere relief valves. It deflects the flow from direct contact with personnel and prevents atmospheric contaminants from entering the valve. The Deflector Cap easily screws on a standard male NPT pipe thread. To order separately, the part number is P_e-4CP4 for ¼" male NPT; P_e-8CP4 for ½" male NPT.

FLOW CAPACITY

"4CP-4CPA" SERIES MAXIMUM FLOW FOR $C_v = 0.35$		
PRESSURE DROP TO ATMOSPHERE (ΔP) PSI	AIR SCFM @70°F	WATER GPM @70°F
10	4.82	1.11
50	13.25	2.48
100	23.49	3.50

"8CP"-"8CPA" SERIES MAXIMUM FLOW FOR C = 1.2		
PRESSURE DROP TO ATMOSPHERE (ΔP) PSI	AIR SCFM @70°F	WATER GPM @70°F
10	16.69	3.79
50	45.44	8.49
100	80.55	12.00

MATERIALS

Body—Brass, anodized aluminum, 316 stainless steel, cadmium plated carbon steel.

Spring—302 stainless steel.

O-Ring—Buna "N" is standard in all valves. Many different types of elastomer O-Rings are stocked by NUPRO Company for specialized applications.

Insert, Poppet—Same material as body except steel valves use 316 stainless steel.

Insert Lock Screw ("4CP-4CPA" Series)—Same material as body. ("8CP-8CPA" Series)—Brass in brass valves; 316 stainless steel in all other valves

Adjusting Screw, Lock Screw ("4CPA" Series only)—303 stainless steel, except 316 valves use 316 stainless steel. ("8CPA" Series only)—316 stainless steel except brass valves are brass.

TECHNICAL DATA

ITEM	"CP" SERIES	"CPA" SERIES
STANDARD CRACKING PRESSURE (PSI)	1	not applicable
ADJUSTABLE RANGE (PSI)	not applicable	3 to 50 50 to 150 150 to 350 350 to 600
WORKING PRESSURE RATING @70°F (PSI)	2500	2500
BACK PRESSURE RATING @70°F (PSI)	2500	2500
FLOW COEFFICIENT, C_v	0.35 ("4CP" Series) 1.2 ("8CP" Series)	0.35 ("4CPA" Series) 1.2 ("8CPA" Series)
TEMPERATURE RATINGS	BUNA "N": General: -40°F to 300°F With Alkalies: -40°F to 158°F	

TABLE OF DIMENSIONS

CATALOG NUMBER	NOMINAL CRACKING PRESSURE RANGE (PSI)	END CONNECTIONS		OVERALL LENGTH	HEX SIZE
		INLET	OUTLET		
-4CP2-1	1	¼ Male NPT	¼ Male NPT	1⅜	9/16
-4CP5-1	1	¼ Male NPT	¼ Female NPT	1¾	¾
-4CPA2-3	3 to 50	¼ Male NPT	¼ Male NPT	1⅜	9/16
-4CPA2-50	50 to 150	¼ Male NPT	¼ Male NPT	1⅜	9/16
-4CPA2-150	150 to 350	¼ Male NPT	¼ Male NPT	1⅜	9/16
-4CPA2-350	350 to 600	¼ Male NPT	¼ Male NPT	1⅜	9/16
-8CP2-1	1	½ Male NPT	½ Male NPT	2 5/32	⅞
-8CP5-1	1	½ Male NPT	½ Female NPT	2¾	1 1/16
-8CPA2-3	3 to 50	½ Male NPT	½ Male NPT	2 9/16	⅞
-8CPA2-50	50 to 150	½ Male NPT	½ Male NPT	2 9/16	⅞
-8CPA2-150	150 to 350	½ Male NPT	½ Male NPT	2 9/16	⅞
-8CPA2-350	350 to 600	½ Male NPT	½ Male NPT	2 9/16	⅞

① For a complete ordering number:

a. Prefix B for brass, SS for 316 stainless steel, A for aluminum, and S for carbon steel.

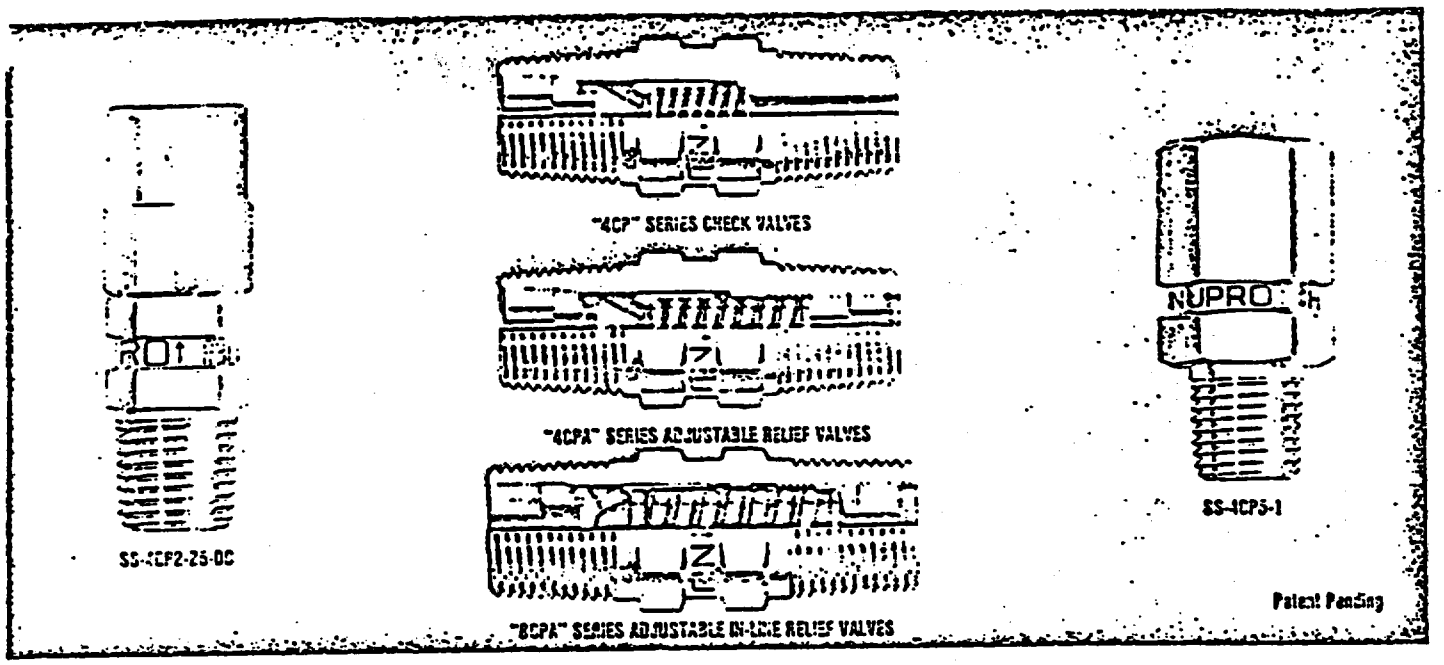
b. To order valves with Deflector Cap installed, suffix the catalog number with -DC. To order Deflector Caps separately, the part number is P_e-4CP4 for ¼" male NPT; P_e-8CP4 for ½" male NPT.

c. Special springs from ½ to 100 psi cracking pressure are available for "CP" Series valves. To order, change the -1 suffix to -½, -10, etc.

NUPRO COMPANY • 15635 Saranac Road • Cleveland, Ohio 44110

NUPRO

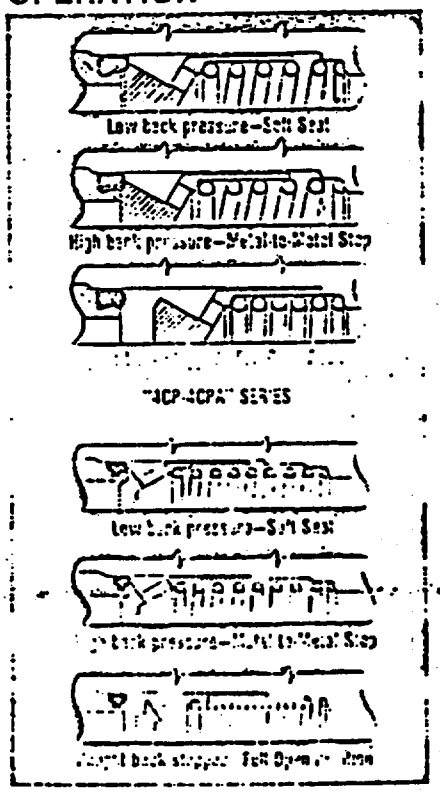
"CP" & "CPA" SERIES PIPE ENDED CHECK, ADJUSTABLE RELIEF & POPOFF RELIEF VALVES



PURPOSE

NUPRO "CP" Series Check and "CPA" Series Adjustable In-line Relief Valves fill the need for improved, compact, inexpensive, pipe ended check and relief valves.

OPERATION



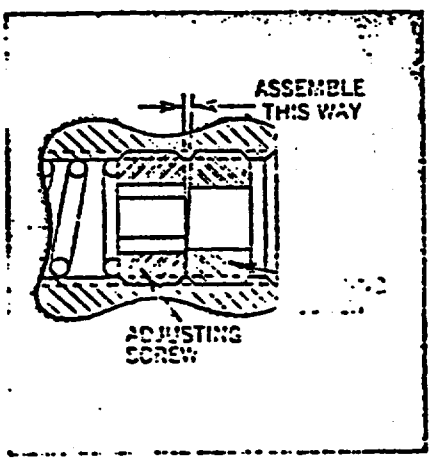
APPLICATIONS

Unidirectional flow control of liquids and gases • Wide variety of industrial, laboratory and process control systems • Prevention of undesirable mixing of two fluid streams • Protection against over-pressurization • Vacuum breaker.

SPECIAL FEATURES

Soft, resilient O-Ring gently stops poppet • Large seating area provides sensitive opening pressures and leak-tight sealing • Contained O-Ring eliminates O-Ring blowout • O-Ring is easily replaced • Metal-to-metal stop prevents over-squeeze of O-Ring and eliminates sticking • No leak path to atmosphere with one piece body • Dual hex distinguishes valves from hex pipe nipples • Smooth floating poppet and soft O-Ring permit chatter-free operation • Smooth flow path provides minimum pressure drop • O-Ring removed from main flow path • Backstopped poppet prevents overstressing spring.

ADJUSTMENT



To adjust "CPA" Relief Valves:

- Insert a standard hex key wrench into the locking screw.
- Break locking screw counter-clockwise until the hex key wrench slides into the adjusting screw.
- Turn both screws to the desired cracking pressure. (Clockwise increases opening pressure).
- Retract hex key wrench into the locking screw.
- Lock against adjusting screw by turning clockwise.

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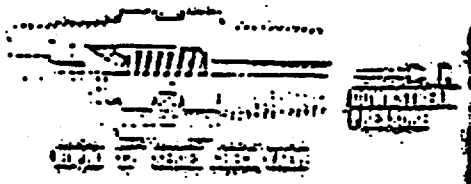
CHECK AND RELIEF VALVES



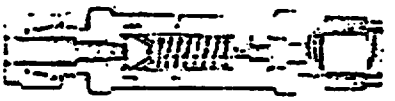
"SCP" Series Check Valves
Pages 2 & 3

Union Bonnet LIR Check Valves
Page 6

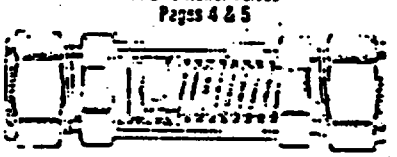
"SCPA" Series Adjustable
In-Line Relief Valves
Pages 2 & 3

"2C" thru "10C" Series Check Valves
Pages 4 & 5

"R" Series Safety Relief Valves
Page 7



"CA" Series Adjustable
In-Line Relief Valves
Pages 4 & 5



"12C" thru "16C" Series Check Valves
Page 6

"P" Series Purge Valves
Page 8

SERIES	SEE PAGE	CHECK	IN-LINE RELIEF	POPOFF TO ATMOSPHERE RELIEF	ADJUST-ABLE	LIQUIDS	GASES	END CONNECTIONS
"C"	4-5-6	X	X	X		X	X	SWAGELOK, Male & Female NPT
"CA"	4-5	X	X	X	X	X	X	SWAGELOK, Male & Female NPT
"CP"	2	X	X	X		X	X	Male to Female NPT Male to Male NPT
"CPA"	3	X	X	X	X	X	X	Male NPT only
"R"	7			X	X	X	X	SWAGELOK, Male & Female NPT
53 56 59	6	X				X		SWAGELOK, Female NPT
"P"	8	MANUAL PURGE OR VENT				X	X	SWAGELOK, Straight tube, Male & Female NPT

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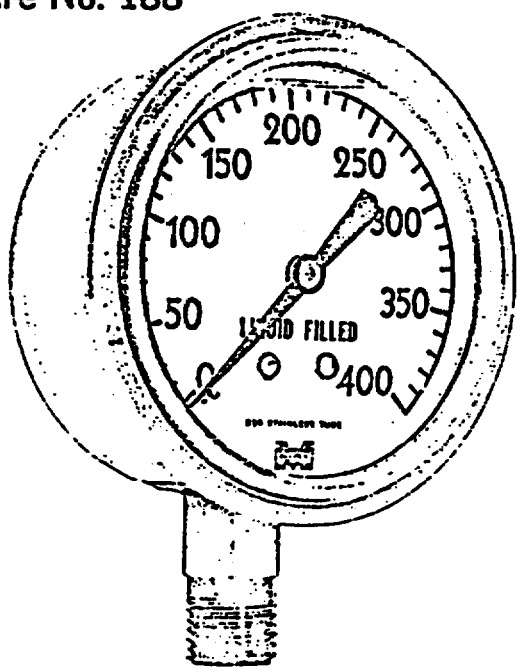
WATER JACKET EXPANSION TANK PRESSURE GAUGE

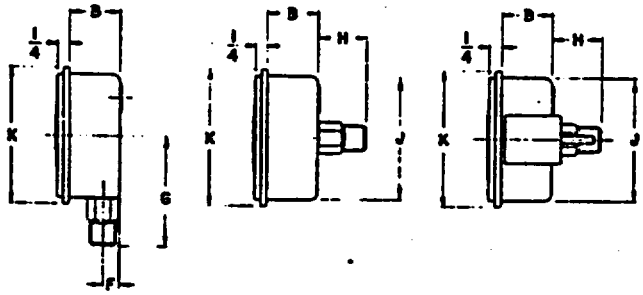
1. Pressure Gauge type: Marshalltown Cat. No. 188
2. Case material: Stainless Steel Type 304
3. Bourdon tube material: Stainless Steel
4. Dial Size: 2-1/2" Dia.
5. Pressure range: 0-400 psi
6. Connection: 1/4" NPT Lower Connection

See Pages VI-18 for Manufacturers Specifications

Marshalltown

Figure No. 188





	K	B	H	G	J	F
Fig. 188	2-55/64	1-11/64		2-9/32		11/32
Fig. 188C	2-55/64	1-11/64	1		2-19/32	
Fig. 188B	2-55/64	1-11/64	1		2-19/32	

SPECIAL FEATURES
Waterproof, weatherproof, hermetically sealed, corrosion resistant, vibration/pulsation proof.

APPLICATION
For severe application requiring a hermetically sealed weatherproof, corrosive service gauge where pulsation/vibration is a problem. Stainless steel case and socket with crystal provide external protection, with internal wetted parts constructed of 316 stainless steel for corrosive service. Shock absorbing polyglide movement plus liquid filling dampen and absorb effect of pulsation/vibration for maximum performance and long service life in all types of severe field and factory applications.

INTERNAL FEATURES
Accuracy — Within 2% of total scale range in middle half of scale, 3% elsewhere.
Bourdon Tube — 316 stainless steel.
Movement — Polyglide movement has nickel-silver working parts and polycarbonate mounting plates for maximum smoothness and shock absorption, providing exceptionally long service life.
Liquid Fill — Viscous lubricant constantly bathes intern and dampens effect of vibration and pulsation dramati extending service life.

EXTERNAL FEATURES
Dial Size — 2-1/2"
Case — 304 stainless steel.
Ring and Crystal — Unbreakable polycarbonate crystal and 304 stainless steel ring gasketed and spun on to case for positive leakproof seal.
Dial and Pointer — Aluminum with baked enamel finish.
Socket — 316 stainless steel, with internal "O" ring for positive leakproof seal to case.
Checkscrew — Standard with .016 orifice checkscrew for oil.
Safety Blowout Plug — Standard, calibrated to blow out at less than 10 psi to vent internal pressure through back of case in the event of tube failure.

CASE VARIATIONS
Fig. 188 — 316 stainless steel 1/4" male bottom connection is standard.
Fig. 188C — 316 stainless steel 1/4" male center back connection is standard.
Fig. 188B — 300 series stainless steel studs and U-clamps for panel mounting. 316 stainless steel 1/4" male center back connection is standard.

STANDARD RANGES

	Pounds Per Square Inch						
Total Graduation	30	60	100	160	200	300	400
Figure Intervals	5	10	10	20	20	50	50
Smallest Sub-Division	1	2	2	5	5	10	10

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FUEL TANK IMMERSION HEATER

1. Heater type: CHROMOLOX Cat. No. AR MT-2155-E2-240V.

See Page VI-21 for Manufacturers Specifications.

TYPES ARMT & ARMTO

2" & 2½" screw plug—copper & steel sheaths

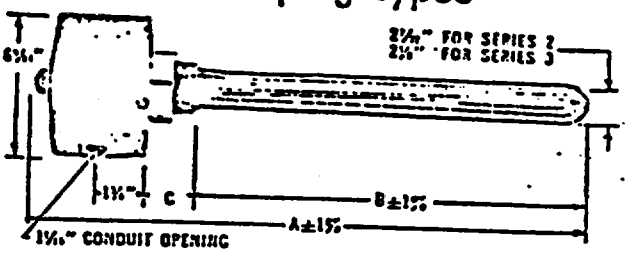
IMMERSION ELEMENTS
screw-plug types

Type ARMT—copper sheath—for heating water and water solutions—has integral thermostat with 50° to 250°F range. Elements and Thermowell are brazed to brass screw plug.

Type ARMTO—steel sheath—for heating of oils—has 150° to 550°F thermostat range. Elements and Thermowell are welded to steel screw plugs.

Temperature control is by means of an adjusting knob on the red enameled steel housing. Connections can be made with standard BX wiring.

Built-in thermostat is connected as a line thermostat for elements up to 6000w on 240v single phase and 4200w on 120v. Thermostat is not wired to elements for all 3-phase and 480v ratings. For larger ratings, 3-ph, 480v and dc circuits, thermostat must be wired to operate holding coil of suitable contactor. For contactors, see page 93.



Temperature Ranges Available:
- T-1— 0-100°F
- T-2— 50-250°F
- T-3—150-550°F

E-1 Housing: Standard sheet metal.
E-2 Housing: Moisture and Explosion Resistant.
Installation: See page 35.

DIMENSIONS—Inches

Overall Length A	Length Inside Tank B	C	KW	Catalog Number	Voltages Available	Temperature Ranges Available °F	Housings Available	Approx. Net Wt. Lbs.
Type ARMT—copper sheath—single phase—2" screw plug—45 watts per sq. in.—2 elements								
19	12	1⅞	1.5	ARMT-2155*	120, 240	T1, T2	E1, E2	6.25
15	8	1⅞	2	ARMT-2205-3	120, 240, 480	T1, T2	E1, E2	5.0
16½	9 9/16	1⅞	2.5	ARMT-2255	240, 480	T1, T2	E1, E2	6.3
19	12	1⅞	3	ARMT-2305	120, 240, 480	T1, T2	E1, E2	6.25
25	18	1⅞	4	ARMT-2405	240, 480	T1, T2	E1, E2	7.0
26 13/16	19⅛	1⅞	5	ARMT-2505	240, 480	T1, T2	E1, E2	7.5
32 3/16	25¼	1⅞	6	ARMT-2605	240, 480	T1, T2	E1, E2	10.0
35 11/16	28¼	1⅞	7	ARMT-2705	240, 480	T1, T2	E1, E2	12.0
47 7/16	40⅝	1⅞	10	ARMT-2105	240, 480	T1, T2	E1, E2	15.0
Type ARMTO—steel sheath—single phase—2" screw plug—20 watts per sq. in.—2 elements								
18 15/16	11⅞	2	1.5	ARMTO-2155	120, 240, 480	T1, T2, T3	E1, E2	6.0
25	17⅞	2	2	ARMTO-2205	120, 240, 480	T1, T2, T3	E1, E2	7.0
26 13/16	19⅞	2	2.5	ARMTO-2255	120, 240, 480	T1, T2, T3	E1, E2	7.5
32 9/16	25	2	3	ARMTO-2305	120, 240, 480	T1, T2, T3	E1, E2	7
40 9/16	33	2	4	ARMTO-2405	120, 240, 480	T1, T2, T3	E1, E2	7
47 7/16	40½	2	5	ARMTO-2505	120, 240, 480	T1, T2, T3	E1, E2	9
55 1/16	48	2	6	ARMTO-2605	240, 480	T1, T2, T3	E1, E2	10
Type ARMT—copper sheath—single & 3 phase—2½" screw plug—45 watts per sq. in.—3 elements								
14⅞	7⅞	2½	3	ARMT-3305	240	T1, T2	E1, E2	8
16⅞	8⅞	2½	3.75	ARMT-3375	240	T1, T2	E1, E2	9
18⅞	11⅞	2½	4.5	ARMT-3455	240, 480	T1, T2	E1, E2	9.25
24⅞	17¾	2½	6	ARMT-3605	240, 480	T1, T2	E1, E2	12
26⅞	19⅞	2½	7.5	ARMT-3755	240, 480	T1, T2	E1, E2	13
32	24½	2½	9	ARMT-3905	240, 480	T1, T2	E1, E2	14
Type ARMTO—steel sheath—single & 3 phase—2½" screw plug—20 watts per sq. in.—3 elements								
25	17½	2½	3	ARMTO-3305	240, 480	T1, T2, T3	E1, E2	8
26⅞	20	2½	3.75	ARMTO-3375	240, 480	T1, T2, T3	E1, E2	8.75
32⅞	24⅞	2½	4.5	ARMTO-3455	240, 480	T1, T2, T3	E1, E2	9.25
40	32⅞	2½	6	ARMTO-3605	240, 480	T1, T2, T3	E1, E2	12
47½	40	2½	7.5	ARMTO-3755	240, 480	T1, T2, T3	E1, E2	13
55	47½	2½	9	ARMTO-3905	240, 480	T1, T2, T3	E1, E2	14

* Single element

See Price Supplement for pricing and Product Code Numbers (PCN) for each assembled heater.

Specify: Quantity, Catalog No., Volts, Watts, Housing, Immersion Heaters.

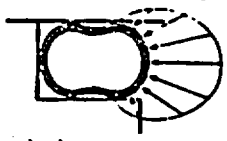
United Metallic O-Rings

Metallic O-rings are made of metal tubing which is formed into circular or other required shapes, the two ends resistance butt-welded together, and the weld ground flush. Welding is sometimes done by tungsten inert gas or plasma arc, in vacuum or inert gas environments.

The typical application places a metallic O-ring in axial compression between parallel faces which are square to the fluid passage or vessel axis. The seal is usually located in a counterbore or closed groove in one face, or in a retainer, which eliminates the need for machining a groove. Other seal or groove configurations may be used.

Upon compression to a predetermined fixed height, the seal tubing buckles slightly, resulting in two contact areas on seal face, and maximum contact stress between the seal and the mating faces. The seal accommodates some deviation from flatness and parallelism of the sealed faces.

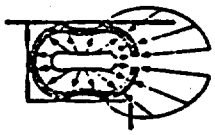
Types of Metallic O-Rings



Plain (Not Pressurized, Not Self-Energized) Metallic O-Ring

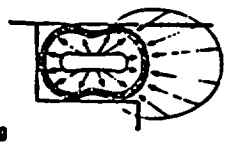
For low to moderate pressure. Most economical type o-ring.

*Solid Wire Rings are available in most materials



Self-Energizing Metallic O-Ring

Inner pheriphery is vented by small holes or large slot, therefore pressure inside ring is same as in the system. Increasing internal pressure increases sealing effectiveness.



Pressure-Filled Metallic O-Ring

Ring is filled with inert gas at about 600 psi 42 kg·cm^2. At elevated temperatures gas pressure increases, offsetting loss of strength in tubing and increasing the sealing stress. This ring cannot support pressures as high as self-energizing ring, but is useful in temperature range of 800 to 2000°F 425 to 1093 C.

Guide to the Selection of UAP Metallic O-Rings

To select the proper Metallic O-Ring for a particular application, it is necessary to determine system temperature, pressure, and fluid to be sealed.

1. O-RING TYPE: Pressure determines if o-ring should be self-energized.

PRESSURE	O-RING TYPE
vacuum to 100 psi 7 kg/cm^2	self-energizing not required
100 psi 7 kg/cm^2 and above	self-energizing desirable

2. O-RING MATERIAL: Temperature determines basic o-ring material.

TEMPERATURE	O-RING MATERIAL
cryogenics to 500°F 260°C	321 stainless steel
to 800°F 427°C	Inconel 600
to 1300°F 687°C	Inconel X-750 ←
above 1300°F 687°C	consult UAP engineering

3. O-RING SIZE: Tubing Diameter is determined by ring O.D., compression force desired, and available space.

OUTSIDE DIA. OF O-RING	O-RING CROSS SECTION
.250 to 4.000 6.35 101.6mm	1/32 0.79mm
.500 to 10.000 12.70 254.0mm	1/16 1.587mm
1.000 to 20.000 25.40 508.0mm	3/32 2.381mm
2.000 to 50.000 50.80 1270.0mm	1/8 3.175mm
over 50.000 1270.00mm	consult UAP engineering

Seal load vs. seal ring diameter to various tubing outer diameters and wall thickness of 321 stainless steel tubing. For tubing made of Inconel 600, multiply loads shown by 1.1; for Inconel X-750, multiply by 1.4.

For the larger tubing sections at the recommended compression the forces are:

321 S.S.
1/4 x .020 333 lb/in
6,350mm x 0.5080 58.93 kg/cm
1/4 x .035 990 lb/in
6,350mm x 0.8890 176.78 kg/cm
1/4 x .049 2000 lb/in
6,350mm x 1.2446 357.16 kg/cm
3/8 x .049 1360 lb/in
9.525mm x 1.2446 242.87 kg/cm
1/2 x .049 850 lb/in
12,700mm x 1.2446 151.79 kg/cm

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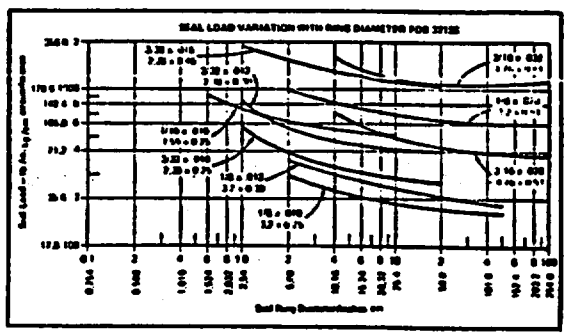
These values are for straight tubing samples. For seals with a seal diameter to tubing diameter ratio of 200 or less allowance should be made for increased compression force over these values.



O-Ring wall thickness: The first requirement is that the tubing be able to supply the force needed to cause its plating to yield.

As a rule of thumb, wall thickness for seals made from 1/8 in. 3,175 mm tubing and smaller should cause yielding of the plating at a load of 400 lb/in 71.4 kg/cm. For tubing over 1/8 in. 3.175 mm dia., 800 lb/in 142.8 kg/cm should be required. However, wall thickness is not important with TFE coatings.

4. O-RING COATING or PLATING:

Unplated Seals: Liquid leakage can be estimated by use of the following expression, which is based on tests at moderate pressure: $Q = 5.0 \times 10^{-6}\ P/u$. Where Q = leakage, cc/sec; P = pressure difference, psi kg/cm^2; and u = liquid viscosity at operating conditions, centipoise.

If the resulting calculated leakage is in the range of 10^{-3} to 10^{-4} or less, actual leakage may be zero because of surface tension. If leakage occurs, it should be proportional to seal diameter, and the above expression multiplied by $D/2$; where D = seal diameter, in. Actual leakage will most probably be less than predicted.

Plated Seals: Helium-leaktight joints will result from proper seal selection and application. Test results vary from 10^{-6} to 10^{-9} cc/sec and lower at one-atmosphere pressure differential.

Cryogenic to 500°F 260°C – Teflon
to 1300°F 687°C – Silver
to 2200°F 1126°C – Nickel

See back cover for other coatings or consult UAP engineering.

Section VII

SECTION VII

WEIGHT CALCULATIONS

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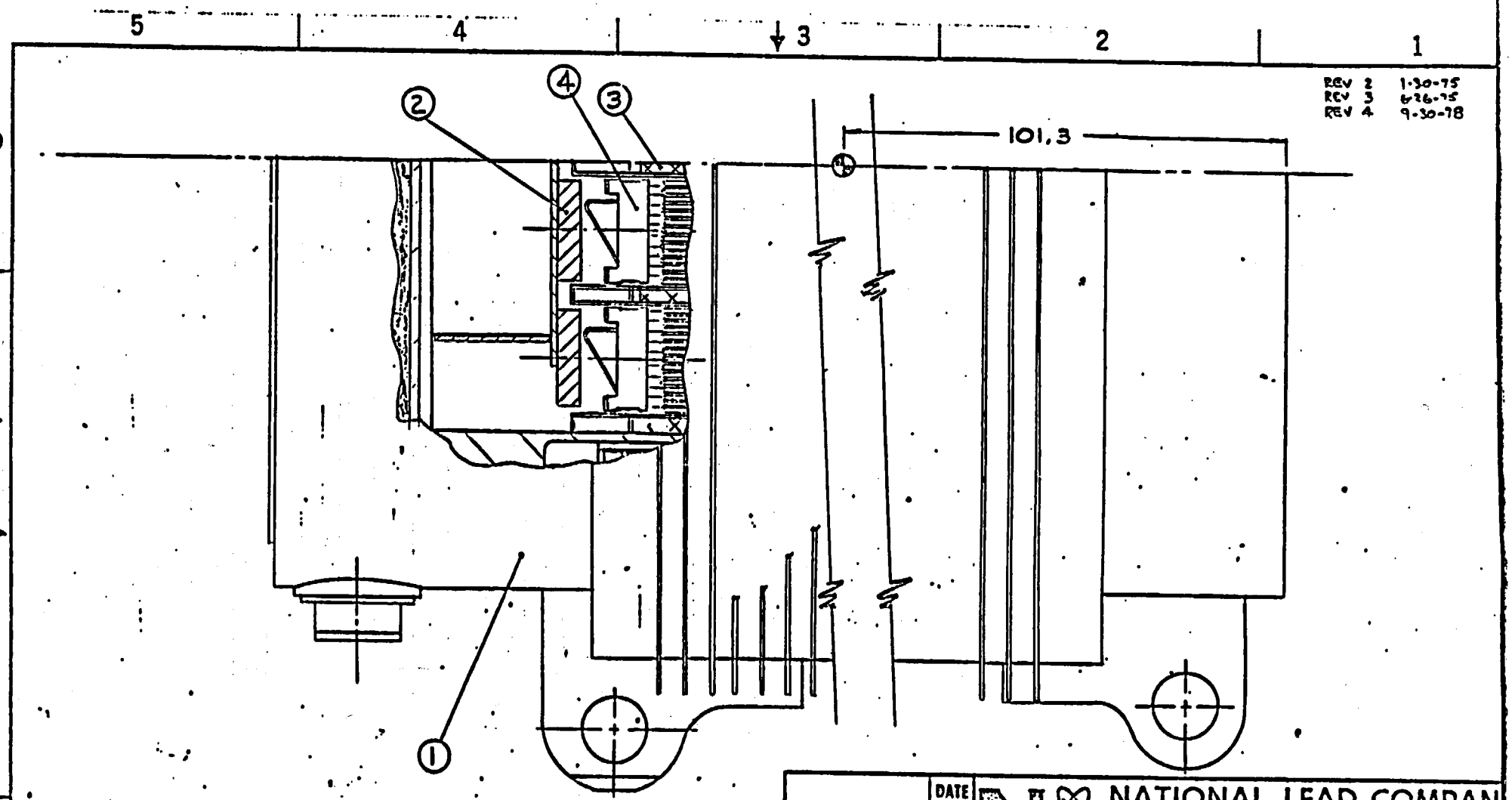

REV 2 1-30-75
REV 3 6-26-75
REV 4 9-30-78

	DATE
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CHKD.	
ENG	
Q.C.	
PROD	

NATIONAL LEAD COMPAN
NUCLEAR DIVISION
WILMINGTON PLANT

10/24 RAILCASK

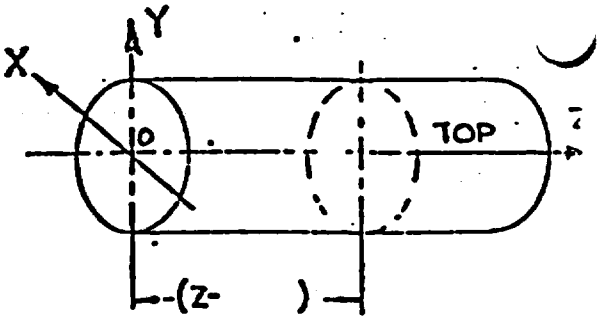
WEIGHT & C.G.

CODE IDENT	PROD Nº	SIZE	DRAWING Nº	R
29932	A-5325	B	20120B	

NL INDUSTRIES, INC. NUCLEAR DIVISION
WILMINGTON, DELAWARE

WEIGHTS & BALANCES

Drawing No. 20120 B



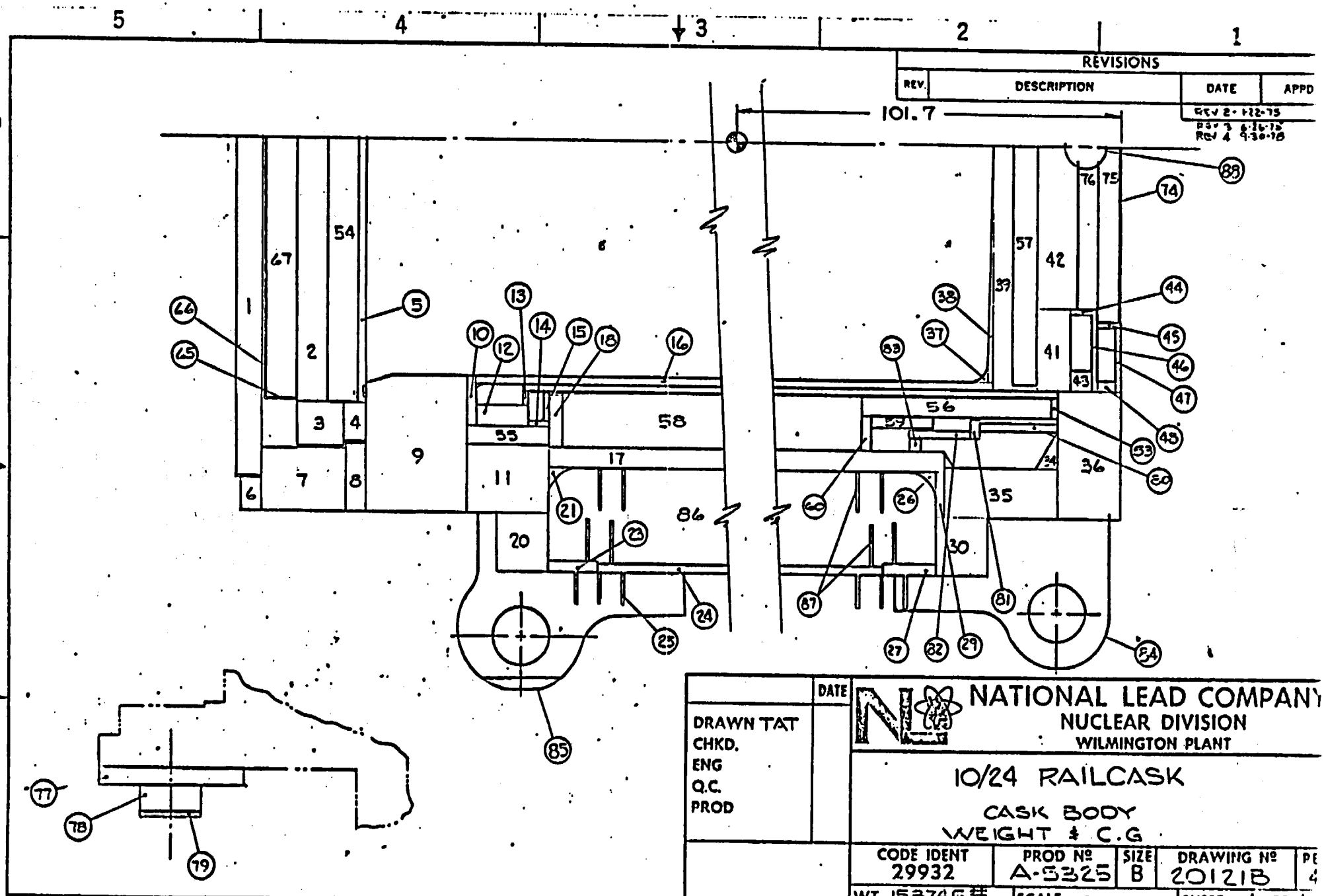
ORIGIN (O) AT

10/24 CASK ASSEMBLY

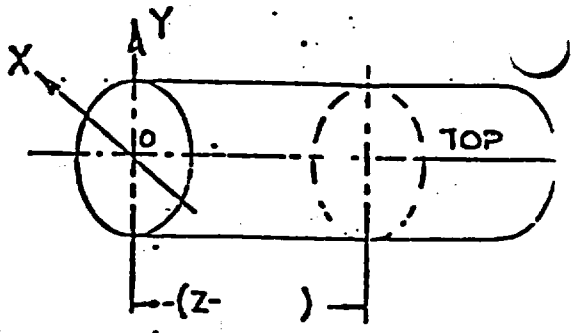
Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
1	CASK BODY	160111			101.81			16287
2	BWR SPACER	303.4			187.18			56790
3	BWR BASKET	15761			93.215			1469162
4	BWR FUEL	18000			103.09			1855620
	Σ	194175						19669502
CENTER OF GRAVITY			Σ5/Σ1	Σ6/Σ1	Σ7/Σ1			
					101.30			

REV. 4 9/78

DATE _____ PAGE _____ OF _____


REVISIONS
REV.
DESCRIPTION
DATE
APPD
Rev 2 - 1-22-75
Rev 3 6-16-75
Rev 4 9-30-78
101.7
DRAWN TAT
CHKD.
ENG
Q.C.
PROD
DATE
NATIONAL LEAD COMPANY
NUCLEAR DIVISION
WILMINGTON PLANT
10/24 RAILCASK
CASK BODY
WEIGHT & C.G
CODE IDENT
29932
PROD Nº
A-5325
SIZE
B
DRAWING Nº
20121B
WT 153745#
SCALE ~
SHEET 1 OF 1

Drawing No. 20121 B



ORIGIN (O) AT

CASK BODY AND CLOSURE HEADS

Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
1		2328			203.25			47317
2		1715			197.25			338283.75
3		653			196.375			128232.88
4		313			192.9375			60389.44
5		398			192.375			76565.25
6		380			203			77105.49
7		2837			197.937			561576.9
8		692			192.937			133441.0
9		7181			186.625			1340154.1
10		53			181.125			9612.30
11		2721			177.625			483402.8
12		470			178.5			83902.14
13		43.			176.25			7652.77
14		74			174.985			12940.14
15		79			173.5			13774.16
16		5742			96.75			555567.5
	Σ							
CENTER OF GRAVITY			Σ5 / Σ1	Σ6 / Σ1	Σ7 / Σ1			

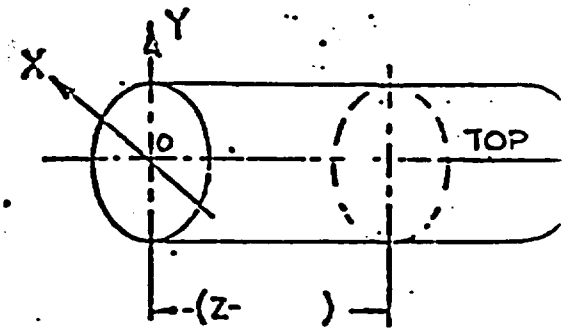
Rev. 3 - 9/78

DATE _____ PAGE _____ OF _____

NL INDUSTRIES, INC. NUCLEAR DIVISION
WILMINGTON, DELAWARE

WEIGHTS & BALANCES

Drawing No. _____



ORIGIN (O) AT

Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z −	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
17		16992			95.578			162406
18		353			172.625			60857.
19								
20		1130			175.5			198343
21		110			172.5			19035.
22								
23		725			169			122563
24		7407			96.125			711959.
25		3907			96.125			375560
26		110			18.687			2062.1
27		725			23.25			16861.5
28								
29		278			17.703			4917.9
30		1130			16.75			18930.
31								
32								
	Σ							
CENTER OF GRAVITY			Σ5 / Σ1	Σ6 / Σ1	Σ7 / Σ1			

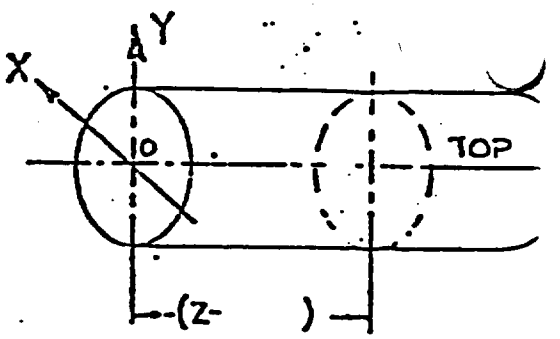
Rev. 3 6/75

DATE _____ PAGE _____ OF _____

NL INDUSTRIES, INC. NUCLEAR DIVISION
WILMINGTON, DELAWARE

WEIGHTS & BALANCES

Drawing No. _____



ORIGIN (O) AT _____

Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
33		56			16.791			940
34		200			6.916			1385.4
35		2912			11.578			33710.1
36		3866			3			11598.7
37		34			13.250			450.5
38		113			12.687			1436.8
39		968			11.5			11131.77
40		52			9.25			476.09
41		874			6.375			5570.9
42		860			6.00			5162.58
43		215			3.75			807.9
44		35			3.50			123.375
45		35			1.125			38.81
46		127			2.5			318.475
47		107			.250			26.712
48		45			1.375			61.7
	Σ							
CENTER OF GRAVITY			ΣS / ΣI	ΣG / ΣI	ΣZ / ΣI			

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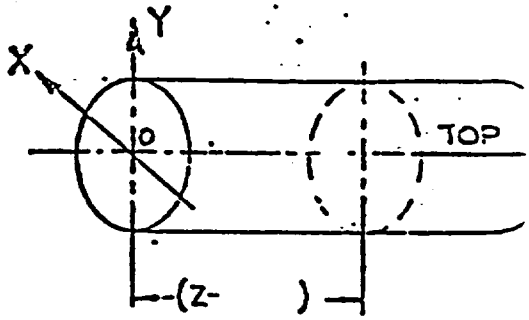
DATE _____ PAGE _____ OF _____

Drawing No. _____



ORIGIN (O) AT _____

Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
49								
50								
51								
52								
53		91			6.375			579.5
54		4077			194.25			791957.
55		1690			177.25			299506
56		5176			16.25			84103.
57		2776			9.250			2568
58		58460			98.875			5780232.
59		410			21.75			8908.
60		231			25.25			5839.
61					200.437			
62					200.437			
63					193.937			
64					200.5			
	Σ							
CENTER OF GRAVITY			Σ5 / Σ1	Σ6 / Σ1	Σ7 / Σ1			

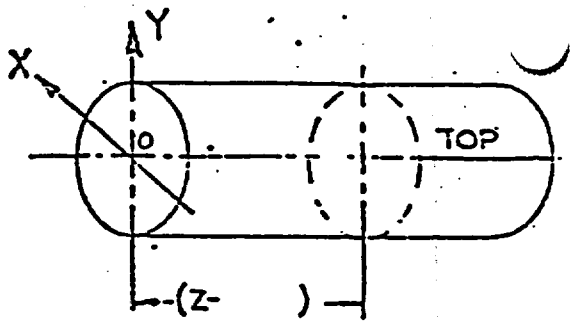
Rev. 3 - 9/78

DATE _____ PAGE _____ OF _____

NL INDUSTRIES, INC. NUCLEAR DIVISION
WILMINGTON, DELAWARE

WEIGHTS & BALANCES

Drawing No. _____



ORIGIN (O) AT _____

Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
65		17			199.312			3477
66		63			201.812			12714.16
67		158			200.25			31639.5
68								
69								
70								
71								
72								
73								
74		31			.062			1.9468
75		54			1.187			64.1
76		39			3.125			121.88
77		160			197.00			31520
78		126			197.00			24822
79		38			197.00			7441.23
80		227			9.8125			2227.43
	Σ							
CENTER OF GRAVITY			Σ5 / Σ1	Σ6 / Σ1	Σ7 / Σ1			

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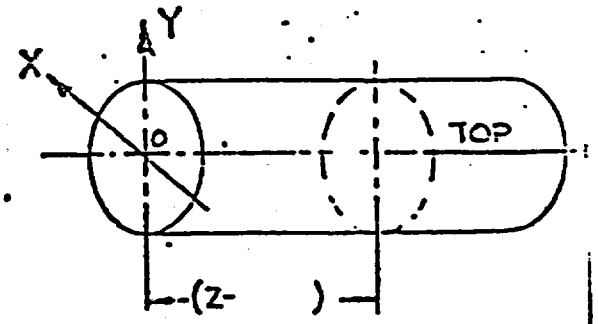
DATE _____ PAGE _____ OF _____

NL INDUSTRIES, INC. NUCLEAR DIVISION
WILMINGTON, DELAWARE

WEIGHTS & BALANCES

Drawing No. _____



ORIGIN (O) AT

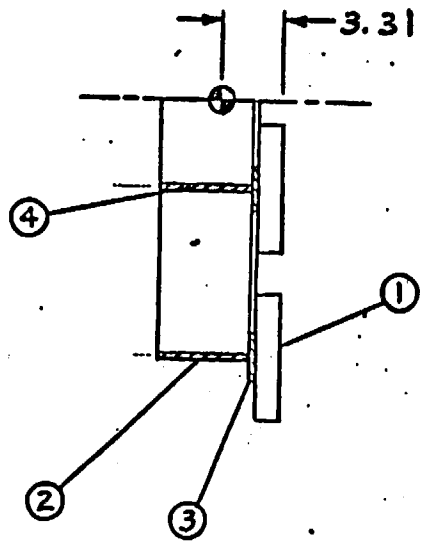
Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
81		84.			14.125			1188.76
82		221			17.625			3888.07
83		86.			20.062			1727.58
84		250			9.0			2250
85		125			172			21500
86		11238			96.125			1080252.8
87		5012			96.125			481778.5
88		226			1.75			395.5
	Σ	160111						16287990
CENTER OF GRAVITY			Σ5 / Σ1	Σ6 / Σ1	Σ7 / Σ1			
					101.73			

Rev. 4 - 9/78

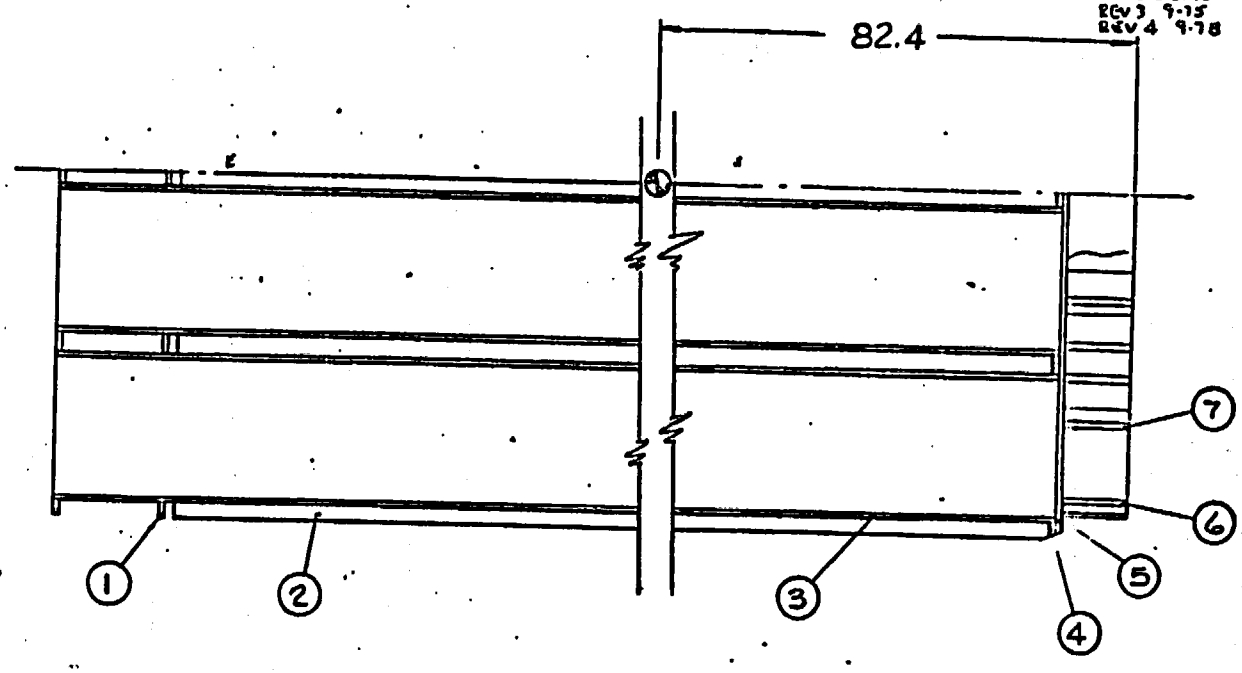
DATE _____ PART _____ OF _____

REVISIONS			
REV.	DESCRIPTION	DATE	APPD

Rev 1 1-24-75
Rev 2 6-21-75
Rev 3 9-75
Rev 4 9-78



SPACER
199 #



FUEL BASKET
15716 #

	DATE
DRAWN TAT	4/11/78
CHKD.	
ENG	
Q.C.	
PROD	

NATIONAL LEAD COMPANY
NUCLEAR DIVISION
WILMINGTON PLANT

10/24 RAILCASK
PWR SPACER & FUEL BASKET
WEIGHTS & C.G'S

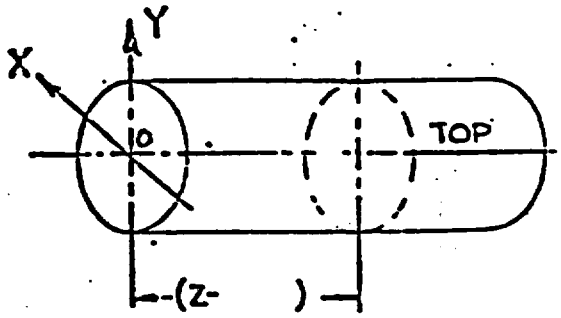
CODE IDENT	PROD Nº	SIZE	DRAWING Nº	REV
29932	A-5325	B	2122B	4

WT ~ SCALE ~

NL INDUSTRIES, INC. NUCLEAR DIVISION
WILMINGTON, DELAWARE

WEIGHTS & BALANCES

Drawing No. 20122 B



ORIGIN (O) AT

PWR FUEL BASKET

Item	Dwg. No.	1	2 -	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
1		70.96		161.437			11455.56	
2		78.00		82.531			6437.4	
3		7701.01		82.953			638821.86	
4		25		4.312			107.8	
5		68		3.875			263.5	
6		10.4		1.8125			18.85	
7		40.6		1.8125		-	73.59	
	Σ	15715.97					1294482.18	
CENTER OF GRAVITY			Σ5 / Σ1	Σ6 / Σ1	Σ7 / Σ1			
				82.37				

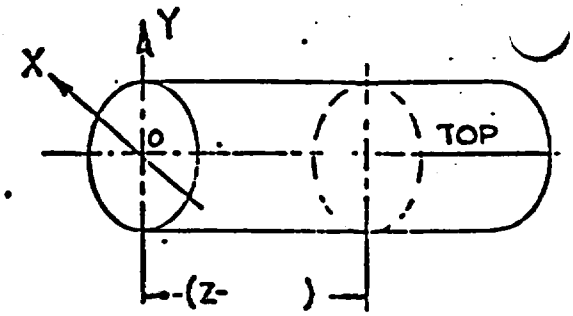
Rev. 4 9/78

DATE _____ PAGE ___ OF ___

NL INDUSTRIES, INC. NUCLEAR DIVISION
WILMINGTON, DELAWARE

WEIGHTS & BALANCES

Drawing No. 20122 B



ORIGIN (O) AT

PWR SPACER

Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
1		125			1.25			156.
2		41.8			8.31			347.36
3		19.0			2.38			45.22
4		13.2			8.31			109.69
	Σ	199						658.52
CENTER OF GRAVITY			ΣS / ΣI	ΣG / ΣI	ΣZ / ΣI			
					3.31	Rev. 3 9/78		

DATE _____ PAGE _____ OF _____

10/24 Rail Cask

Top & Bottom Impact Structure

Weights & C. G's

Drawing No. 20123B Rev. 0

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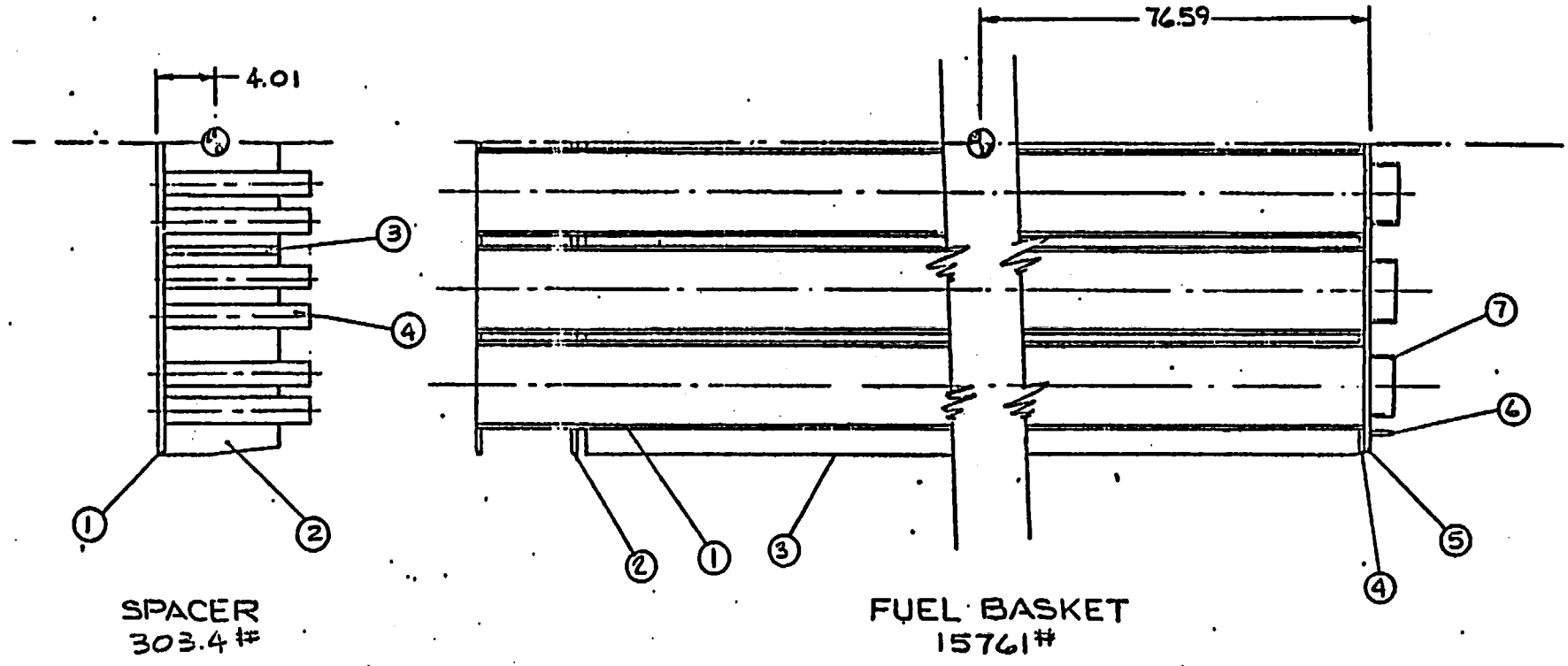
REVISIONS			
REV.	DESCRIPTION	DATE	APPD
REV 1	[illegible]		
REV 2	4/23/85	JEC	



	DATE
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CHKD.	
ENG	
Q.C.	
PROD	

NATIONAL LEAD COMPANY
NUCLEAR DIVISION
WILMINGTON PLANT

10/24 RAILCASK
BWR SPACER & FUEL BASKET
WEIGHTS & C.G's

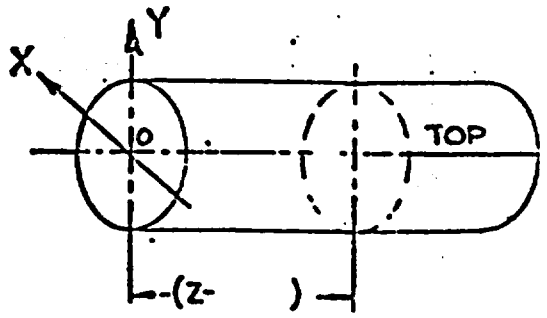
CODE IDENT	PROD Nº	SIZE	DRAWING Nº	R
29932	A-5325	B	20124B	

WT ~ SCALE ~ SHEET 1 OF 1

NL INDUSTRIES, INC. NUCLEAR DIVISION
WILMINGTON, DELAWARE

WEIGHTS & BALANCES

Drawing No. 2C124B



ORIGIN (O) AT

BWR FUEL BASKET

Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
1		5551.98			77.812			435348.0
2		69.20			153.312			10609.1
3		9557.23			77.50			740685.3
4		69.20			2.312			159.99
5		145.29			1.937			281.22
6		19.80			1.062			21.02
7		48.51			1.062			51.51
	Σ	15761.13						1207156.4
CENTER OF GRAVITY			ΣS / ΣI	ΣG / ΣI	ΣZ / ΣI			
					-3.59			

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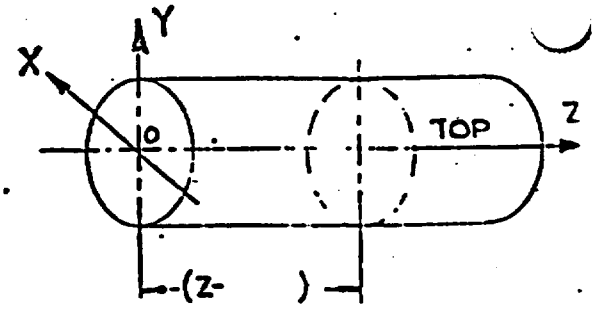
NL INDUSTRIES, INC. NUCLEAR DIVISION
WILMINGTON, DELAWARE

WEIGHTS & BALANCES

Drawing No. 20124B



ORIGIN (O) AT

BWR SPACER

Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z -	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
1		76			.25			19.0
2		55			4.7			258.5
3		50.4			4.7			236.88
4		122			5.775			704.55
	Σ	303.4						1218.93
CENTER OF GRAVITY			Σ5/Σ1	Σ6/Σ1	Σ7/Σ1			
					4.01			

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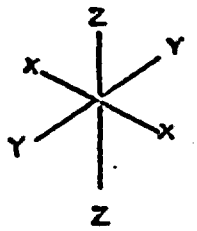
NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

WEIGHTS BALANCES

Drawing No. _____



Coordinate Designation

Item	Dwg. No.	1	2	3	4	5	6	7
		W	X	Y	Z	Wx	Wy	Wz
		Lb.	In.	In.	In.	In. Lb.	In. Lb.	In. Lb.
	Σ							
Center Of. Gravity			Σ5 / Σ1	Σ6 / Σ1	Σ7 / Σ1			

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Section VIII

SECTION VIII

THERMAL ANALYSIS

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Section VIII

THERMAL ANALYSIS

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Section VIII

Thermal Analysis

1.0 Summary

This section summarizes the thermal analyses which were performed to demonstrate fulfillment of the thermal capability requirements established by AEC regulations, Title 10 CFR Part 71.

The shipping cask is designed to safely contain irradiated fuel under a variety of normal conditions (71.35 and Appendix A of 10 CFR 71) and accident conditions (Appendix B of 10 CFR 71). In order to verify the adequacy of the design, a detailed analysis of the reference design for shipment of 10 PWR elements was performed considering extreme normal operating conditions and hypothetical fire conditions in conjunction with appropriate postulated cask interactions. Because the decay heat load for the PWR and BWR cases is the same, a separate analysis was not performed for the BWR case. It is evident that for both PWR and BWR cases the cask body temperatures will be equal and a calculation was performed to determine the maximum BWR basket temperature (Appendix B). Other modes of operation such as forced convection in the cooling system, natural convection in the cooling system and cold weather performance were evaluated on a qualitative basis.

1.1 Normal Conditions of Transport

During hot operation the ambient temperature is 130°F. The air is still and direct solar heating has an effective intensity of 92 BTU/hr-ft^2 on all exposed surfaces of the cask. The average design decay heat load is 70 kw (238910 BTU/hr) for both PWR and BWR shipments. As axial peaking factor of 1.1 is applied to this heat load to establish the design maximum local heating rate in the cask.

Decay heat is removed from the fuel to an aluminum basket by thermal radiation and conduction through helium and by conduction through the neutron absorber liners. After conduction to the periphery of the basket, the decay heat is then transferred to the cask through a helium gap by conduction and thermal radiation and by conduction and convection through the cask body to the atmosphere. The cooling channels are dry.

This mode of heat dissipation is considered as the normal mode of operation for the purposes of determining metal temperatures and pressures for the structural evaluation of the cask.

Forced circulation in the auxiliary cooling system is the actual normal mode of operation even though the normal condition of transport analysis considers the cooling system not to be operating. The cask temperatures will be lower when the cooling system is operating. This will increase the safety of the package above what is required for normal conditions of transport.

The auxiliary cooling system is provided to circulate coolant through cooling channels that are located in the cask at the outer surface of the inner shell.

The auxiliary cooling system has the capacity to dissipate more than 100% of the decay heat. In operation, not all of the heat is removed by the cooling system. A portion of the decay heat load is removed by a combination of conduction, convection and radiation to the atmosphere.

During cold operation the ambient temperature is -40°F in still air without any solar heat load (the cask is in the shade). Detailed analyses of the cask have not been performed under these conditions since a uniform cask temperature of -40°F may be conservatively assumed. The neutron shield will be protected from freezing to -40°F by using an antifreeze solution.

1.2 Off Normal Conditions

The following off normal conditions have been evaluated:

a). A hypothetical fire accident with conditions applied sequentially as specified in Appendix B of 10 CFR 71. In some areas additional conservative assumptions were involved.

b). Forced convection and loss of forced convection are evaluated qualitatively.

1.2.1 Hypothetical Accident Conditions

Beginning with an assumed ambient temperature of 100°F, the cask experiences a 30 foot drop onto a flat surface followed by a puncture.

This is assumed to result in the complete and instantaneous loss of all neutron shield water as well as all auxiliary cooling system water. Although the auxiliary cooling system is severed from the cask, the neutron shield jacket and fins are assumed not removed because of crash barrier effectiveness. Concurrent with the instantaneous loss of water, the cask is exposed to a thermal radiation environment of 1475°F for 30 minutes with an emissivity of 0.9 and a cask surface absorption coefficient of 0.8. Convection heat transfer at the outer surface has been conservatively considered even though regulations do not specify convection heat transfer during the fire.

After the fire the cask is cooled naturally until equilibrium conditions are reached in ambient air at 130°F with the full design solar heat load. The cask surface emissivities after the fire are conservatively assumed to be the same as before the fire. Heat transfer within the cask occurs only by a combination of conduction, radiation and gaseous natural convection. The auxiliary cooling system ducts are assumed completely inoperable. No solar heat load is applied during the fire. The decay heat load throughout the entire transient is as specified under normal conditions.

1.2.2 Loss of Forced Convection Conditions

When the auxiliary cooling system loses its forced circulation capability, possibly due to a power failure, the entire system becomes a natural circulation system. Whereas during normal operation, half of the cask ducts are in forced circulation with the other half assumed passively wet, all ducts are now essentially operating at a higher temperature under natural circulation. Except for the loss of power the cask and the auxiliary cooling system are assumed to be in the same condition as under normal transport.

1.3 Results

Normal condition of transport without auxiliary cooling at full heat load was analyzed using the ANSYS computer program. The actual shipment of the cask will be done with the cooling system operating.

Steady state and transient analyses were performed on two dimensional models of the central portion of the cask assuming negligible end effects. This covers the hottest portion of the cask. Detailed analyses of the cask ends are not justified because upper bounds on temperatures may be established on the basis of the above analyses and because design features at the cask ends are not thermally limiting.

The pin-to-pin temperature distribution is presented for the post-fire

and normal condition PWR and BWR loadings. The post fire condition gives the highest fuel pin temperatures and this detailed analysis shows that the maximum pin temperature will not be excessive. The temperatures applicable to the central portion of the cask are based on a maximum local heat load which includes an axial peaking factor of 1.1. The peaking factor is based on the maximum burnup distribution curve of Figure III-2. This is a conservative representation of the axial specific power distribution applicable to decay heat calculation.

A summary of the results of the cask thermal analysis is provided in Table VIII-1 for the PWR and BWR design loading. The following general conclusions are based on these results:

a.) The BWR and PWR loadings have approximately the same temperature distribution.

b.) Under normal conditions of transport without auxiliary cooling, the highest average fuel temperature is 833°F which occurs in the hottest PWR fuel assembly.

c.) The maximum fuel pin temperature following the hypothetical fire is 991°F for the PWR fuel and 854°F for the BWR fuel. Average temperatures will be approximately 938°F for the PWR fuel and 805°F for the BWR fuel.

d.) With the exception of the cask surface, the highest temperatures are reached not during the fire transient but under steady-state conditions following the fire. Steady-state conditions are reached approximately 75 hours after the start of the accident.

e.) The aluminum basket will not melt under any conditions.

f.) The lead shielding will not melt either during the fire or under post fire steady state condition.

g.) With a complete loss of auxiliary coolant (normal conditions of transport) the lead does not melt and the neutron shield water reaches an average temperature of 340°F corresponding to a saturation pressure of 103.3 psig.

h.) Under post-fire steady-state conditions the cask surface will not exceed a temperature of 324°F.

i.) The outer head gasket is estimated to have a temperature no higher than 335°F under post-fire steady-state conditions and a maximum temperature of 1190°F at the end of the fire. The inner head gasket is estimated to have a temperature no higher than 425°F under normal conditions of transport, and a maximum temperature no higher than 615°F under post-fire steady-state conditions.

j.) Cold conditions are conservatively estimated to result in a uniform cask temperature of -40°F. Freezing in the auxiliary cooling system and the neutron shield will be avoided by addition of antifreeze.

TABLE VIII-1

SUMMARY OF RESULTS
CASK TEMPERATURES WITH PWR AND BWR LOADINGS

Location	Post-Fire Steady-State	Without Auxiliary Cooling Steady-State
Outer Surface	324 (1230*)	324
Outer Shell Surface	525	343
Outer Lead	545	364
Inner Lead	599	407
Duct Fluid	599 (dry)	407 (dry)
Inner Shell	609	418
Aluminum (Min.)	627	439
Aluminum (Max.)	732	542
Fuel (Max.) PWR	991	900
Fuel (Avg.) PWR	938	659
Fuel (Max.) BWR	854	724
Fuel (Avg.) BWR	805	659
Outer Seal	approx. 335 (1190*)	approx. 333
Inner Seal	approx. 615	approx. 425

* Maximum, occurs at end of fire.

2.0 THERMAL CONSTANTS

The thermal constants used in the analysis are grouped into basic thermal properties of cask materials and into effective thermal properties which have been derived. The latter consist of natural convection coefficients, simple film coefficients, radiation exchange factors and effective conductivities and heat capacity of composite materials.

2.1 Basic Thermal Properties

Table VIII-2 lists the basic thermal properties of the cask materials such as density, specific heat, conductivity, latent heat of fusion and associated melting temperatures. Surface emissivities are listed in Table VIII-3.

2.2 Effective Thermal Properties

Natural convection coefficients are expressed in terms of equation (1) where T_i and T_o are sink and source temperatures in ^oF.

$$h_{NC} = C\,(T_i - T_o)^b \qquad (1)$$

Ordinary film coefficients, h_f, accounting for simplified gap heat transfer and crud and forced convection coefficients in the cooling ducts, are represented by equation (1) with b equal to zero.

Radiation is considered across all gaps including the ports in the PWR basket. Radiation coefficients are defined by equation (2) where T_i and T_o are sink and

TABLE VIII-2

MATERIAL PROPERTIES

Material	Density (lb/ft³)	Spec. Heat (Btu/lb-°F)	Temp. (°F)	Conductivity (Btu/hr-ft-°F)	Temp. (°F)	Heat of Fusion (Btu/lb)	Melting Pt. (°F)
Stainless Steel (304) Ref. 1	494.0	0.120	32.0	7.74	32.0	—	—
		0.135	752.0	9.43	212.0		
				12.6	932.0		
				15.0	1292.0		
Lead Ref. 1	708.6	0.0305	32.0	20.0	68.0	10.26	621.5
		0.0315	212.0	19.6	208.9		
		0.0338	621.5	18.3	400.1		
		0.0340	621.7	16.9	498.9		
		0.0328	1832.0	14.5	581.0		
				12.1	630.0		
				9.68	717.1		
				9.02	799.9		
				8.71	980.1		
				8.66	1276.0		
Helium Ref. 1 & 2	0.011	1.24	—	0.079	0.0	—	—
				0.100	200.0		
				0.119	400.0		
				0.137	600.0		
				0.152	800.0		
				0.167	1000.0		
				0.197	1400.0		
Aluminum (1180) Ref. 1, 8 & 10	169.0	0.215	32.0	139.6	32.0	170.8	1210.0
		0.270	932.0	131.2	212.0		
		0.303	1220.0	126.8	392.0		
		0.259	1220.2	124.1	572.0		
		0.259	4472.6	122.3	752.0		
				121.0	932.0		
				120.0	1112.0		
				119.5	1210.0		
Fuel (UO_2) Ref. 1	608.8 (PWR)	0.063 (PWR)	—	—	—	—	—
	628.6 (BWR)	0.063 (BWR)	—	—	—	—	—
Fuel (Clad) Ref. 3	408.9	0.0732	—	—	—	—	—
Absorber (Clad) Ref. 1	494.0	0.120	—	10.3	—	—	—
Absorber Ref. 1 & 11	655.0	0.0564	—	41.8	—	—	—
Carbon Steel Ref. 1 & 4	490.0	0.14	—	22.0	—	—	—
Duct Wall Ref. 4	—	—	—	9.5	—	—	—
Duct Fluid (wet) Ref. 4	60.0	1.00	—	—	—	—	—
Duct Fluid (dry) Ref. 4	0.05	0.24	—	0.02	—	—	—
Neutron Shield Fluid (wet) Ref. 1	60.0	1.00	—	—	—	—	—
Neutron Shield Fluid (dry) Ref. 1	0.08053	0.24	—	—	—	—	—

TABLE VIII-3

EMISSIVITIES

Material	ϵ	References
Fuel Pins	0.4	3
Stainless Steel Except as Noted Below	0.5	5,6
Neutron Shield Interior	0.8	5
Aluminum	0.2	6
Cask Exterior Before and After Fire	0.5	5,6
Cask Exterior During Fire	0.8	5, (10 CFR 71)

source temperatures in ^{o}R.

$$h_R = 0.174 \times 10^{-8} F \ (T_i^2 + T_o^2) \ (T_i + T_o) \qquad (2)$$

The values for C, b and F are listed in Table VIII-5. Heat transfer coefficients, h, are in Btu/hr-ft^2 - ^{o}F.

Effective conductivities and heat capacities have been computed for the cooling ducts, the neutron absorber liners around the fuel elements and for single point models of the fuel elements themselves. They are summarized in Table VIII-4.

The semi circular cooling duct walls and duct fluid where forced convection cooling is used have been modeled in the analysis by a single point rectangular block with a conductance of the wall and a heat capacity of the fluid.

The effective conductivity of the absorber liner is based on an equivalent series resistance across the thickness of the liner considering the geometric distribution of stainless steel and absorber shown in Figure X-9. The effective heat capacity of the liner is a volume weighted average heat capacity of stainless steel and the absorber material.

The effective conductivity of the single point fuel element is used in conjunction with a fuel radiation exchange factor. It is obtained by fitting peak temperature results from a detailed fuel element analysis to the single point fuel element model used in the cask analysis. The effective heat capacity is a volume weighted average of active fuel and cladding distributed over the entire fuel element volume. The heat

TABLE VIII-4

EFFECTIVE HEAT TRANSFER FACTORS

Application	C	b	F	pC (Btu/ Ft^3 -°F)	K (Btu/hr-ft°F)
PWR Element (single point)	--	--	0.1626	12.87	0.4374 @881°F 0.5863 @1192°F
Homogenized PWR Absorber Plate	--	--	--	43.3	22.4
Absorber -Aluminum Gap	95.2	--	.1667	--	--
Aluminum-Inner Shell Gap	--	--	0.1667	Helium*	Helium*
Neutron Shield (wet)	65.0	0.3333	--	60.0*	--
Neutron Shield (dry)	0.18	0.3333	0.6925	0.0193*	--
Ducts (dry)	--	--	--	0.012*	0.02
Cask Surface Before & After Fire	0.5599	0.3333	0.8585	--	--
Cask Surface	0.5599	0.3333	0.9843	--	--

* Same as Table VIII-2 for basic properties

TABLE VIII-5

VIEW FACTORS FOR PWR BASKET PORTS

REGION	f_{1-2}	A_1	A_2	F_{1-2}
Rect. Port 1-2	.125	3.15	4.85	.0685
Rect. Port 2-2	.16	4.85	4.85	.0702
Triag. Port 1-2	.08	4.7	3.0	.0439
Triag. Port 3-1	.2	5.0	4.7	.0754
Triag. Port 2-3	.14	3.0	5.0	.0738

capacity of helium as well as inactive pins and pin supports has been conservatively ommitted.

3.0 ANALYTICAL METHODS

The thermal analyses of the cask are based primarily on digital computer program results with hand calculations providing a significant amount of input to the computer programs used.

3.1 Hand Calculations

Hand calculations were performed to establish:

a) The heat transfer coefficients applicable to the finned surface of the cask and the heat generation rate simulating the solar heat load.

b) The heat transfer coefficients applicable to the finned surface of the neutron shield.

c) The radiation exchange factor of the gap between the aluminum basket and the inner shell and neutron absorber sleeves, and across the cutouts in the PWR aluminum basket.

d) The film coefficient of the gap around the neutron absorbers and the effective conductivity and heat capacity of the neutron absorber liners.

e) The effective heat capacity of the single point fuel element model and

its effective radiation exchange factor as well as conductivity and the decay heat generation rate.

The calculations are based mainly on the recommendations in the "Cask Designers Guide" [5] and References 6 and 7.

3.2 Computer Programs

Detailed steady-state transient thermal analysis of the cask were performed with the ANSYS [12] digital computer program. Explicit temperature distributions within the fuel element under dry conditions were calculated using the program WATSON developed by J. S. Watson. [9]

ANSYS was used for the normal conditions of transport (without auxiliary cooling) and fire accident steady state and transient response. The results of this analysis were then used in the ANSYS program to calculate thermal stresses in the aluminum basket.

3.2.1

ANSYS solves a general nonlinear parabolic partial differential equation describing various kinds of potential fields, such as fields of temperature, pressure, electricity and magnetism. ANSYS will also solve for the stress and strain resulting from the imposition of these fields.

Thermal problems may include heat transport by conduction, free and forced convection, radiation, and mass flow. Heat may be produced or absorbed by internal heat sources and sinks and phase changes. Boundary conditions may include insulation, specified surface temperatures or heat fluxes, a heat transfer by radiation, free convection, or forced convection. Thermal properties, heat generation rates, and surface heat transfer coefficients may be tabulated functions of time. Initial conditions may vary with spatial position.

3.2.1.1 Geometric Model

The geometric model analyzed by ANSYS consists of the cask with PWR loading. The geometry is shown in Fig. VIII-1 and the ANSYS element arrangement is shown in Fig. VIII-2. The nodes corresponding to the elements of Fig. VIII-2 are shown in Fig. VIII-3.

Each fuel element has been modeled by a number of triangular elements with a single control node. A 15 mil hot gap between the neutron absorber liner and the aluminum basket and a 20 mil hot gap

between the aluminum basket and the inner shell have been assumed. Heat transfer across the gap is by conduction and radiation through the helium. Perfect contact is assumed at all lead steel interfaces. The cask auxiliary cooling ducts are represented by rectangular elements distributed evenly around the inner shell. Under normal conditions the ducts are considered dry. The fins inside the neutron shield jacket and on the outer surface of the cask have been implicitly represented by appropriate heat transfer coefficients and by assigning the heat capacity of the fins to the water jacket and outer shell. Nominal dimensions given in Fig. VIII-1 have been applied in the ANSYS analytical model.

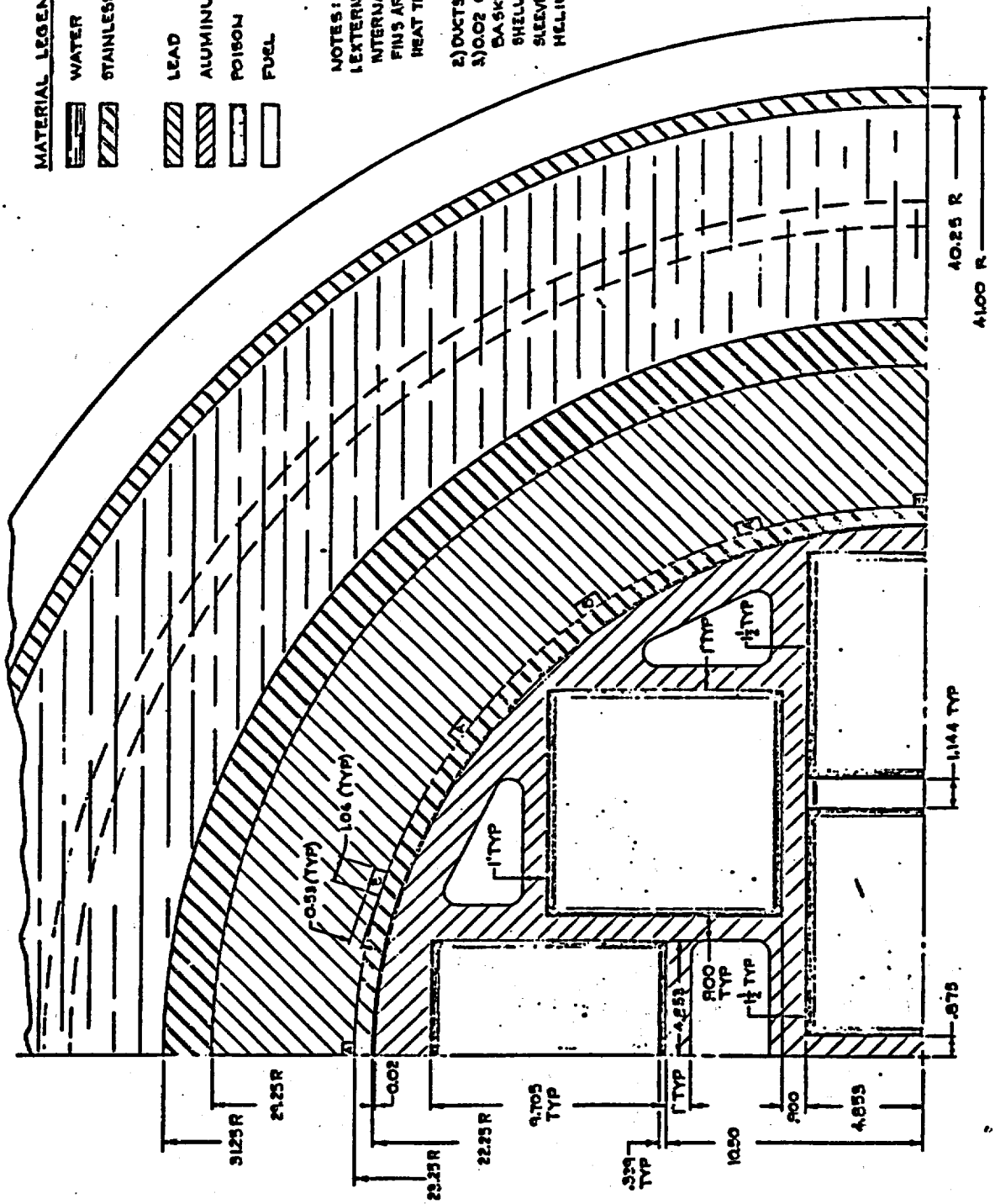


FIGURE VIII-1
THERMAL ANALYSIS
MODEL OF RAIL CASK
WITH PWR LOADING

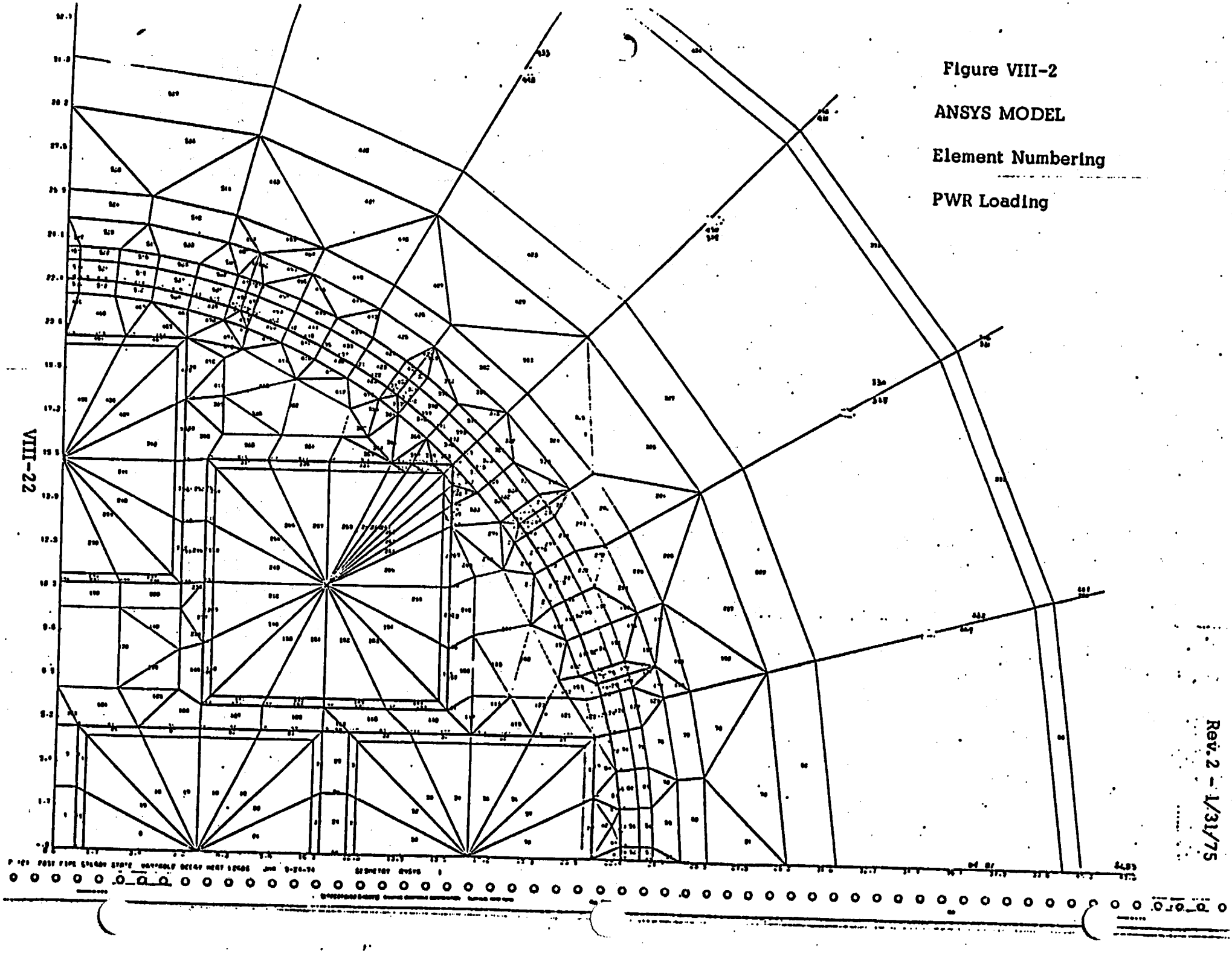


Figure VIII-2

ANSYS MODEL

Element Numbering

PWR Loading

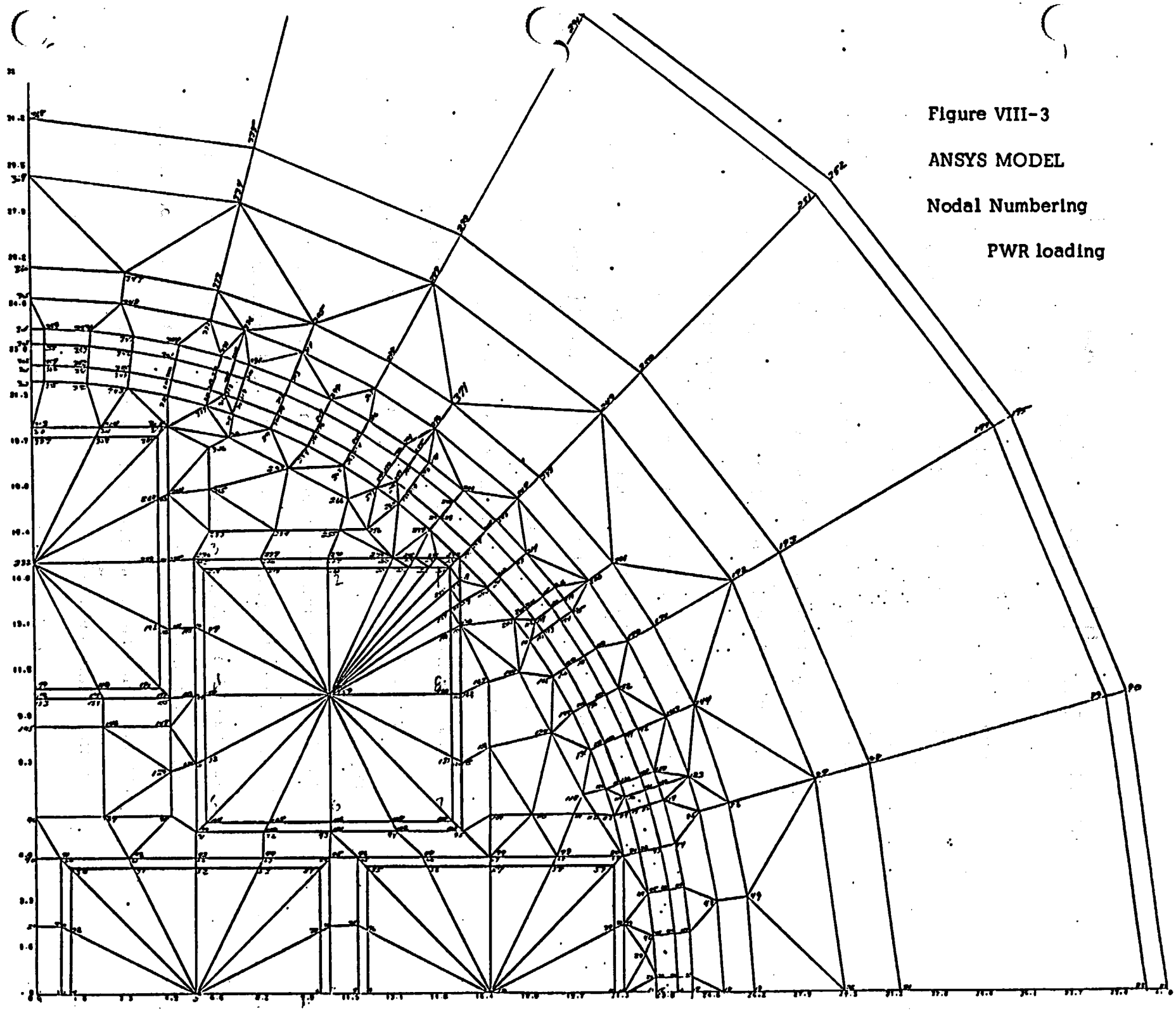


Figure VIII-3

ANSYS MODEL

Nodal Numbering

PWR loading

3.2.1.2 Internal and Boundary Conditions

The entire decay heat load, corresponding to an axial maximum value, has been concentrated in the fuel for simplicity and conservatism. A solar heat load has been distributed over the neutron shield jacket at all times except during the fire.

The heat transfer from each fuel element to the neutron absorber liner is modeled by a combination of thermal radiation and conduction. The results of detailed fuel pin temperature analyses have been used to establish the radiation-conduction characteristics of a single point fuel element node with a heat transfer area corresponding to the inside of the neutron absorber liner. This detailed pin temperature analysis was performed by a modified version of WATSON.

The temperature of the center node of the fuel element, as calculated by ANSYS, is not an actural temperature since this center node in ANSYS is a zero volume node. The average of the three nodes that form one of the triangular elements will give the average temperature of that element.

An effective conductivity of the clad neutron absorber liner is calculated based on a series resistance of the clad and absorber normal to the surface of the basket. Thermal radiation supplements conduction heat transfer across the 20 mil gap between the aluminum basket and the cask inner shell and across the 15 mil gap between the neutron absorber plate and the aluminum basket. Thermal radiation is also considered across the ports in the PWR basket.

Turbulent natural convection occurs in the neutron shield because the effective convection length exceeds 9 inches[7] and the Grashof number exceeds 10^9. Radiation in the dry neutron shield has been assumed to be one dimensional in the radial direction.

Radiation and natural convection heat transfer on the surface of the cask and inside the water jacket have been modeled as recommended in the "Cask Designers Guide"[5]. A fin effectiveness corresponding to that of the limiting post-fire steady-state conditions has been used throughout all transients. The fire analysis is perfomed with the neutron shield jacket in place. This is the design basis and is more likely than complete stripping of the jacket structure.

3.2.2 WATSON - FETA

The WATSON program provides an approximate two dimensional steady state temperature distribution applicable to the pins of a square array dry fuel element. The temperatures calculated by this program are conservative since only radiation is considered. Because of this the 1.1 axial peaking factor is not included in heat generation rate. Only uniform wall temperatures may be considered. Each pin is assumed to have one temperature and all fuel pin locations are assumed to generate identical power. The emissivity within the array is constant and must be the same as the enclosure wall.

The program FETA (Fuel Element Thermal Analysis) was developed from

WATSON and is basically the same except conduction is considered through the gas coolant. FETA was used to determine fuel element conduction and radiation properties for use in ANSYS.

3.2.2.1 Analytical Considerations

The heat transfer in a dry fuel element consists of radiation, conduction and convection. The WATSON analysis has been restricted to radiation and FETA to radiation and conduction. Natural convection may be significant but has not been incorporated in either program because of the much greater complexity of an appropriate mathematical model, the large number of additional uncertainties introduced and what is currently judged to be a small return on the added effort required to evaluate natural convection explicitly.

The computation of temperature in FETA is based on an energy balance of a single fuel pin considering the heat transfer to surrounding pins as far as two rows removed (Figure VIII-4). Radiation heat transfer occurs between the reference pin and the four closest (primary) pins, four diagonal pins, and eight (secondary) pins located two rows from the reference pin. Conduction heat transfer occurs between the reference pin and the adjacent four primary pins as well as four diagonal pins. With the exception of the corner pins, heat transfer at the boundary of the element occurs by means of one conduction link and one combined radiation link between the enclosing wall and each pin of the outside

row of the fuel element. One combined radiation link and three conduction links are utilized at each corner of the fuel element.

The energy equation for each pin of unit length is given in equation (3).

$$q_i = \sum_{j=1}^{N} \left[A_i \, \sigma \, F \, (T_i^4 - T_j^4) \right] + \sum_{j=1}^{N^1} \frac{kA_i}{l_j} (T_i - T_j) \quad (3)$$

where

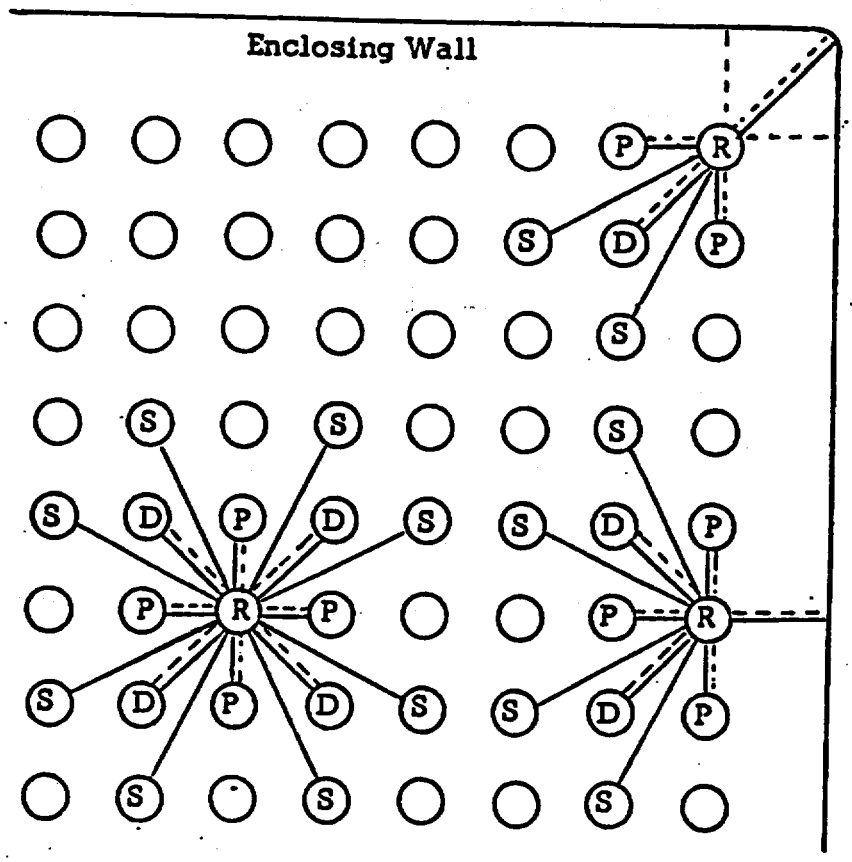
q_i = heat generated in pin i, BTU/hr-ft

A_i = surface area of pin i, (π D), ft^2/ft

σ = Stefan-Boltzmann constant, 0.1713×10^{-8} BTU/hr-ft^2-$^oR^4$

FIGURE VIII-4

HEAT TRANSFER LINKS IN FUEL ELEMENT



---- = Conduction Links

—— = Radiation Links

R = Reference Pin

P = Primary Pin

D = Diagonal Pin

S = Secondary Pin

T_i = surface temperature of pin i, ^oR

T_j = surface temperature of pin j, ^oR

F = general radiation exchange factor

k = gas conductivity, BTU/hr-ft-^oF

A_j = area for conduction heat transfer between pins i and j, ft^2/ft

ℓ_j = conduction length between pins i and j, ft

N = number of radiation links applicable to pin i, (N = 16 for internal pins, N = 6 for outside corner pins, N = 10 for outside row pins, excluding corners).

N' = number of conduction links applicable to pin 1, (N = 8 for internal pins, N = 6 for all outside row pins).

The general radiation exchange factor is evaluated by

$$F = \left(\frac{1}{F_{ij}} + \frac{1}{E_1} + \frac{1}{E_2} - 2\right)^{-1} \tag{4}$$

where

E_1, E_2 = surface emissitivities, $E_1 = E_2$ for heat transfer between pins; E_1 is pin emissivity and E_2 is wall emissivity for pin to wall heat transfer.

F_{ij} = configuration factor. (Reference 9)

Justification for the use of equation (4) is given in Reference 9 .

Conduction heat transfer for pins within the fuel element array is approximated by four links to the primary pins and four links to the diagonal pins. The primary effective area is the projected area of a pin. For the diagonal links the effective area is the difference between the surface area of a rod and the projected area used for the primary links. The corresponding effective conduction lengths are the mean distances between pin surfaces. In effect:

$$\text{Primary } A_j = D \tag{5}$$

$$\text{Diagonal } A_j = \frac{\pi D}{2} - D = 0.571\ D \tag{6}$$

$$\text{Primary } \ell_j = P - \frac{\pi}{4} D \tag{7}$$

$$\text{Diagonal } \ell_j = 1.414\ P - \frac{\pi}{4} D \tag{8}$$

Conduction to the enclosing wall is simulated by two primary links and one diagonal link for corner pins. For the remaining pins of the outer row only one primary conduction link is utilized.

The above procedure is considered sufficiently conservative because convection is not considered. It is judged that the above conservatism combined with pessimistic emissivities compensates for the possible unconservatism mentioned in Reference 5.

Uniform wall temperatures were used in FETA to establish the single point fuel element heat transfer characteristics used in ANSYS.

The above description of FETA generally is the same for WATSON except for references to condition. The second term of equation (3), which includes the conduction effects, is not included in the WATSON analysis.

4.0 Calculations and Results

This section provides the details of hand calculations and computer calculations applicable to the thermal analysis of the cask.

4.1 Hand Calculations

4.1.1 Surface Conditions

Under steady state conditions the combined convection-radiation heat transfer coefficient of the finned surface is:

$$h = \frac{A_c}{A_o}(0.18)(T_1 - T_o)^{0.3333}$$
$$+ \frac{A_R}{A_o}(0.173)(10)^{-8}(F)\left[(T_1+460)^2 + (T_o+460)^2\right]$$
$$x\left[(T_1+460) + (T_o+460)\right] \qquad (9)$$

A_o is the convection heat transfer area as defined in Reference 5. It is based on a water jacket length of approximately 13 feet and a fin efficiency-established-iteratively as recommended in Ref. 5. For the external fins A_c 832.75 in^2 with a fin efficiency (N_c) = .817 (See appendix A for calculations)

A_o is the unfinned jacket surface and A_R is the radiation heat transfer or "String" area.[5] Both are based on the water jacket length of 13 feet.

F is the fin radiation exchange factor [5] given by:

$$F = \left[\frac{1}{\epsilon_e} + \frac{A_1}{A_2} \left(\frac{1}{\epsilon_2} - 1 \right) \right]^{-1} \qquad (10)$$

ϵ_e is the effective fin emissivity [5] given by:

$$\epsilon_e = \left[1 + \frac{S}{S+2_1} \left(\frac{1}{\epsilon r} - 1 \right) \right]^{-1} \qquad (11)$$

where

l = fin height

S = spacing between fins

ϵ_r = actual surface emissivity

Based on a fin thickness of 3/16 in., and an actual emissivity of .5, the effective emissivity of the finned surface is .8. Since A_2 is infinite compared to A_1, $F = \epsilon_e = .8$

For the half hour fire condition, the actual surface emissivity is perscribed by regulation to be 8 . The effective emissivity is .941 for the finned surface. Since for the fire condition $A_2 = A_1$ and the fire emissivity is .9, the radiation exchange factor is .852.

The solar heat load is based on a daily average normal heat flux of

114 BTU/hr ft^2 [5] applied to the projected area of the neutron shield. The equivalent surface heat flux in the ANSYS calculations assuming the solar heat load is distributed over one half the cask surface is then 92 BTU/hr ft^2. With a water jacket node thickness of .75 in., the heat generation in the water jacket is .8565 BTU/hr-in^3.

4.1.2 Neutron Shield Conditions

Natural convection in the water filled neutron shield is based on a conventional turbulent convection coefficient applicable to a vertical wall [6,7] which yields:

$$h_{NC} = 65 (\Delta T)^{1/3} \qquad (12)$$

for an average temperature of 200°F. This coefficient increases with water temperature. Its use at high temperatures is thus conservative. The turbulent correlation has been used because the active convection height exceeds 9 inches and the Grashof number is greater than 10^9.

With a dry neutron shield, the natural convection coefficient is assumed to be the same as on a vertical surface.[5]

$$h_{NC} = 0.18 (\Delta T)^{1/3} \qquad (13)$$

ΔT is the temperature difference between the shell and the bulk water temperature.

The neutron water jacket is finned on both inner and outer diameters to assist in removing heat when the tank is dry under post fire conditions.

In order to determine the fin convection area A_c to be used with equation (12) and (13) a study was done to determine the effect of fin temperature and fluid temperature on fin efficiency. The results of this study are shown in Appendix A.

During fire accident conditions, the inner fin convection area is 789.4 in^2/in of water jacket with a fin efficiency of .8. The outer water jacket fin convection area is 932.9 in^2/in of water jacket.

To include radiation across the neutron shield during the fire accident condition, an effective emissivity is calculated for both finned surfaces. Since the fins are carbon steel and the shell is stainless an actual area weighted average emissivity is calculated as .66. This is based on a carbon steel emissivity of .7 and a stainless steel emissivity of .5. Using .66 as an average the effective fin emissivity, calculated from equation (11), is .9066. The radiation exchange factor for the neutron shield cavity is .845.

When the neutron shield is filled with water, the fins are not as efficient as when the neutron shield is dry. The fin efficiency is .11 and the convection area is 263.1 in^2/in for the inner neutron shield fins and 327.4 in^2/in for the outer neutron shield fins.

4.1.3 Duct Conditions

During normal conditions of transport, the cooling ducts are dry. For the ANSYS analysis, the ducts have a conductivity and specific heat equal to air. Radiation across the duct is also considered by applying appropriate radiation shape factors.

4.1.4 Gap Conditions

Combined conduction and radiation occurs across the 20 mil gap between the aluminum basket and the inner shell. Equation (10) is used with an aluminum emissivity of .2 and a stainless steel emissivity of .5 to establish a radiation exchange factor of .1667 for this gap. For radiation across the gap, A_1 equals A_2.

4.1.5 Neutron Absorber Liner Conditions

The effective conducitivity of the neutron absorber liner is equal to the combined series resistance of the absorber and cladding material as calculated by equation (14).

$$k_{eff} = \left(\frac{t_c}{k_c} + \frac{t_p}{k_p} \right)^{-1} \qquad (14)$$

where

k_{eff} = effective conductivity normal to liner surface, BTU/hr-ft-^{o}F

t_c = clad thickness, in.

t_p = absorber thickness, in.

k_c = clad conductivity, BTU/hr-ft-^oF

k_p = absorber conductivity, BTU/hr-ft-^oF

The parameters used in equation (14) applicable to the PWR and the BWR liners are listed in Table VIII-6. They are based on Figure X-9. Tolerances have been selected so that the effective conductivity is minimized.

An effective film coefficient of 95.2 BTU/hr-ft^2-^oF applicable to the 15 mil gap between the absorber liner and the aluminum basket is based only on a helium conductivity of .119 BTU/hr-ft-^oF. The effects of variable temperature on the conductivity have been found to be negligibly small.

The effective heat capacities of the neutron absorber liners are based on an absorber to liner volume fraction of .716 for the PWR design. Appropriate material properties are listed in Table VIII-2.

TABLE VIII-6

ABSORBER LINER PARAMETERS

	PWR
k_{eff}, Btu/hr-ft-°F	21.23
t_c, in. *	.05
t_p, in. *	.30
k_c, Btu/hr-ft-°F @ 450°F	10.3
k_p, Btu/hr-ft-°F @ 450°F	41.8

Reference Equation (14)

* Values used for poison and clad thickness vary slightly from design values.

4.1.6 Fuel Element Conditions

In all ANSYS calculations each fuel element has been simulated by a single node considering only radiation heat transfer and conduction between the fuel and the element walls.

An effective volumetric specific heat for the fuel element node is obtained from equation (15). The heat capacity of the active pins has been distributed over the cross-sectional area of the entire fuel element.

$$\rho C_{eff} = N\frac{\pi}{4}\left[\frac{\rho_c C_c (D_o^2 - D_i^2) + \rho_f C_f (D_i^2)}{H^2}\right] \qquad (15)$$

where:

ρC_{eff} = effective specific heat of node, Btu/ft^3 - ^oF

N = number of active pins in an element

ρ_c = density of clad, lb/ft^3

C_c = specific heat of clad, Btu/lb-^oF

ρ_f = density of fuel, lb/ft^3

C_f = specific heat of fuel, Btu/lb-^oF

D_o = outer diameter of fuel pin, in

D_i = inner diameter of fuel pin, in

H = width of liner wall, in

The basic material properties applicable to the PWR fuel elements and equation (15) are listed in Table VIII-2. Associated dimensions are given in Table VIII-7.

Table VIII-7

Fuel Element Geometric Parameters

	PWR
N	204
Do, in.	.4220
Di, in.	.3734
H, in.	8.875

Reference Equation (15)

The temperature of the central fuel node in ANSYS does not represent the maximum fuel temperature since this is a zero volume node. Only the average fuel temperature can be determined from ANSYS by averaging the nodes that make up each corner of the triangular fuel elements.

The radiation exchange factor for the fuel element model in ANSYS is established by equation (16) and is based on wall and peak fuel pin temperatures obtained from explicit FETA analyses considering only radiation heat transfer.

$$F = \frac{\varphi}{\sigma(T_i^4 - T_o^4)} \qquad (16)$$

where

F = radiation exchange factor

φ = heat flux at the element wall, Btu/hr-ft^2

σ = Stefan-Boltzmann constant $.173 \times 10^{-8}$ Btu/hr-ft^2-$^oR^4$

T_i = maximum temperature, ^oR

T_o = wall temperature, ^oR

The effective conductivity for the fuel element model in ANSYS is then established by equation (17)

$$k = \frac{\ell \left[\varphi - F\sigma \left(T_i^4 - T_o^4\right)\right]}{T_i - T_o} \qquad (17)$$

where:

k = node conductivity, Btu/hr-ft-^oF

ℓ = conduction length from node center to wall (half the fuel element width), ft.

φ = heat flux at the element wall, Btu/hr-ft^2

F = radiation exchange factor from eq.(16)

σ = Stefan-Boltzmann constant (eq. (16)).

T_i = maximum temperature, ^oR

T_o = wall temperature, ^oR

The temperatures in equation (17) are obtained from explicit FETA analyses considering both radiation heat transfer and conduction. The helium conductivities used in FETA have been conservatively based on wall temperatures instead of the higher average fuel element temperatures. This also eliminates the need for iterative analyses.

A summary of parameters used in equations (16) and (17) to find the PWR single point heat transfer coefficients is presented in Table VIII-8.

Table VIII - 8

PWR Single Point Fuel Element Characteristics

	PWR	
φ, Btu/hr-ft^2	1121	
ℓ, ft	.3698	
T_o, ^oR *	960	1460
T_i, ^oR (radiation only)	1483	1709
F	.1626	.1626
T_i, ^oR (radiation & cond.)	1341	1652
k, Btu/hr-ft-^oF	.4374	.5863
T_k, ^oF **	881	1192

Reference Equations (16) & (17)
* Calculations have been performed at two wall temperatures to determine the temperature dependence of the fuel element characteristics.
**This is the node temperature, which establishes the temperature dependence of conductivity in the thermal property table. (Table VIII-4)

The decay heat generation rate in the fuel is uniform for each basket design and is assumed to be the maximum based on an axial peaking factor of 1.1. In the computer analysis the volumetric heat generation of each fuel node is established by equation (18).

$$q''' = \frac{Q'(1.1)}{N\,L\,A}$$

$$= 3.413(10)^3 \frac{(Q)\ (1.1)\ (144)}{13\ NH^2} \qquad (18)$$

where

q''' = heat generation rate, Btu/hr-ft^3

Q' = cask average decay heat load, Btu/hr

N = number of fuel elements in the cask

L = axial conduction length, 13 ft.

A = fuel element cross section, ft^2

Q = cask average decay heat load, kw

H = fuel element width, in.

Using equation (18) a heat generation rate of 3594 Btu/hr-ft^3 is calculated for the PWR fuel nodes in ANSYS based on a decay heat load of 70 kw, and a loading of 10 elements with a width of 8.998 in. each.

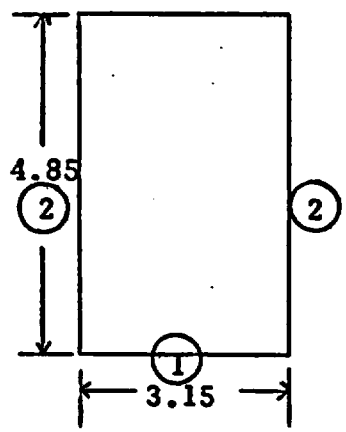
Thirteen feet is used in equation (18) rather than 12 feet, the active fuel length. 13 feet is the length of the water jacket and the approximate

distance between end impact structures. This is the length of the cask over which radial conduction occurs. It is considered acceptable for use in equation (18) since much of the generated heat is produced by neutron and gamma absorption outside of the fuel elements.

4.1.7 PWR Basket Port Radiation Factors

In the ANSYS analysis, radiation across the ports in the PWR basket has been considered. The view factors are calculated as follows. Since the analysis is done in two dimensions only, the basket is considered to be 1 inch deep.

Rectangular Port



From Fig. 5-17 of Ref. 4 with $X = 1$ and f_{1-2} to be found.

$Y = 3.15 \qquad Z = 4.85$

$f_{1-2} = .125$

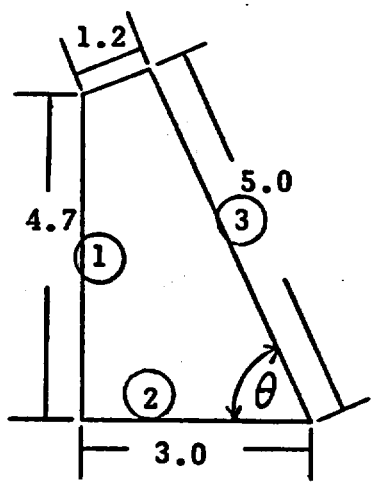
For f_{2-2}, using Fig. 5-18 of Ref. 4 with small side $= 1$. The ratio equals .32 and using curve 4, f_{2-2} equals .16.

Now, calculate F_{1-2} for input to ANSYS

$$F_{1-2} = \frac{1}{\frac{1}{f_{1-2}} + \left(\frac{1}{\epsilon_1} - 1\right) + \frac{A_1}{A_2}\left(\frac{1}{\epsilon_2} - 1\right)}$$

The results of this calculation for the triangular and rectangular ports are shown in Table 5. (NOTE: f_{1-2} is the geometric shape factor. F_{1-2} is the gray body shape factor.)

Triangular Ports



f_{1-2} can be found from Fig. 5-17 of Ref. 4 with $X = 1$

$Y = 4.7 \qquad Z = 3.0$

$$f_{1-2} = .08$$

f_{3-1} is for two parallel planes separated by an amount δ and inclined slightly

$$\delta = \frac{(3 + 1.2)}{2} = 2.1 \qquad \theta = 64^\circ$$

$$f_{3-1} = f \text{ Parallel } (\sin \theta)$$

f parallel = .22 for ratio = .48 and curve 4 of Fig. 5-18 of Ref. 4.

$$f_{3-1} = .22(.9) = .2$$

For f_{2-3}, two perpendicular planes inclined slightly were used. Using Fig. 5-17 of Ref. 4, with $X = 1$

$Y = 3 \qquad Z = 5$

$f_{\perp} = .125$

$$f_{2-3} = \frac{f_{\perp}}{\sin\theta} = .14$$

4.2 Computer Program Input

4.2.1 ANSYS Input

The ANSYS calculations from the main basis for satisfying the AEC requirements concerning the safety of the cask in regard to thermal performance and also for determining the temperatures used in the structural section (Section XI). The two dimensional thermal ANSYS analysis was performed on the cask with the PWR basket for the steady state normal conditions of transport without auxiliary cooling and for fire accident conditions.

The element and node arrangement and the geometric information applicable to them in ANSYS were generated by hand. (Figure VIII-2 and Figure VIII-3). The natural convection coefficients and radiation exchange factors are listed in Table VIII-4. The material properties are listed in Table VIII-2.

The heat generation rate at the surface is presented in section 4.1.1 and the fuel node heat generation rate is presented in section 4.1.6. Boundary conditions, ie. ambient temperatures, are those specified in Reference 5 and AEC regulation, Title 10 CFR Part 71. Under normal steady-state conditions and post-fire steady state, the ambient temperature is 130°F.

The initial condition for the fire accident transient assumes an ambient temperature of 100°F which is considered more reasonable

than 130°F. The cask enters the fire with a temperature distribution equivalent to normal conditions of transport, that is, no auxiliary cooling and the neutron shield tank full of water. Just before the start of the fire, the neutron shield water is assumed instantly lost. After the half hour fire, the ambient temperature is 130°F.

4.2.2 FETA - WATSON Input

A FETA analysis was performed for the PWR reference fuel element to establish the heat transfer characteristics of the single point fuel element model used in ANSYS for two representative wall temperatures, 500°F and 1000°F. A summary of the FETA input is given in Table VIII-9. Background information on FETA is provided in section 3.2.2 and 4.1.6.

The WATSON program was used to determine fuel pin temperatures for normal steady state and post fire steady state conditions. The WATSON calculations were performed for PWR and BWR representative fuel. Wall temperature used was the average of the wall temperatures determined by ANSYS. Fuel element power was 70 kw. The axial peaking factor of 1.1 was not used since WATSON only considers radiation heat transfer and would be expected to give high conservative temperatures.

TABLE VIII-9

FETA INPUT SUMMARY

	PWR
Pin array size	15 x 15
No. of active pins	204
Pin power, Btu/hr-ft	16.26
Pin diameter, in.	.422
Pin pitch, in	.563
F_{12} *	.1263
F_{13} *	.08536
F_{14} *	.01917
Pin emissivity	.4
Wall emissivity	.5
He conductivity, Btu/hr-ft-°F	.128 @ 500°F
	.167 @ 1000°F
Pin to wall clearance, in	.2855

* These are configuration factors defined in Reference 9 and in Figure VIII-4. F_{12} applies to primary pins, F_{13} applies to diagonal pins and F_{14} applies to secondary pins.

TABLE - 9A

WATSON INPUT SUMMARY

	PWR	BWR
Pin array size	15 x 15	7 x 7
No. of active pins	225	49
Pin power-watts/cm	.085	.162
Pin dia - cm	1.07	1.43
F_{12}	.1264	.121
F_{13}	.0854	.083
F_{14}	.0191	.0233
Pin emissivity	.4	.4

In order to calculate fuel thermal expansion, it was necessary to determine an average fuel temperature of the hottest fuel element. An average temperature was used since fuel tie rods, not the hottest pin temperature, determine element expansion. For PWR fuel a center to edge average temperature was used since PWR fuel tie rods are distributed about halfway between the center and the edge of the element. Since BWR fuel tie rods are on the edge of the element, an edge average was used for BWR fuel.

A summary of the WATSON input is given in Table VIII-9A. Background information on WATSON is provided in section 3.2.2 and 4.1.6.

4.3 Normal Conditions of Transport Results without Auxiliary Cooling

The steady-state normal conditions of transport has been analyzed for the PWR case only. Because PWR and BWR heat loads are the same the shell and lead temperatures will be the same for each loading. But the temperature distributions in the fuel basket will be different because of different geometries. Appendix B calculates the maximum BWR basket temperature is approximately equal to the maximum PWR basket temperature. The complete summary of ANSYS node temperatures is presented in Table VIII-10.

ANSYS - ELECTROTHERMAL ANALYSIS SYSTEM UPTIT REV 2 CDC-CYBER JAN 1,1972
SWANSON ANALYSIS SYSTEMS, INC. ELIZABETHTOWN PENNSYLVANIA 15637 PHONE (412) 751-1940

P=1.1 130 DEG F AMBIENT 70RW DECAY HEAT LOAD STEADY STATE SOLN JHA 1A-A 22.0400 10/ 8/74
ANU = 222.825

***** TEMPERATURE SOLUTION ***** TIME = 0. TIME STEP= 3 ITERATION= 2 CUM. ITER= 4

NODE	TEMP	NODE	TEMP	NODE	TEMP	NODE	TEMP	NODE	TEMP
1	542.24	2	542.44	3	549.94	4	551.84	5	1009.5
6	445.40	7	445.17	8	443.64	9	594.12	10	977.91
11	445.37	12	444.18	13	434.65	14	433.85	15	414.11
16	417.14	17	346.72	18	496.63	19	383.22	20	363.52
21	342.48	22	329.10	23	423.01	24	433.92	25	413.24
26	403.02	27	394.12	28	434.58	29	540.43	30	540.62
31	546.02	32	548.24	33	543.59	34	580.17	35	566.87
36	576.17	37	445.99	38	444.88	39	435.79	40	434.59
41	412.73	42	402.98	43	394.81	44	436.96	45	413.87
46	403.48	47	399.60	48	391.18	49	383.42	50	538.07
51	54[illegible].89	52	539.11	53	536.78	54	532.64	55	518.18
56	498.23	57	477.89	58	462.76	59	445.08	60	537.02
61	539.78	62	537.95	63	535.23	64	527.77	65	515.33
66	498.33	67	476.66	68	460.54	69	443.48	70	535.83
71	535.80	72	532.42	73	529.46	74	525.79	75	515.38
76	507.43	77	447.51	78	446.71	79	453.12	80	441.13
81	445.[illegible]	82	415.82	83	444.74	84	400.25	85	391.95
86	384.27	87	344.[illegible]	88	343.13	89	329.60	90	323.40
91	528.07	92	525.54	93	515.07	94	495.60	95	472.61
96	535.53	97	533.19	98	528.53	99	529.83	100	530.94
101	521.46	102	500.99	103	475.35	104	529.16	105	531.58
106	522.42	107	503.52	108	476.98	109	466.90	110	457.98
111	448.34	112	444.72	113	442.97	114	442.02	115	417.59
116	406.57	117	400.07	118	443.09	119	441.73	120	418.42
121	41[illegible].95	122	398.51	123	391.54	124	442.03	125	441.30
126	417.97	127	4[illegible]6.48	128	490.00	129	522.75	130	521.87
131	528.[illegible]5	132	528.62	133	469.72	134	469.24	135	468.32
136	457.29	137	438.81	138	439.89	139	439.25	140	415.64
141	404.01	142	40[illegible].59	143	391.93	144	386.00	145	519.43
146	514.[illegible]1	147	515.5[illegible]	148	438.58	149	437.96	150	414.71
151	414.37	152	408.20	153	519.67	154	518.09	155	513.15
156	511.85	157	518.48	158	519.47	159	970.78	160	458.85
161	457.44	162	447.94	163	444.85	164	440.87	165	439.11
166	438.44	167	414.71	168	414.36	169	440.12	170	391.92
171	384.21	172	364.11	173	343.18	174	329.64	175	323.45
176	524.79	177	523.28	178	512.78	179	527.81	180	524.66
181	512.73	182	500.68	183	510.06	184	493.70	185	492.63
186	497.49	187	500.33	188	451.57	189	449.86	190	441.38
191	439.34	192	438.65	193	415.85	194	405.53	195	399.35
196	438.57	197	416.65	198	448.94	199	397.78	200	391.15
201	383.52	202	439.12	203	438.44	204	415.66	205	405.32
206	399.15	207	448.62	208	446.53	209	439.88	210	438.37
211	437.40	212	414.31	213	443.77	214	399.43	215	444.58
216	441.74	217	438.46	218	472.33	219	466.43	220	456.98
221	45[illegible].[illegible]1	222	447.6[illegible]	223	444.83	224	438.52	225	470.92
226	445.9[illegible]	227	445.76	228	448.53	229	445.65	230	441.45
231	437.06	232	446.49	233	480.20	234	478.99	235	469.32
236	467.15	237	456.85	238	446.16	239	439.81	240	438.74
241	437.8[illegible]	242	437.24	243	436.64	244	413.85	245	403.62
246	399.[illegible]	247	391.52	248	383.86	249	363.59	250	342.74
251	329.26	252	323.07	253	442.12	254	454.22	255	443.47
256	44[illegible].02	257	437.48	258	436.84	259	413.79	260	403.37
261	399.[illegible]9	262	444.53	263	462.21	264	455.07	265	453.48
266	438.57	267	437.96	268	437.26	269	414.79	270	404.81
271	398.61	272	437.19	273	415.61	274	408.03	275	397.04
276	390.58	277	437.73	278	437.07	279	414.67	280	404.52
281	398.46	282	436.95	283	436.32	284	413.22	285	403.13
286	389.[illegible]	287	389.91	288	383.33	289	363.40	290	342.57
291	329.[illegible]	292	322.91	293	436.49	294	436.23	295	435.71
296	413.[illegible]	297	402.99	298	398.93	299	437.46	300	436.82
301	413.07	302	403.42	303	399.17	304	398.77	305	383.00
306	443.57	307	447.57	308	447.80	309	445.85	310	446.40
311	446.[illegible]	312	444.13	313	437.16	314	437.83	315	441.31
316	44[illegible].47	317	439.19	318	438.43	319	415.33	320	404.88
321	398.43	322	438.60	323	416.35	324	408.54	325	397.21
326	390.44	327	439.41	328	438.69	329	415.58	330	405.05
331	394.70	332	390.94	333	343.37	334	363.33	335	342.40
336	329.[illegible]1	337	322.84	338	449.09	339	438.33	340	414.49
341	4[illegible]3.48	342	399.29	343	437.51	344	436.88	345	413.45
346	4[illegible]3.15	347	394.02	348	390.70	349	382.89	350	436.39
351	435.70	352	413.03	353	412.81	354	398.60	355	435.95
356	435.38	357	413.84	358	413.96	359	398.07	360	435.92
361	435.36	362	414.02	363	437.30	364	396.65	365	398.23
366	382.0[illegible]	367	363.13	368	347.32	369	328.85	370	322.69
371	382.05	372	0.	373	0.	374	0.	375	130.00
376	13[illegible].00	377	130.00	378	130.00	379	130.00	380	130.00
381	13[illegible].00								

The maximum temperature in the water filled neutron shield is 343°F, which corresponds to a saturation pressure of 123 psia. Since this is the hottest temperature on the interior of the neutron shield, it is unlikely that this pressure would ever be reached in the neutron shield tank.

A brief summary of temperatures thru the cask is given in Table VIII-11. The maximum lead temperature is 407°F (node 116) which is below the lead melting point of 621°F.

TABLE VIII-11

TEMPERATURE DISTRIBUTION NORMAL CONDITIONS OF TRANSPORT

Node No.	Description	Ambient Temp. 100°F	130°F
90	Water Jacket	300	324
88	Outer Shell	320	343
87	Outer Lead	341	364
116	Inner Lead	383	407
115	Inner Shell	394	418
137	Aluminum (Min.)	416	439
1	Aluminum (Max.)	519	542

4.4 Hypothetical Fire Results

A brief summary of the cask thermal response during and after the fire is provided in Figure VIII-5. The outer lead temperature is seen to peak within the first half hour and then slowly decrease to its steady state value. The inner lead temperature does not reach its maximum until steady state conditions. The transient and steady state calculations show that the lead will not melt either during or after the fire.

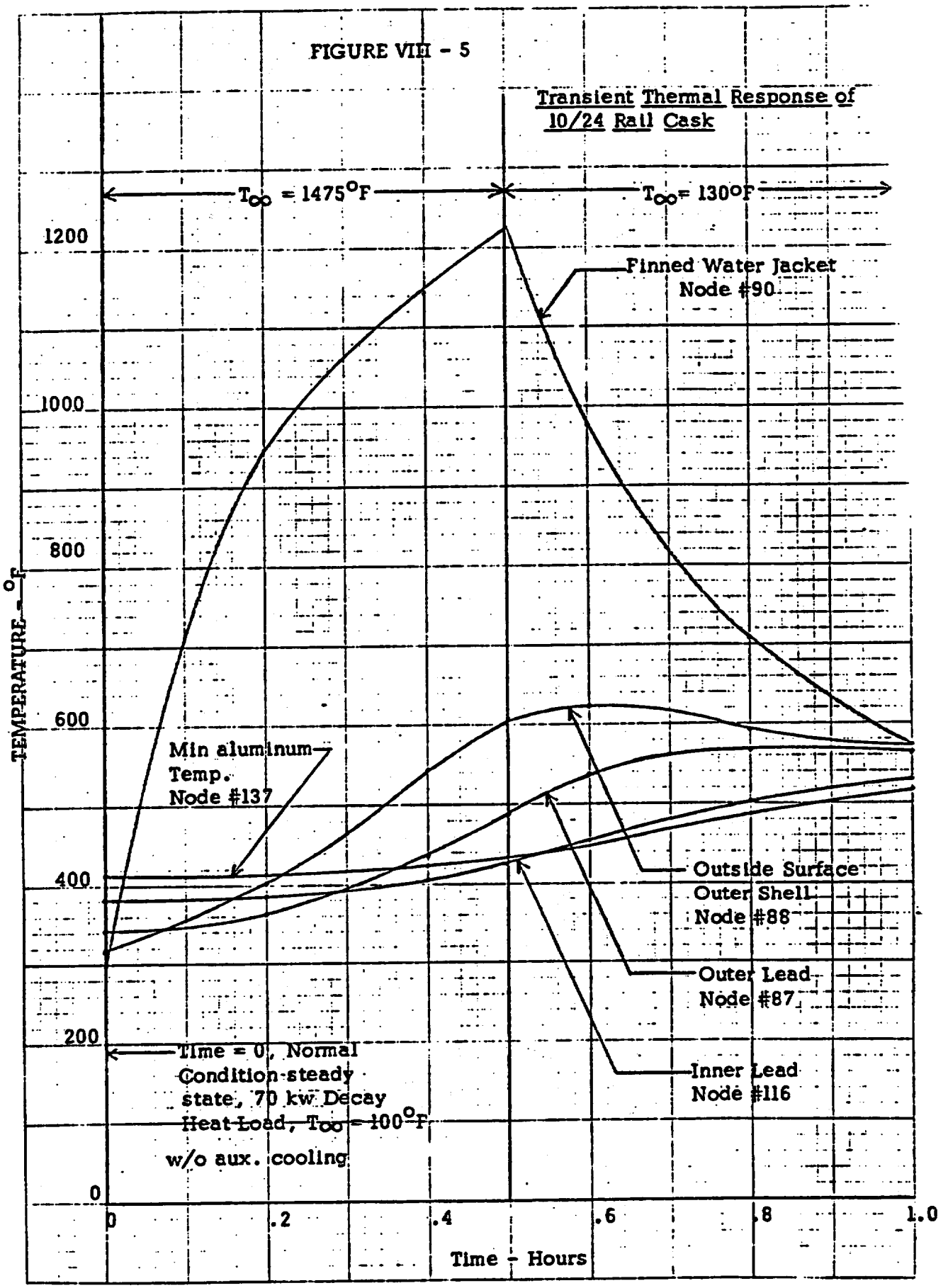
A brief summary of the cask temperature distribution is shown in Table VIII-12. The complete summary of ANSYS node temperatures is presented in Table VIII-13

TABLE VIII-12

TEMPERATURE DISTRIBUTION FIRE ACCIDENT STEADY STATE

Node No.	Description	Temp. °F
90	Water Jacket	323
88	Outer Shell	525
87	Outer Lead	545
116	Inner Lead	599
115	Inner Shell	609
137	Aluminum (Min.)	627
1	Aluminum (Max.)	732

FIGURE VIII - 5



K&E 10 X 10 TO THE 1½ INCH KEUFFEL & ESSER CO. 359-3DG

ANSYS - ENGINEERING ANALYSIS SYSTEM UP117 REV 2 CDC-CYBER JAN 1,1972
SWANSON ANALYSIS SYSTEMS, INC. ELIZABETH, PENNSYLVANIA 15037 PHONE (412) 751-1940

P-101 130 DEG F AMBIENT 70KW DECAY HEAT POST TIME STEADY STATE JMA 10-9 11.6494 10/ 9/74
ARU = 266.163

***** TEMPERATURE SOLUTION ***** TIME = 0. TIME STEP= 3 ITERATION= 3 CUM. ITER= 5

NODE	TEMP	NODE	TEMP	NODE	TEMP	NODE	TEMP	NODE	TEMP
1	731.78	2	731.94	3	738.67	4	739.75	5	1142.5
6	790.12	7	789.60	8	780.41	9	788.82	10	1096.1
11	632.95	12	631.74	13	623.21	14	622.38	15	605.19
16	598.78	17	586.26	18	577.85	19	508.62	20	544.36
21	524.93	22	328.97	23	322.73	24	622.45	25	609.43
26	599.76	27	588.04	28	623.11	29	729.86	30	730.05
31	735.62	32	737.06	33	768.86	34	705.64	35	754.27
36	757.48	37	633.63	38	632.45	39	624.34	40	623.11
41	603.88	42	594.69	43	589.01	44	628.47	45	605.17
46	595.60	47	590.04	48	578.75	49	508.75	50	727.11
51	729.50	52	727.55	53	724.96	54	728.18	55	709.48
56	686.16	57	665.74	58	658.67	59	633.06	60	726.11
61	728.41	62	726.39	63	723.39	64	715.43	65	703.73
66	686.27	67	664.58	68	648.45	69	631.61	70	725.18
71	724.84	72	721.75	73	718.78	74	715.06	75	704.52
76	696.62	77	676.58	78	655.60	79	641.84	80	629.66
81	628.80	82	607.31	83	597.03	84	598.82	85	579.68
86	569.85	87	544.94	88	525.44	89	329.24	90	322.99
91	718.27	92	714.82	93	704.29	94	684.71	95	661.54
96	724.87	97	722.51	98	717.81	99	718.24	100	719.60
101	718.09	102	684.45	103	663.92	104	718.26	105	720.20
106	711.06	107	691.96	108	665.49	109	655.78	110	646.74
111	636.99	112	633.31	113	631.54	114	630.56	115	609.25
116	599.18	117	590.38	118	631.67	119	630.29	120	609.97
121	602.40	122	588.37	123	579.08	124	630.59	125	629.85
126	608.97	127	599.05	128	590.26	129	712.03	130	711.15
131	716.66	132	717.19	133	657.64	134	657.17	135	649.15
136	646.07	137	627.35	138	620.43	139	627.79	140	607.13
141	597.19	142	591.16	143	579.61	144	569.38	145	708.73
146	707.31	147	704.73	148	627.04	149	626.49	150	609.10
151	596.50	152	590.71	153	708.97	154	707.36	155	702.36
156	701.05	157	706.91	158	707.89	159	1088.4	160	649.53
161	645.31	162	636.58	163	633.48	164	629.43	165	627.64
166	626.96	167	606.12	168	596.51	169	590.64	170	579.67
171	569.77	172	545.02	173	525.52	174	329.28	175	323.02
176	715.28	177	711.94	178	701.98	179	716.29	180	713.29
181	701.75	182	688.95	183	689.21	184	682.78	185	681.69
186	686.19	187	688.73	188	639.28	189	637.56	190	629.95
191	627.88	192	627.18	193	607.31	194	597.87	195	589.50
196	627.09	197	608.00	198	600.85	199	587.52	200	578.62
201	568.83	202	627.64	203	626.95	204	607.10	205	597.62
206	589.27	207	636.41	208	634.35	209	628.41	210	620.86
211	626.16	212	605.59	213	595.77	214	589.75	215	632.49
216	629.72	217	626.89	218	660.68	219	654.25	220	644.59
221	637.86	222	635.34	223	631.91	224	626.54	225	658.85
226	653.63	227	643.36	228	636.16	229	633.42	230	629.40
231	626.06	232	1083.7	233	668.06	234	666.84	235	658.18
236	655.98	237	645.58	238	634.77	239	628.32	240	627.23
241	626.25	242	625.68	243	625.10	244	605.09	245	595.61
246	590.02	247	579.27	248	569.41	249	544.42	250	524.98
251	328.99	252	322.76	253	658.88	254	642.91	255	632.03
256	629.34	257	625.94	258	625.28	259	604.99	260	595.28
261	589.33	262	652.39	263	650.08	264	643.74	265	644.83
266	627.04	267	626.43	268	625.72	269	606.09	270	590.75
271	588.52	272	625.65	273	606.81	274	590.76	275	580.66
276	577.87	277	626.19	278	625.51	279	605.96	280	590.65
281	588.45	282	625.39	283	624.75	284	604.42	285	595.02
286	589.30	287	578.50	288	568.78	289	544.25	290	524.81
291	328.89	292	322.66	293	624.93	294	624.65	295	624.12
296	604.20	297	594.82	298	589.19	299	625.86	300	625.24
301	605.09	302	595.38	303	589.50	304	578.27	305	568.26
306	632.06	307	635.04	308	635.35	309	633.84	310	633.86
311	634.14	312	632.17	313	625.61	314	626.27	315	629.76
316	629.12	317	627.62	318	626.86	319	606.68	320	597.09
321	588.62	322	627.03	323	687.62	324	600.36	325	586.83
326	577.81	327	627.85	328	627.12	329	606.95	330	597.31
331	588.84	332	578.53	333	568.83	334	544.15	335	524.71
336	328.83	337	322.60	338	627.51	339	626.73	340	605.77
341	595.72	342	589.67	343	625.92	344	625.27	345	604.64
346	595.05	347	589.35	348	578.21	349	568.16	350	624.79
351	624.18	352	604.12	353	594.61	354	588.78	355	624.36
356	623.78	357	604.98	358	595.96	359	588.00	360	624.33
361	623.76	362	605.71	363	598.91	364	586.28	365	577.63
366	568.27	367	543.93	368	524.52	369	328.71	370	324.49
371	568.18	372	0.	373	0.	374	0.	375	139.00
376	130.00	377	130.00	378	130.00	379	130.00	380	139.00
381	130.00								

The post fire steady state condition represents the most thermally limiting condition of the cask. The same condition will be reached if all the neutron shield water as well as the auxiliary cooling capability is lost even if there is no fire.

4.5 Normal Operations with Auxiliary Cooling

The cask is normally designed to operate with an auxiliary cooling system. This cooling system is redundant in that there is a second stand-by system, should the first one fail. The cooling system is designed to have a maximum average coolant temperature of 180°F. The dry coolant duct temperature (without auxiliary cooling) is approximately 400°F. The difference between these temperatures is 220°F. On an order of magnitude basis it can be expected during the auxiliary cooling, the normal condition of transport temperature (without auxiliary cooling) would be reduced by this amount (220°F).

The auxiliary cooling system provides an extra measure of safety above what is required by AEC normal conditions of transport. This safety is enhanced by having the second system on stand-by and also by system design that allows both systems to operate by natural circulation.

Natural circulation begins when power is lost to both cooling systems. The driving force is the difference in density caused by boiling in the

cask cooling channels and condensation in the cooling system heat exchanger. The maximum coolant temperature during natural circulation is 250°F. The difference between the dry coolant duct temperature (400-250) is 150°F. On an order of magnitude basis it can be expected that during natural circulation operation the normal condition of transport temperature (without auxiliary cooling would be reduced by this amount (150°F).

4.6 Fuel Pin Temperatures

Fuel Pin Temperatures have been calculated using WATSON. The WATSON analysis was performed for the hottest element which is node 5 Figure VIII-3. The analysis for the BWR element was performed assuming the same wall temperatures as the PWR analysis. The capabilities and limitations of WATSON are discussed in Section 3.2.2. In view of WATSON's limitations, the calculated temperatures should be considered conservative maximum values.

Average temperatures of the hottest element were calculated from the WATSON results. The PWR average was 833°F and the BWR average was 654.9°F for normal conditions. This compares to an ANSYS average of 703°F for the hot PWR element. The ANSYS average is determined by averaging the corner node temperatures of each ANSYS element that models the hot fuel element. It is evident therefore that WATSON results are conservatively high.

The temperatures obtained by the WATSON program are presented in Table VIII-14 and VIII-15.

4.7 Cold Operation

Under cold conditions it is conservatively assumed that the cask is uniformly at -40°F. Freezing of the water in the auxiliary cooling system and in the neutron shield tank will be avoided by the appropriate addition of antifreeze. Under actual cold conditions the interior of the cask will

always be warmer than the outside of the cask. Thermal gradients and consequently the relative thermal expansion of cask materials should therefore not differ substantially from normal operating conditions.

4.8 Seal Temperatures

Explicit thermal analyses of the cask seals have not been performed because the gasket characteristics are not adversely affected by upper bound temperatures which are readily obtained from the analyses at the cask midplane. Thus the temperature of the outer seal is judged to be no greater than the temperature at the inside of the 3/4 inch thick neutron shield jacket with a PWR loading. Similarly the temperature of the inner seal is judged to be no greater than the temperature in the middle of the inner shell hot sector with a PWR loading. A summary of applicable neutron shield jacket and inner shell temperatures in conjunction with the estimated outer and inner seal temperatures under various operating conditions is presented in Table VIII-14

Table VIII-14

Seal Temperature Basis

Cask Condition	Jacket NODE 89 °F	Outer Seal Estimate °F	Inner Shell NODE 115 °F	Inner Seal Estimate °F
Normal Steady-State	330	335	418	425
End-of-Fire	1174	1190	428	435
Post-Fire Steady-State	330	335	609	615
Cold Steady-State*	> -40	> -40	> -40	> -40

*see section 4.7 for details.

4.9 Thermal Expansion and Contraction

Under complete loss of auxiliary cooling conditions the neutron shield water reaches a local maximum temperature of 343°F which corresponds to a saturation pressure of 123 psia. Under fire conditions it is possible that the pressure in the neutron shield tank will become excessively high if the tank has not already been punctured as postulated in the hypothetical accident.

The effects of differential expansion on gap sizes has been considered in all analyses on the basis of fixed normal steady-state gap conditions. As shown in Figure VIII-5 a slight reversal of the temperature gradient occurs at the gap between the basket and inner shell for about .5 hours after the end of the fire (compare inner lead and cold aluminum temperatures). This reversal actually tends to open the gap temporarily which reduces the heat flow from the lead to the aluminum, thus serving as a self-correcting mechanism. The similarity of the relative temperature distribution at the start and during the latter part of the transient confirms that the use of a constant normal steady-state gap is appropriate for analysis of the transient.

4.10 Uncertainty Factors and Design Margins

The entire thermal analysis of the cask has been directed at achieving conservative results by use of maximum heat loads and low heat capacities in the fuel elements. The use of nominal dimensions has

been restricted to conditions where tolerances have a negligible effect on thermal response. Physical phenomena such as blowdown and ruptures have been simulated in a severe and conservative manner. No part of the analysis depends on artificial cooling following any accident. The attenuation of temperatures due to axial heat transfer has not been taken into account. It is therefore anticipated that the actual temperatures in the cask will always be lower than those predicted by this analysis.

5.0 General Conclusions

Under actual operating conditions the auxiliary cooling system is used in order to dissipate approximately 80% of the decay heat load. Although the system will function acceptably in a natural circulation mode, forced circulation will be used normally because of the lower cask temperatures attainable and the better control on heat transport performance possible. This reduces the required cask cool down time prior to unloading. Even with the auxiliary cooling system completely inoperative complete shielding effectiveness and heat dissipation is assured.

The thermally most limiting cask temperatures occur many hours after the hypothetical accident. These same temperatures will be reached even without a fire if the cooling ducts and the neutron shield become dry. Melting of the lead will not occur under any postulated design

conditions. Previous transient calculations indicate that it takes approximately 3 days to reach thermal equilibrium conditions after the start of the fire.

Neither the neutron absorber liners nor the aluminum basket will melt under any accident conditions assuring criticality control and spatial separation of the fuel elements.

Since heat loads are the same for BWR and PWR cases, a separate analysis was not performed for each case. The basis for this is discussed in Appendix B. The use of a single point fuel element model has been found to be acceptable in evaluating the thermal response of an entire cask.

A summary of the most significant results of the rail cask thermal analysis is presented in Table VIII-1 and in Figure VIII-5 as well as in Section 1.3.

6.0 REFERENCES

1. Edwards, A. L., "A Compilation of Thermal Property Data for Computer Heat Conduction Calculations", UCRL-50589, February 24, 1969.

2. Transactions ASME, Vol. 76 pp. 967-985, 1954.

3. Scott, D. B., "Physical and Mechanical Properties of Zircaloy 2 and 4", WCAP-3269-41, May 1965.

4. Kreith, F., "Principles of Heat Transfer", Second Edition, International Textbook Co., 1965.

5. Shappert, L. B., "Cask Designers Guide", ORNL-NSIC-68, February 1970.

6. McAdams, W. H., "Heat Transmission", Third Edition, McGraw-Hill Book Co., 1954.

7. Lauer, B. E. , "How to Evaluate Film Coefficients for Heat-Transfer Calculations", Technical Manual reprinted from THE OIL AND GAS JOURNAL, 1953.

8. The Aluminum Association, "Aluminum Standards & Data", 1972-1973.

9. Watson, J. S., "Heat Transfer from Spent Reactor Fuels During Shipping: A Proposed Method for Predicting Temperature Distribution in Fuel Bundles and Comparison with Experimental Data", ORNL-3439, May 1963.

10. ALCOA letter to N. L. Industries dated 4/23/73.

11. USAEC, "Neutron Absorber Material for Reactor Control", NR (DED), 1962.

12. ANSYS, Engineering Analysis Program, Swanson Analysis Systems, Inc., Elizabeth, Pennsylvania, 1973.

SECTION VIII

Appendix A

In order to prevent the lead shield from melting after the fire accident, fins were added to the annular area that forms the neutron shield tank. During and after the fire the neutron shield tank is dry and the fins were added to transfer heat more efficiently across the gap. The convection coefficient on the finned surface of the neutron shield tank is given by equation (1A) for dry fire accident conditions and (2A) for wet conditions.

$$h_c = .18 \Delta T^{.333} \qquad (1A)$$

$$h_c = 65 \Delta T^{.333} \qquad (2A)$$

The Grashof number is greater than 10^9 based on the neutron shield inner diameter. Therefore, the convective flow in the neutron shield is turbulent. Because of the large gap between the tank walls and the turbulent flow, the convection coefficients are calculated from correlations for vertical walls with the nuetron shield inner diameter as the characteristic length.[1][2]

When fins are added, more convection area is added. But not all of the added area contributes in an equal amount to an increase in heat transfer, since the fin temperature changes along its length and approaches the surrounding fluid temperature. In order to account for

this change in temperature, a fin efficiency is used in calculations. The fin efficiency η is the ratio of the heat transferred across the fin surface to the heat which would be transferred if the entire surface were at the base temperature.

Reference 5 details a method of calculating a fin efficiency. This method is used to calculate a corrected fin efficiency η_c for various average fin temperatures and fin to fluid temperature differences. The results of this study are shown in Table 1A and plotted in Figure 1A. The variation of η_c with fin temperature and ΔT is small.

For the ANSYS thermal solution of the fire accident transient and steady state conditions, a constant value for η_c will be used. Assuming a ΔT of 300°F and a metal temperature of 600°F, the value taken for η_c is .8.

During normal conditions, the neutron shield is water filled. Table 2A shows the results of efficiency calculations for heat transfer to the water filled neutron shield. The calculations show that the fins are not very efficient when in a water environment. The efficiency used for normal conditions was .11.

TABLE 1A

FIN EFFICIENCY - DRY NEUTRON SHIELD

ΔT	Fin Temp. 200°F η_c	Fin Temp. 600°F η_c	Fin Temp. 1000°F η_c
40	.901	.890	.871
100	.871	.858	.831
150	.856	.838	.811
200	.845	.826	.794
300	.827	.807	.774
400	.814	.793	.757

FIGURE - 1A

FIN EFFICIENCY - DRY NEUTRON SHIELD

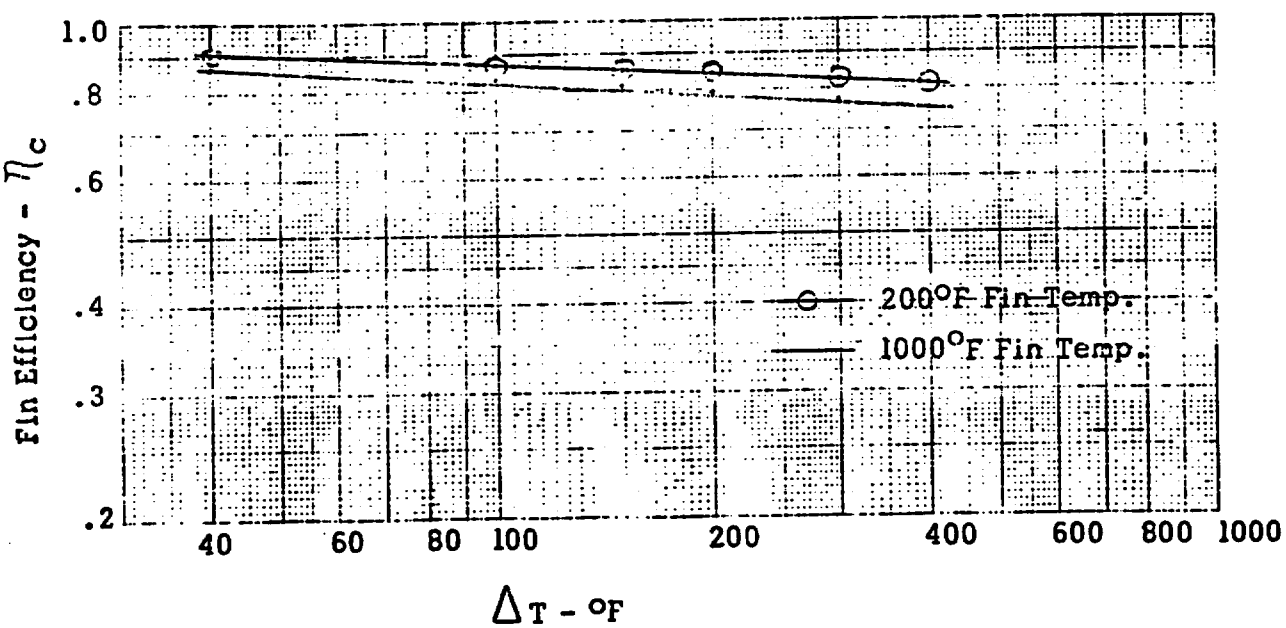


TABLE 2A

FIN EFFICIENCY - WET NEUTRON SHIELD

ΔT	Fin Temp. 200°F η_c
10	.123
20	.109
40	.097

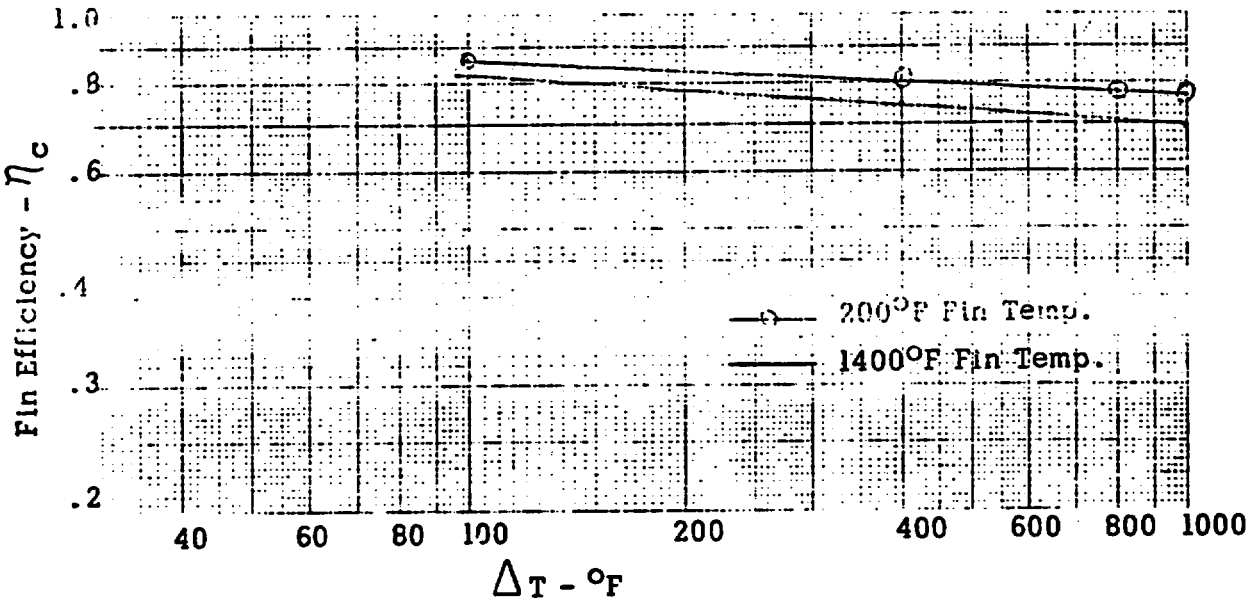
TABLE 3A

EXTERNAL FIN EFFICIENCY

ΔT	Fin Temp. 200°F η_c	Fin Temp. 800°F η_c	Fin Temp. 1400°F η_c
100	.819	.832	.869
400	.747	.763	.810
800	.704	.722	.776
1000	.690	.708	.764

FIGURE 2A

EXTERNAL FIN EFFICIENCY



The same calculations were performed for the cask external fins. The results are shown in Table 3A and Figure 2A. The convection coefficient for the external surface is as recommended in Ref. (5), (Equation 1A).

The results plotted in Figure 2A again show that fin efficiency does not greatly depend on fin temperature or temperature difference. The value η_c = .817 was assumed primarily to allow a conservative amount of heat to flow into the cask during the fire accident condition.

At the steady state ($\Delta T = 194^{o}F$) condition this value of η_c is slightly unconservative. But it is within 2% of the correct value and, since the cask temperatures are not excessively high, it is considered acceptable.

SECTION VIII

Appendix B

The present thermal calculations for the 10/24 rail cask were performed only on the cask with the PWR basket and fuel since the PWR and BWR heat loads are the same. Because the heat loads are equal, the cask body temperatures will be equal. The only difference will be in basket temperatures. The calculations and results presented in this Appendix will show the maximum BWR basket temperature.

Calculations have been performed for a BWR heat load of 363600 Btu/hr (106.6 kw). These results will be used to calculate a BWR basket thermal resistance which will be used to calculate maximum basket temperatures for the lower heat load.

The thermal resistance is given by equation (1B)

$$R = \frac{\Delta T}{Q} \qquad \text{(1B)}$$

ΔT is the temperature difference across the BWR basket. The analysis for 106.6 kw calculated a ΔT of 145°F for post-fire conditions (1020 - 875 = 145) and a ΔT of 137°F for the cask with auxiliary cooling (443 - 306 = 137). This gives a thermal resistance of .000399°F hr-ft^2/Btu for post fire conditions and .000377 °F - hr-ft^2/Btu for auxiliary cooling conditions. The thermal resistance of the BWR basket is seen to be fairly constant over a wide range of temperatures.

Since thermal resistance depends on temperature and geometry, not heat load, these values of thermal resistance may be used for the present lower heat load.

The minimum BWR basket temperature will equal the minimum PWR basket temperature since heat transfer across the gap to the inner shell will be equal. For normal conditions the present minimum PWR temperature is 439°F; for post fire conditions, it is 627°F. From these temperatures and the thermal resistances obtained previously, the ΔT and maximum basket temperature, may be calculated.

<u>Post fire</u>

$$\Delta T = 70 \text{ kw } (1.1)\ (3413)\ (.000399)$$

$$\Delta T = 104.8^{\circ}F$$

Max. BWR basket temp. = 627 + 104.8

= 731.8°F

Compare this to a maximum PWR basket temperature of 732°F as calculated by ANSYS for fire accident conditions.

Normal conditions

$$\Delta T = 70 \text{ kw } (1.1)\ (3413)\ (.000377)$$

$$\Delta T = 99$$

Max. BWR basket temp. = 439 + 99

= 538°F

Compare this to a maximum PWR basket temperature of 542°F as calculated by ANSYS for normal conditions. The PWR and BWR baskets are therefore considered thermally equal and the temperatures are used interchangeably.

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