71-9001

VECTRA

March 30, 1995
CJT-95-011

Docket No. 71-9001

United States Nuclear Regulatory Commission
Office of Nuclear Materials Safety and Safeguards
Division of Industrial and Medical Nuclear Safety
Washington, D.C. 20555

ATTENTION: C. R. Chappell, Section Leader
Cask Certification Section
Storage and Transportation Systems Branch

SUBJECT: IF-300 Shipping Cask
Certificate of Compliance No. 9001
Application for Renewal

ENCLOSURE: Pages to Update VECTRA IF-300 Shipping Cask Consolidated Safety Analysis Report (CSAR) from NEDO-10084-3 (September 1984) to NEDO-10084-4 (March 1995) (10 copies)

Dear Mr. Chappell:

In accordance with the provisions of 10CFR71 and 10CFR §170.31 10.A, VECTRA Technologies, Inc. requests the renewal of the Subject Certificate of Compliance. Attachments A, B, and C of this letter present the following:

Attachment A - Requested modifications and revisions to the existing certificate. These modifications and revisions are related to the deletion of Shoreham-specific fuel details and all supplements referenced by the existing certificate.

Attachment B - The disposition of all supplements to the existing certificate.

Attachment C - Instructions for the insertion of enclosed IF-300 shipping cask CSAR revised pages to update NEDO-10084-3 (September 1984) to NEDO-10084-4 (March 1995).

The enclosed pages to update NEDO-10084-3 to NEDO-10084-4 include the following:

Ltr. Encl. NF01 1/B
Change: LA 4 0

VECTRA Technologies, Inc. · 6203 San Ignacio Ave., Suite 100 · San Jose, CA 95119 · Tel: (408) 629-9800
Fax: (408) 281-6186 Engineering · Fax: (408) 281-6202 Fuel Services

1) New Tables of Contents for existing CSAR Volumes 1 and 2.

2) Revised CSAR Volume 1 pages to provide references to the appendices of new Volume 3.

3) New CSAR Volume 3 which incorporates the Channelled BWR Fuel Basket, High Burnup PWR Fuel, and Outer Plastic Wrap "standalone" SAR amendments referenced by the existing certificate into new Appendices A, B, and C, respectively (see Attachment B).

Text revised by VECTRA in CSAR Volume 1 is indicated by a vertical line in the left-hand margin and all pages containing these revisions have headers reading "NEDO-10084-4 / March 1995". All unrevised CSAR Volume 1 pages have headers reading "NEDO-10084-3" with either September 1984 or February, April, or May 1985 revision dates. NEDO-10084-3 Volume 1 and 2 text revised by the original IF-300 primary License holder (General Electric Company) is indicated by a vertical line in the right-hand margin with an "E" or an "N" to indicate an "Editorial" or "New" change, respectively.

Written communications and questions should be directed to the undersigned.

Respectfully submitted,

VECTRA Technologies, Inc.

James W. Cyphers for C. Temus

Charles J. Temus, P.E.
Licensing Manager
Transportation Products

Attachments:

A) Requested Modifications and Revisions to Existing Certificate of Compliance No. 9001, Revision No. 29

B) Disposition of Certificate of Compliance No. 9001, Revision No. 29 Referenced Supplements

C) Page Insertion Instructions to Update NEDO-10084-3 to NEDO-10084-4

cc: R. K. Kunita - CP&L (w/ Atts. & 1 Encl.)
D. C. Poteralski - CP&L (w/o Atts. or Encl.)
R. Heatherington - CP&L (w/o Atts. or Encl.)
T. E. Tehan - VECTRA/Morris (w/ Atts. & 2 Encls.)
K. A. Hoedeman - VECTRA/San Jose (w/ Atts. & 1 Encl.)
C. H. Froehlich - VECTRA/San Jose (w/ Atts. & 1 Encl.)
File: 0132-00163.000

Attachment A

REQUESTED MODIFICATIONS AND REVISIONS TO EXISTING CERTIFICATE OF COMPLIANCE NO. 9001, REVISION NO. 29

(Crossed-out text [~~text~~] to be deleted and italicized text [*text*] to be added.)

1. **Page 2, Paragraph 5.(a)(2):** Revise as follows:

   "The cask has ~~four~~ *three* types of fuel baskets which can be interchanged to accommodate various fuels. The PWR basket holds seven assemblies, the unchannelled BWR basket holds eighteen assemblies, *and* the channelled BWR basket holds seventeen assemblies~~, and the Shoreham BWR fuel basket holds seventeen Shoreham BWR fuel assemblies. The channelled and unchannelled BWR fuel baskets may be provided with supplementary shielding (depleted uranium) near the cask closure.~~"

2. **Page 2, Paragraph 5.(a)(3):** Revise as follows:

   "... 420-11-3006, Sheet 1, Rev. 0~~; 2045.3000, Sheets 1 to 4, Rev. 1; 2045.3001, Sheets 1 and 2, Rev. 1; 2045.3002, Sheet 1, Rev. 1; and 2045.30003, Sheet 1, Rev. 0~~."

3. **Page 4, Paragraph 5.(b)(1)(ii):** Delete this paragraph completely.

4. **Page 5, Paragraph 5.(b)(1)(iii):** Renumber this paragraph 5.(b)(1)(ii).

5. **Page 5, Paragraph 5.(b)(2)(ii):** Revise as follows:

   "Seven PWR fuel assemblies, seventeen channelled BWR assemblies, *or* eighteen unchannelled BWR fuel assemblies~~, or seventeen Shoreham BWR fuel assemblies~~.

6. **Page 5, Paragraph 5.(b)(2)(iii):** Revise as follows:

   "Above fuel assemblies to be contained in their respective fuel baskets as shown in GE Drawing No. 159C5238 - Sheet 6, Rev. 8, *or* PNSI Drawing No. 420-111-3000, Sheet 1 through 9, Rev. 0~~, or PNSI Drawing No. 2045.3002, Sheet 1, Rev. 1~~.

7. **Page 6, Paragraph 20:** Delete this paragraph completely.

8. **Page 6, Paragraphs 21 and 22:** Renumber as paragraphs 20 and 21, respectively, and revise expiration date in new Paragraph 21.

9. **Page 7, REFERENCES:** Revise as follows:

"~~General Electric Uranium Corporation consolidated application dated September 24, 1984.~~

~~General Electric supplements dated: February 8, April 4, and May 10, 1985, and March 12, 1990.~~

~~Pacific Nuclear Systems, Inc. supplements dated: July 26, 1990; March 28, April 12, July 19, and August 30, 1991; January 3, 1992; and February 25, April 9, July 29, and August 10, 1993.~~

~~VECTRA Technologies, Inc. supplement dated: April 25, 1994.~~

VECTRA Technologies, Inc. consolidated application dated March 30, 1995."

Attachment B

DISPOSITION OF CERTIFICATE OF COMPLIANCE NO. 9001, REVISION NO. 29 REFERENCED SUPPLEMENTS

1. General Electric supplements dated:

   a. February 8, 1985: Incorporated into NEDO-10084-3.

   b. April 4, 1985: Incorporated into NEDO-10084-3.

   c. May 10, 1985: Incorporated into NEDO-10084-3.

   d. March 12, 1990: 1990 license renewal request.

2. Pacific Nuclear Systems, Inc. supplements dated:

   a. July 26, 1990: See August 30, 1991.

   b. March 28, 1991: See August 30, 1991.

   c. April 12, 1991: See August 30, 1991.

   d. July 19, 1991: Incorporated as Appendix C in NEDO-10084-4 (Outer Plastic Wrap).

   e. August 30, 1991: Incorporated as Appendix A in NEDO-10084-4 (Channelled BWR Fuel Basket).

   f. January 3, 1992: Deleted (Shoreham basket).

   g. Feb. 25, 1993: Deleted (Shoreham basket).

   h. April 9, 1993: Deleted (Shoreham basket).

   i. July 29, 1993: Deleted (Shoreham basket and tarpaulin).

   j. August 10, 1993: Deleted (Shoreham tarpaulin).

3. VECTRA Technologies, Inc. supplement dated:

   a. April 25, 1994: Incorporated as Appendix B in NEDO-10084-4 (High Burnup PWR Fuel).

Attachment C

PAGE INSERTION INSTRUCTIONS TO UPDATE NEDO-10084-3 TO NEDO-10084-4

A. Volume 1 (in front cover inside pocket of new Volume 3 binder)

1. Replace existing outside binder cover insert (1 sheet) with new binder cover insert (1 sheet).

2. Replace existing binder spine (1 piece) with new binder spine (1 piece).

3. Replace existing title/NOTICE AND DISCLAIMER pages (1 sheet) with new title/NOTICE AND DISCLAIMER pages (1 sheet).

4. Replace existing REVISION SUMMARY (1 sheet) with new REVISION CONTROL SHEETs (2 sheets).

5. Replace existing TABLE OF CONTENTS (1 sheet) with new TABLE OF CONTENTS (2 sheets).

6. Section I

   Replace existing pages 1-1 & 1-2 (1 sheet) with new pages 1-1 & 1-2 (1 sheet).

7. Section II

   a. Replace existing pages 2-1 & 2-2 (1 sheet) with new pages 2-1 & 2-2 (1 sheet).

   b. Replace existing pages 2-3 & 2-3a (1 sheet) with new pages 2-3 & 2-3a (1 sheet).

   c. Replace existing pages 2-4 & 2-5 (1 sheet) with new pages 2-4 & 2-5 (1 sheet).

   d. Replace existing pages 2-8 & 2-9 (1 sheet) with new pages 2-8 & 2-9 (1 sheet).

A. Volume 1 (Continued)

7. Section II (Concluded)

e. Replace existing pages 2-10 & 2-11 (1 sheet) with new pages 2-10 & 2-11 (1 sheet).

f. Replace existing pages 2-14 & 2-15 (1 sheet) with new pages 2-14 & 2-15 (1 sheet).

8. Section III

a. Replace existing pages 3-1 & 3-2 (1 sheet) with new pages 3-1 & 3-1a (1 sheet) and 3-2 & 3-2a (1 sheet).

b. Replace existing pages 3-15 & 3-16 (1 sheet) with new pages 3-15 & 3-16 (1 sheet).

9. Section IV

a. Replace existing pages 4-1 & 4-2 (1 sheet) with new pages 4-1 & 4-2 (1 sheet).

b. Replace existing pages 4-3 & 4-4 (1 sheet) with new pages 4-3 & 4-4 (1 sheet).

c. Replace existing pages 4-5 & 4-6 (1 sheet) with new pages 4-5 & 4-6 (1 sheet).

d. Replace existing pages 4-9 & 4-10 (1 sheet) with new pages 4-9 & 4-10 (1 sheet).

10. Section V

a. Replace existing pages 5-1 & 5-2 (1 sheet) with new pages 5-1 & 5-2 (1 sheet).

b. Replace existing pages 5-3 & 5-4 (1 sheet) with new pages 5-3 & 5-4 (1 sheet).

c. Replace existing pages 5-5 & 5-6 (1 sheet) with new pages 5-5 & 5-6 (1 sheet).

A. Volume 1 (Continued)

10. Section V (Concluded)

d. Replace existing pages 5-101 & 5-102 (1 sheet) with new pages 5-101 & 5-102 (1 sheet).

e. Replace existing pages 5-269 & 5-270 (1 sheet) with new pages 5-101 & 5-102 (1 sheet).

11. Section VI

a. Replace existing pages 6-1 & 6-2 (1 sheet) with new pages 6-1 & 6-1a (1 sheet) and 6-2 & 6-2a (1 sheet).

b. Replace existing pages 6-33 & 6-34 (1 sheet) with new pages 6-33 & 6-34 (1 sheet).

c. Replace existing pages 6-35 & 6-36 (1 sheet) with new pages 6-35 & 6-36 (1 sheet).

12. Section VII

Replace existing pages 7-1 & 7-2 (1 sheet) with new pages 7-1 & 7-2 (1 sheet).

13. Section VIII

a. Replace existing pages 8-1 & 8-2 (1 sheet) with new pages 8-1 & 8-2 (1 sheet).

b. Replace existing pages 8-3 & 8-4 (1 sheet) with new pages 8-3 & 8-4 (1 sheet).

c. Replace existing pages 8-19 & 8-20 (1 sheet) with new pages 8-19 & 8-20 (1 sheet).

A. Volume 1 (Concluded)

14. Section IX

a. Replace existing pages 9-1 & 9-2 (1 sheet) with new pages 9-1 & 9-2 (1 sheet).

b. Replace existing pages 9-5 & 9-6 (1 sheet) with new pages 9-5 & 9-6 (1 sheet).

15. Section X

Replace existing pages 10-3 & 10-4 (1 sheet) with new pages 10-3 & 10-4 (1 sheet).

B. Volume 2 (inside back cover sleeve of new Volume 3)

1. Replace existing outside binder cover insert (1 sheet) with new binder cover insert (1 sheet).

2. Replace existing binder spine (1 piece) with new binder spine (1 piece).

3. Replace existing title/NOTICE AND DISCLAIMER pages (1 sheet) with new title/NOTICE AND DISCLAIMER pages (1 sheet).

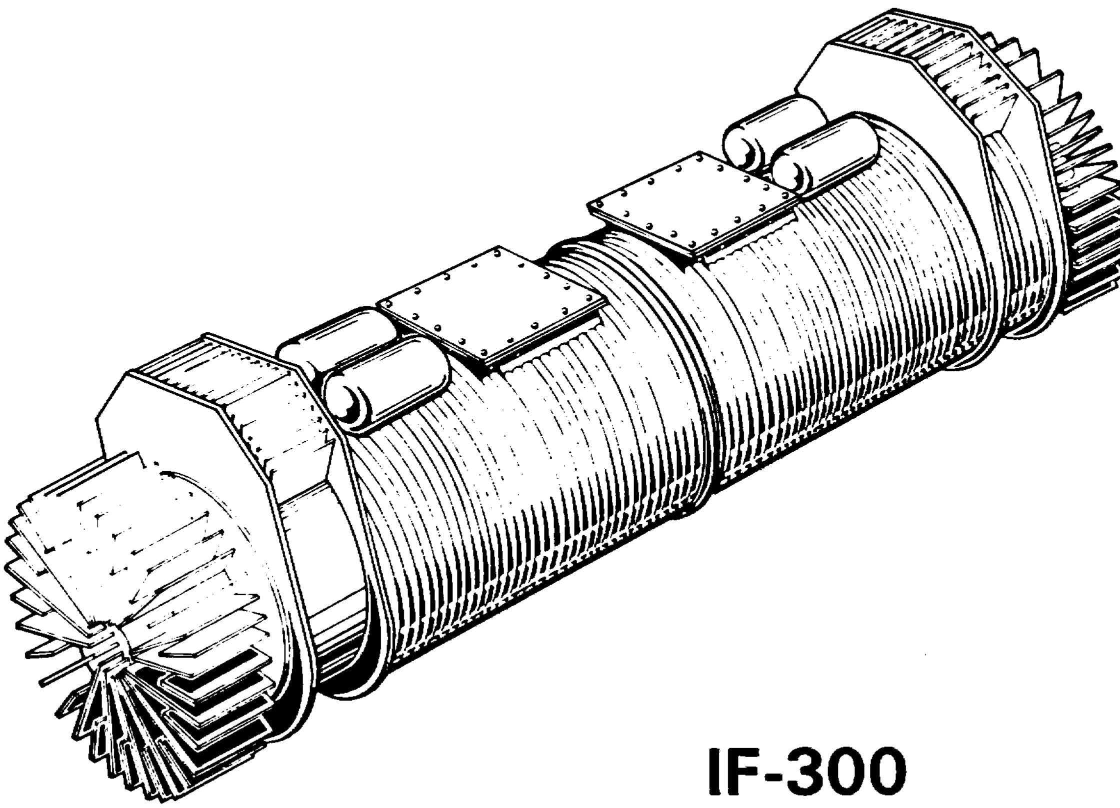
4. Replace existing REVISION SUMMARY (1 sheet) with new REVISION CONTROL SHEET (1 sheet).

5. Replace existing TABLE OF CONTENTS (1 sheet) with new TABLE OF CONTENTS (2 sheets).

C. Volume 3 - Completely new binder with new Appendices A, B, and C.

VECTRA

NEDO-10084-4
MARCH 1995



IF-300 SHIPPING CASK

CONSOLIDATED SAFETY ANALYSIS REPORT

VOLUME
1

9505080266

VECTRA

IF-300 SHIPPING CASK

CONSOLIDATED SAFETY ANALYSIS REPORT

NEDO-10084-4

MARCH 1995

VOLUME 1

NOTICE AND DISCLAIMER

All revision of this report through NEDO-10084-3 were prepared by General Electric Company solely for the use of the U.S. Nuclear Regulatory Commission (NRC) in licensing the IF-300 Shipping Cask. General Electric assumes no responsibility or damage which may result from any other use of the information disclosed in any revision of this report through NEDO-10084-3.

The information contained in revisions of this report through NEDO-10084-3 is believed by General Electric to be an accurate and true representation of the facts known, obtained, or provided to General Electric through May 1985. General Electric Company and the contributors to revisions of this report through NEDO-10084-3 make no express or implied warranty of accuracy, completeness, or usefulness of the information contained in this report with respect to any change of fact or law set forth therein, whether material or otherwise, and General Electric Company makes no warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, other than for the licensing of the IF-300 Shipping Cask or that the use of any information disclosed in this report may not infringe privately owned rights including patent rights.

In 1988, VECTRA became the principal Licensee holder for the IF-300 Shipping Cask. VECTRA is responsible for all changes to this report starting with NEDO-10084-4.

REVISION CONTROL SHEET

TITLE: Consolidated Safety Analysis Report for IF-300 Shipping Cask

DOCUMENT NO.: NEDO-10084

AFFECTED PAGE(S)	DOC. REV.	REMARKS
1-i	3	Last revision prepared by General Electric Company.
1-1 & 1-2	"	Incorporates all C of C 9001, Revision 29 references
2-i & 2-ii	"	from 2/8/84 through 5/10/85.
2-1 - 2-15	"	A vertical line on the right hand margin indicates a
3-i & 3-ii	"	revision. "N" denotes new information while "E"
3-1 - 3-16	"	denotes an editorial change.
4-i & 4-ii	"	
4-1 - 4-21	"	
5-i - 5-vi	"	
5-1 - 5-311	"	
6-i - 6-iv	"	
6-1 - 6-82	"	
7-i - 7-ii	"	
7-1 - 7-22	"	
8-i & 8-ii	"	
8-1 - 8-22	"	
9-i & 9-ii	"	
9-1 - 9-6	"	
10-i & 10-ii	"	
10-1 - 10-15	"	
A-i/A-ii	"	
V1-i - V1-iv	"	
V1-1 - V1-52	"	
V1-A-i/ii	"	
V1-A-1 -	"	
V1-A-3	"	
V1-B-i/ii	"	
V1-B-1 &	"	
V1-B-2	"	

PAGE 1 OF 3

REVISION CONTROL SHEET

TITLE: Consolidated Safety Analysis Report for IF-300 Shipping Cask

DOCUMENT NO.: NEDO-10084

AFFECTED PAGE(S)	DOC. REV.	REMARKS
V1-C-i/ii	3	
V1-C-1 -	"	
V1-C-8	"	
V1-D-i -	"	
V1-D-vi	"	
V1-D-1 -	"	
V1-D-132	"	
V1-E-i &	"	
V1-E-ii	"	
V1-E-1 -	"	
V1-E-34	"	
V2-i - V2-iv	"	
V2-1 - V2-64	"	
V3-i - V3-iv	"	
V3-1 - V3-32	"	
VI-i/VI-ii	"	
VI-1 - VI-6	"	
i - viii	4	First revision prepared by VECTRA Technologies, Inc.
1-1	"	Incorporates C of C 9001, Revision 29 references
2-1 & 2-2	"	from 7/26/90 through 4/25/94.
2-3 & 2-3a	"	A vertical line on the left hand margin indicates a
2-4 & 2-5	"	revision.
2-8	"	
2-10 & 2-11	"	
2-14 & 2-15	"	
3-1 & 3-1a	"	
3-2a	"	
3-16	"	
4-1, 4-3, & 4-5	"	
4-9 & 4-10	"	

PAGE 2 OF 3

REVISION CONTROL SHEET

TITLE: Consolidated Safety Analysis Report for IF-300 Shipping Cask

DOCUMENT NO.: NEDO-10084

AFFECTED PAGE(S)	DOC. REV.	REMARKS
5-1, 5-3, & 5-5	4	
5-101 & 5-270	"	
6-1 & 6-1a	"	
6-2 & 6-2a	"	
6-33 & 6-35	"	
7-1 & 7-2	"	
8-1 & 8-3	"	
8-19 & 8-20	"	
9-1, 9-2, & 9-6	"	
10-4	"	
A-i/ii	"	
A-iii - A-xiii	"	
A-1-1 - A-1-10	"	
A-2-1 - A-2-338	"	
A-3-1 - A-3-86	"	
A-4-1 - A-4-5	"	
A-5-1 - A-5-92	"	
A-6-1 - A-6-131	"	
A-7-1 & A-8-1	"	
A-9-1 - A-9-13	"	
B-i/ii	"	
B-iii & B-iv	"	
B-1 - B-55	"	
C-i/ii	"	
C-iii/iv	"	
C-1 - C-14	"	

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TABLE OF CONTENTS

VOLUME 1

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VOLUME 1
(Concluded)

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VOLUME 2

TABLE OF CONTENTS
(Concluded)

VOLUME 3

I. INTRODUCTION

1.1 PURPOSE

Prior to 1988, General Electric was the principle Licensee holder for the IF-300 shipping cask. Starting in 1988, VECTRA became the principle Licensee holder for this system. All references to General Electric as the principle Licensee holder in this report should currently be understood to be referring to VECTRA.

This Consolidated Safety Analysis Report (CSAR) represents the technical basis for Certificate of Compliance (C of C) Number 9001, including revisions, for the IF-300 shipping cask. This CSAR can be amended by VECTRA through the submittal of changes and/or additions which must be reviewed and accepted by the United States Nuclear Regulatory Commission (NRC).

Originally, the authorized contents of the IF-300 cask was restricted to Group I type fuel bundles, which included 14 x 14 and 15 x 15 fuel rod arrays for PWR bundles and 7 x 7 fuel rod arrays for BWR bundles.

In 1982, the C of C was amended to authorize shipment of Group II fuel bundles which had begun to replace the Group I fuel bundles in operating reactors. The Group II fuel bundles have a larger number of smaller diameter fuel rods. Group II fuel bundles have a larger number of smaller diameter fuel rods. Group II includes the 16 x 16 and 17 x 17 fuel rod arrays for PWRs and the 8 x 8 fuel rod array for BWRs.

Shipment of solid, non fissile, irradiated hardware was authorized in a 1984 amendment.

Shipments of BWR fuel with channels in a 17 element fuel basket (Volume 3, Appendix A) was authorized in a 1991 amendment and shipments of high burnup PWR fuel (Volume 3, Appendix B) was authorized in a 1994 amendment. The use of an outer plastic wrap to contain "weeping" was also authorized in a 1991 amendment (Volume 3, Appendix C).

1.2 IF-300 CASK

The IF-300 cask is designed to meet or exceed all NRC and

Department of Transportation (DOT) regulations governing the shipment of radioactive material. The primary transportation mode is by railroad, although the shipping package is designed to facilitate truck shipment on a special overweight basis for short distances. This features allows the servicing of reactor sites and other facilities which lack direct railroad access.

The IF-300 cask body is a depleted uranium shielded, stainless steel clad annular cylinder, closed at one end. Fuel is loaded into the cask through the open end and the cask is closed with a bolted and sealed head. The head construction is similar to the body of the cask.

Fuel bundles are located within the cask cavity by a removable stainless steel basket. There are several basket configurations which may be used, depending on the specific fuel being shipped. There are also two heads which permit a variation in cask cavity length. When solid, nonfissile, irradiated hardware is shipped, it is placed within a non-reusable steel liner liner built specifically for that hardware. The cask cavity is air-filled and utilizes a rupture disk device for over pressure protection.

The cask outer surface has large circumferential fins designed for impact protection. Encircling the active fuel zone is a water-filled annulus with corrugated jacket which acts as a neutron shield. The upper and lower ends of the cask are also equipped with sacrificial fins for impact protection.

The cask is cooled, when desired, by diesel engine driven blowers which maintain outer surfaces at temperatures facilitating handling. The cooling system is not required to preserve cask integrity or retain coolant. Four longitudinal ducts direct air from two blowers onto the corrugated surface.

The cask, cask supports, and cooling system are all mounted on a steel skid. Exclusion from the cask and cooling system is provided by a wire mesh enclosure which is retractable for cask removal and locks in place during transport. The skid mounted equipment forms a completely self-contained irradiated fuel and hardware shipping package.

II. DESIGN SUMMARY

2.1 CASK DESCRIPTION

The VECTRA IF-300 Spent Fuel Shipping Cask is designed in accordance with the criteria of Federal Regulations 10CFR71 and 49CFR173.

Prior to 1994, the fuel loadings which could be contained in the IF-300 were as follows:

Table II-1

FUEL LOADINGS

Reactor Type	NSSS Manufacturer	No. of Bundles	Fuel Rod Array	Cladding Material	Fuel Group
BWR	General Electric	18	7 x 7	Zircaloy	I
PWR	Westinghouse	7	14 x 14	Stainless Steel	I
	Westinghouse	7	14 x 14	Zircaloy	I
	Westinghouse	7	15 x 15	Stainless Steel	I
	Westinghouse	7	15 x 15	Zircaloy	I
PWR	Combustion	7	15 x 15	Zircaloy	I
PWR	Babcock-Wilcox	7	15 x 15	Zircaloy	I
BWR	General Electric	18	8 x 8	Zircaloy	II
PWR	Westinghouse	7	17 x 17	Zircaloy	II
PWR	Combustion	7	16 x 16	Zircaloy	II
PWR	Babcock-Wilcox	7	17 x 17	Zircaloy	II

Since 1994, the IF-300 cask has also been permitted to contain 15 x 15 PWR fuel with a maximum burnup of up to 45,000 MWd/MTU with a minimum cooling time of 60 months as described in Volume 3, Appendix B.

Either BWR or PWR fuel bundles can be accommodated through the use of removable fuel baskets, spacers and two different length closure heads. In addition to irradiated fuel bundles, the IF-300 cask may be used to transport solid non-fissile irradiated hardware.

The cask weight when loaded is between 130,000 and 140,000 pounds depending on the particular type of fuel being shipped. The skid and cooling system weigh approximately 45,000 pounds.

The cask is mounted horizontally on an equipment skid during transport. Although transportation is primarily by rail, the skid is designed to accept wheel assemblies for short haul, special permit trucking. This dual-mode shipping configuration permits the use of the IF-300 cask at those reactor sites which have no direct rail access.

The cask is supported on the skid by a saddle at the head end and a cradle at the bottom end. The cradle forms the pivot about which the cask is rotated for vertical removal from the skid. There is one pickup position on the cask body just below the closure flange. The support saddle engages the cask at this section. The lifting trunnions are removed during transport. The pivot cradle trunnions are slightly eccentric to ensure the proper rotation direction for cask lay-down. The cradle is counter-weighted to remain horizontal when the cask is removed.

The cask is lifted by one of two special yokes, a normal unit and a redundant unit. Either yoke accepts the reactor building crane hook in its upper end and engages the cask lifting trunnions with its lower end. Each yoke is designed to be used with either head. The cask head is removed using four steel cables which are attached to the lifting yoke. The same yoke is used for cask uprighting and cask lifting.

All external and internal surfaces of the cask are stainless steel. The outer shell of the cask body is CG-8M (317) stainless steel. The inner shell is 317 or 216 stainless steel. The circumferential fins are 216 stainless steel and the flanges and end fins are 304 stainless steel. The fuel baskets are made of 216 and 304 stainless steel.

Gamma and fast neutron shielding, respectively, are provided in the IF-300 cask by depleted uranium metal between the cask shells and a water/ethylene glycol mixture filled annulus surrounding them. The thin walled jacket which retains the neutron shielding water is fabricated from stainless steel and is corrugated for maximum strength and heat transfer.

The closure head is sealed with a Grayloc metallic ring. The cavity maximum normal operation pressure (LOMC) is 29 psig. However, the design working pressure is 400 psig and overpressure protection is provided by a rupture disk device designed to have a bursting pressure of 350-400 psig at 443 degrees fahrenheit. The rupture disk device is located in one of two cavity valve boxes.

Each cavity valve box is equipped with one nuclear service fill, drain, and vent valve. For ease in servicing, these valves have a quick disconnect fitting which may, as an option, be replaced by a stainless steel pipe cap or pipe plug during cask shipment.

The neutron shielding annulus is partitioned into two separate sections, each protected from overpressure by a 200 psig relief valve located in one of two neutron shielding valve boxes. Service to each section is provided annually through fill, drain, and vent valves also located in the neutron shielding valve boxes. These valves may be replaced by a stainless steel blind flange. All valve handles and/or blind flange bolts are lockwired during transit to prevent loosening.

A thermocouple well is attached to the outside of the inner shell at a point expected to be at the highest temperature. The thermocouple well emerges from the cask bottom and accepts a replaceable chromelalumel thermocouple.

The fuel bundles are contained within a removable, slotted, stainless steel basket. For the fuel baskets licensed prior to 1991, criticality control is achieved by using B_4C-filled, stainless steel tubes installed in the basket. For the channelled BWR fuel basket licensed in 1991, criticality control is achieved by using borated stainless steel poison plates (Volume 3, Appendix A). Fuel bundles are restrained axially by spacers mounted on the inside of the closure head or in the bottom of the fuel basket. The basket is centered within the cask cavity by disk spacers. Nine such spacers are mounted along the fuel basket length. Fuel bundles are inserted and removed from the basket using standard grapples. The basket is removed when the cask is to be used for the shipment of another fuel type or for cask cavity cleaning. the BWR basket has stainless steel clad uranium shielding pieces mounted on the end adjacent to the cask flange.

The outer surface of the cask body is finned for impact protection. These fins are stainless steel and are circmuferential to the cask diameter. The cask ends and valve boxes are also finned for impact protection. All fins are welded to the cask body. The external water jacket is constructed of thin-walled material and does not contribute to the impact protection of the cask.

2.2 STRUCTURAL ANALYSIS

2.2.1 The IF-300 cask is specifically designed to meet the structural requirements of 10CFR71. Safety factors are based on allowable loads, stresses and deflections. For some components yield strength of the material is the limiting parameter; for others, ultimate strength is the true limit.

In general, for normal or slightly off-normal conditions material yield is the basis for the safety factor. It is usually under accident conditions where some components yield and take a permanent set. Under these conditions integrity of the component is the primary concern. Since almost all of the cask components are of austenitic stainless steel having good ductility, there is a significant difference between the stress required to yield the material and that needed to actually fail (break) it. Thus, for certain loadings and components ultimate strength is the safety factor basis.

2.2.2 Stress analyses have been performed for the following conditions:

- A combined 10 g axial, 5 g lateral and 2 g vertical load on the cask-to-skid tiedowns.
- With the cask acting as a beam supporting five times its weight.
- With an external pressure of 25 psig applied to the cask.
- Accidents, which include a 30 foot drop in various attitudes, a 40 inch puncture, and a 30 minute fire.
- Cask handling, in both unloading and loading operation.

- Off-normal conditions including loss-of-mechanical cooling, partial loss-of-shielding water and vandalism.

The normal transport analysis demonstrates the ability of the cask and its tiedowns to sustain both internal and external loads and maintain a yield-based safety factor greater than unity.

The cask uprighting and lifting analysis shows that no component stress level exceeds the material yield strength.

The accident analysis is divided into several parts: 1) the cavity as a pressure vessel; 2) the cask as a structure; and, 3) the cask contents.

1. The cavity sees its maximum internal pressure under accident conditions. The cavity component stresses do not exceed yield.

2. The cask body undergoes severe loading in all of the 30 foot drop orientations. There is slight flange yielding in the corner drop and slight outer shell yielding in the side drop. However, the cask remains sealed with no significant reduction in gamma shielding.

3. Group I and Group II fuel bundles were independently analyzed for the 30 foot drop. The analytical results were similar. The fuel and fuel basket undergo severe loading in the 30 foot drop and some yielding occurs. However, the extent of yielding of the fuel basket structure is limited by the short duration of the loading and the confinement of the cavity walls.

Based on allowable limits, no fuel or basket failures occur in the 30-foot drop.

II. DESIGN SUMMARY

2.3 THERMAL ANALYSIS

2.3.1 Design Basis Conditions

Five basic conditions of operation were analyzed. The characteristic features of each of these conditions are summarized in Table II-4. All the analyses are for a heat load of 40,000 Btu/hr and dry shipments.

Table II-4

CHARACTERISTICS OF CONDITIONS ANALYZED

Operating Condition

Parameter	Cooling	LOMC	50% SWL	30-Minute Fire	PFE
Mechanical Cooling	Yes	No	Yes	No	No
Neutron Shielding Cavity Contents	Water	Water	Water/Air	Air	Air
Solar Heat Input	No	No	No	No	Yes
Ambient Temp, °F	130	130	130	1475	130

2.3.2 Results of Design Basis Analyses

The results obtained from analyzing the above noted conditions are summarized in Table II-5 for fuels licensed prior to 1991 (for fuels licensed since 1991, see Volume 3, Appendices A and B). They are based on the use of the thermal analysis code THTD to obtain cask temperature distributions and the Wooten-Epstein correlation for a dry, air-filled, horizontal cask to obtain cladding temperatures.

Table II-5

RESULTS OF THERMAL ANALYSES

Parameter		Normal Cooling	LOMC	50% SWL	30-Minute Fire	PFE
Ambient Temp, °F		130	130	130	1475	130
Heat Load Btu/hr		40,000	40,000	40,000	40,000	40,000
Max Barrel Temp, °F		155	213	173	1274	228
Max Outer Shell Temp, °F		163	219	284	452	369
Max Inner Cavity Surface Temp, °F		173	229	292	353	377
Hottest Rod Max Temp, °F						
Gp.1	7x7 BWR	492	537	587	635	654
	15x15 PWR	503	549	601	651	670
Gp.2	8x8 BWR	498	544	595	643	662
	7x17 PWR	508	555	607	658	677
Inner Cavity Pressure, psig		14	29	70	152	267

2.3.3 Miscellaneous Thermal Conditions

The following miscellaneous thermal conditions are considered in Section VI:

- Cask operation at -40°F
- Effects of antifreeze on cask operation
- Thermal expansion of neutron shielding liquid
- Effects of residual water on cavity pressure

2.3.4 Pressure Relief and Drain, Fill, Vent Devices

a. Ruptive Disk Device: A rupture disk device designed to burst at 350-400 psig at 443°F is used to provide overpressure protection to the cask inner cavity.

b. 200 psig Pressure Relief Valve: This valve provides overpressure protection to the neutron shielding cavities.

c. 1-Inch Globe Valve: Valves of this type are used for draining, filling, and venting of the cask inner cavity and, optionally, the neutron shielding cavities.

These components are described in greater detail in Section 6.

2.3.5 Thermal Testing of the Cask

Section 6.8 discusses the details of the thermal test procedures, cask thermal acceptance criteria, and the results of tests on casks 301 through 304. The data obtained from these tests was used to determine the maximum permissible wet shipment load for each cask and to "calibrate" the thermal model.

The difference in maximum permissible wet shipment heat load between casks 301 through 304 was less than 10%. For dry shipment all casks are rated at 40,000 Btu/hr. maximum heat load.

In addition to above described beginning-of-life thermal tests, each cask has temperature measurements taken while in use. These measurements are reviewed and evaluated on an annual basis to determine if there has been any degradation in the casks ability to dissipate heat.

2.4 CRITICALITY ANALYSIS

Table II-6 summarizes the most reactive criticality conditions for the reference fuels and the configurations indicated licensed prior to 1991 (Volume 3, Appendices A and B provide details for fuels licensed since 1991). In both the BWR and PWR cases, the use of criticality control members is necessary. For fuels licensed prior to 1991, these are in the form of boron carbide-filled stainless steel tubes (as opposed to borated stainless steel poison plates described in Volume 3, Appendix A) fixed to the fuel basket components. These rods are patterned after the BWR control blade elements. Boron density is 1.75 gm/cc. The poison locations are shown in Section VII.

Table II-6

MAXIMUM k_{eff} VALUES

Fuel Type	No. of Bundles	Enrichment w/o U^{235}	k_{eff}
BWR	18	4.0	0.880
PWR	7	4.0	0.955

An infinite array of casks in air with no spacing between the casks raises the k_{eff} a very small amount thus classifying the cask fissile Class I.

Prior to making a determination of cask k-effective (k_{eff}) it was necessary to compute the most reactive fuel bundle geometry. This was done by varying rod pitch within the confines of the corresponding basket channel. To determine maximum cask k_{eff} the peak bundle geometries were placed in the appropriate cask array models and k_{eff} was computed as a function of cask cooling temperature. Peak cask reactivity is at 20°C.

2.5 SHIELDING ANALYSIS

The analysis only considers the case of 7 PWR bundles with an exposure of 35 GWD/T, a specific power of 40 kw/kg, and a 120-day cooling time. This represents a "worse" case loading of any reference fuel licensed prior to 1991 for which the cask was originally designed. Both gamma and fast neutron radiation must be considered in designing a shipping package for high exposure light water moderated reactor fuels.

2.5.1 Gamma Shielding

The gamma source arises from the decay of the radioisotopes created from the fission process during reactor operation. The source strength is a function of specific power, operating time, and cooling time. Section VIII describes the source term in detail for fuels licensed prior to 1991. Volume 3, Appendices A and B provide details for fuels licensed since 1991. Depleted uranium metal is the principal gamma shield in the IF-300 cask, although there is a significant contribution from the stainless steel inner and outer shells. The uranium is an annular casting four inches thick clad in stainless steel, and forms the cask body. Head end shielding is accomplished with three inches of stainless-clad uranium. The bottom end requires three and three-quarters of an inch of uranium.

2.5.3 Calculational Results

49CFR173 prescribes the allowable dose rates as 10 mr/hr total radiation at a point 6 feet from the vertical projection of the outer edges of the transport vehicle. Furthermore, DOT and NRC regulations specify a limit of 1 R/hr three feet from the cask surface following the hypothetical accident conditions. Table II-7 indicates that the IF-300 cask shielding meets both normal and accident shielding requirements.

2.6 FISSION PRODUCT RELEASE

For the reduced heat load of dry shipments in the IF-300 cask, the analyses of Section VI show that there is no release of any of the cask contents to the environs for either normal or accident conditions.

2.7 REGULATIONS

The IF-300 irradiated fuel shipping cask is designed to meet both the normal transport and accident conditions of the NRC and DOT. Section IX summarizes the design results in light of these regulatory criteria.

2.8 OPERATION AND MAINTENANCE

2.8.1 Operation

A complete operating manual has been written and is provided to each cask user. In addition, VECTRA offers training on cask handling prior to use. VECTRA-supplied technical assistance will also be offered in support of cask handling. The user is expected to provide all operating and health physics personnel. The user will bear the responsibility of proper cask operations.

Each shipping package will have in-transit instructions which provides guidance to the carrier personnel in the event of an abnormal condition.

2.8.2 Maintenance

Maintenance and repair of the IF-300 cask will be performed following written instructions. The same level of quality specified for the initial fabrication will be applied to maintenance and repair items. Where applicable manufacturer's recommendations or accepted industry standards will be followed. Records will be maintained on a cask-by-cask basis in accordance with regulatory requirements. An approved Quality Assurance Plan will be applied to all items of maintenance and repair.

2.9 FABRICATION AND QUALITY ASSURANCE

Since it is necessary to have some uniform and familiar set of criteria to govern equipment fabrication, General Electric had chosen the ASME Nuclear Vessel Code, Section III as guidance for the IF-300 cask fabrication and quality control. The design portion of Section III is excluded, due to the unique requirements of shipping casks.

All IF-300 basic components identified in Chapter IX are designed, fabricated, tested, used, and maintained under an NRC approved quality assurance program that satisfies requirements in 10CFR71 Subpart H "Quality Assurance" criteria for packaging and transportation of radioactive material.

3.1 INTRODUCTION

The IF-300 spent fuel cask is designed as a general purpose shipping container. With its various length heads and removable fuel baskets, the IF-300 is capable of servicing all of the present and planned light-water moderated power reactors which have a building crane capacity of greater than 70 tons.

The fuels are segregated into two generic groups, BWR and PWR. Within each group is a design basis or reference fuel bundle. This assembly is a composite of parameters based on the present and projected fuel designs of General Electric, Westinghouse, Babcock & Wilcox, and expected values for present generation power reactors. These critical parameters include exposure, specific operating power, enrichment critical parameters include exposure, specific operating power enrichment, uranium content, active length, and bundle cross-section geometry. All of these are necessary inputs to a shipping cask design analysis.

Prior to 1991, the IF-300 cask was designed to ship either eighteen (18) of the BWR reference fuel bundles or seven (7) of the PWR reference fuel bundles. Since 1991, the IF-300 cask has also been permitted to ship seventeen (17) channelled BWR fuel bundles. This approach permits the shipment of any BWR or PWR fuel without specific analysis as long as it is within its respective design basis envelope.

The reference fuels and their bases licensed prior to 1991 are summarized in Table III-1. Fuel designs licensed in 1991 for the channelled BWR fuel basket are summarized in Volume 3, Appendix A, page A-1-9, Table A-1.2-2. PWR fuels with a maximum burnup of 45,000 MWD/MTU licensed in 1994 are described in Volume 3, Appendix B.

3.2 FISSION PRODUCT ACTIVITIES AND POWERS FOR THE DESIGN BASIS FUELS[1]

This section addresses Design Basis fuel licensed prior to 1991. Volume 3, Appendices A and B address fuel licensed for a 17-cell channelled BWR fuel basket in 1991 and 15x15 Westinghouse PWR f... licensed with a maximum burnup of up to 45,000 MWd/MTU with a minimum cooling time of 60 months in 1994, respectively.

3.2.1 Design Basis for PWR Bundles

3.2.1.1 Determination of the Thermal Neutron Flux:

As a basis for design, the specific power is taken to be 40 kw/kgU, and the burn-up to be 35,000 MWd/MTU. The irradiation time is:

1 Calculations by Charles B. Magee; Denver Research Institute, Denver, Colorado. Results of calculations used in Table III-1.

Table III-1

CHARACTERISTICS OF EXPECTED AND DESIGN BASIS FUELS

Reactor	Type	Mfg.	Enrich. (%)	Exposure (Gwd/MTU)	Op. Pwr. (kw/kgU)	Active Length (in.)	Overall Length (in.)	Slot Envelope (in.)	Rod Array	Rod Pitch (in.)	Rod Dia. (in.)	kgU/ Assy.	Clad Mat'l	Clad Th'k (in.)
Oyster Cr. 1	BWR	GE		15	14.5	144	171 5/8	5.440	7x7	0.738	0.570	197	Zr-2	0.036
Monticello 1	BWR	GE	2.00	15.0	15.8	144	171 7/32	5.47	7x7	0.738	0.570	197	Zr-2	0.0355
Browns Ferry 1, 2, 3	BWR	GE	2.19	30.0	22.0	144	176 7/32	5.47	7x7	0.738	0.562	195	Zr-2	0.032
Group I Design Basis BWR		-	4.0	35.0	30.0	146	-	5.75	7x7	0.647 to 0.809	0.500 to 0.600	198	Zr-2	0.029 min.
Haddam Neck 1	PWR	W			21.0	120	137.66	8.420	15x15	0.553	0.422	433	SST	0.016
Turkey Pt. 3, 4	PWR	W	2.73	27.0	29.3	144	160.10	8.426	15x15	0.563	0.422	457	Zr	0.0243
H. B. Robinson 1	PWR	W	2.73	27.0	29.0	144	160.10	8.426	15x15	0.563	0.422	457	Zr	0.0243
Indian Pt. 2, 3	PWR	W	2.92	27.0	32.0	144	161.38	8.426	15x15	0.556	0.422	447	Zr	0.0243
Diablo Canyon 1	PWR	W	2.67	33.0	36.9	144	160.10	8.426	15x15	0.563	0.422	457	Zr	0.0243
Palisades 1	PWR	CE	2.74	24.0	24.7	132	151	8.323	15x15	0.553	0.413	437	Zr-4	0.024
Calvert Cliffs 1	PWR	CE	2.74	30.0	28.0	132	151	8.323	14x14	0.553	0.44	437	Zr-4	0.026
Rancho Seco 1	PWR	B&W	3.09	28.2	31.8	144	165 5/8	8.522	15x15	0.568	0.430	456	Zr-4	0.026
Oconee 1, 2, 3	PWR	B&W	3.09	28.2	31.8	144	165 7/8	8.522	15x15	0.552	0.420	456	Zr-4	0.026
Point Beach 1	PWR	W	3.05	27.0	27.6	144	161.275	7.803	14x14	0.556	0.422	392	Zr	0.0243
San Onofre 1	PWR	W	3.86	30.0	21.0	120	137.66	7.813	14x14	0.556	0.402	375	SST	0.013
R. E. Ginna 1	PWR	W	3.05	27.0	27.6	144	161.275	7.803	14x14	0.556	0.422	392	Zr	0.0243
Group I Design Basis PWR		-	4.0	35.0	40.0	145	-	8.75	14x14 15x15	0.502 to 0.582	0.380 to 0.460	465	SST Zr	0.013 min. 0.020 min.
Group II Design Basis BWR		-	4.0	35.0	30.0	150	-	5.75	8x8	0.630 to 0.645	0.475 to 0.505	197	Zr-2	0.029 min.
Group II Design Basis PWR		-	4.0	35.0	40.0	144 to 150	-	8.75	16x16 17x17	0.496 to 0.507	0.374 to 0.400	475	Zr	0.020 min.

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Again, discounting gross bundle distortion which is rare, the bundle or cask k_∞ or k-effective is independent of cladding integrity.

From the standpoint of bundle structural integrity, again, the expected types of failures are generally insignificant since they are localized and small in size and number. There may be some further damage to the failed areas as a result of the 30-foot drop loadings but gross loss of bundle integrity will not occur.

3.5.2.3 Releases from Cask

Since the primary concern over failed fuel is what will be released under accident conditions, two things should be recognized:

(1) Each pin hole (or larger) failure has vented the accumulated fission gas in that rod long before the fuel is shipped. From the standpoint of fission gas release under accident conditions, the safest fuel to ship is that with all pins vented.

(2) Cask coolant activity is a function of fuel external contamination and anything which might escape through a failure opening.

A failed rod has operated in that condition for some time followed by a minimum of 1 year of cooling in a storage pool. It is obvious that by shipping time the failure points will have been greatly depleted of those soluble fission products which would contribute to coolant activity.

Most fuel shippers have observed that coolant water activity primarily comes from fuel external contamination ("crud") and that the contribution from leaking fuel is practically masked when performing gross measurements. This is because the very low solubility of the fuel material in water puts a practical upper limit on the quantity of fission product activity in the coolant even if the pellets were completely exposed. Since the fission products generally must migrate through pin holes or small cracks, the rate of activity buildup in the coolant is exceedingly small.

3.6 NON-FUEL CONTENTS

3.6.1 Poison/Criticality Control Components

PWR and BWR fuel bundles occasionally contain burnable poison rods or reactivity control rods which are non-fuel bearing components. These items do not contribute measurably to the decay heat or shielding requirements of the cask and thus may be included as cask contents. No analysis of these benign components is required.

3.6.2 Residual Contamination

In the course of operation the interior cavity and fuel basket of a cask becomes contaminated with radioactive material. This residual material consists of mixed fission and activation products. Although the quantity is not always known, experience has shown that it is likely to be greater than a Type A quantity. This material may be present in both the empty and loaded cask.

3.6.3 Irradiated Hardware

Solid non-fissile irradiated hardware contained within a steel liner specifically designed for that hardware may be shipped. Limitation on weight, maximum decay heat in the package and external radiation dose rates are the same as for spent fuel shipments, except the fissionable material dose shall not exceed a Type A quantity and shall not exceed the mass limits of 10CFR71.53.

IV. EQUIPMENT DESCRIPTION

4.1 SHIPPING PACKAGE

General Electric drawing 159C5238 sheets 1 and 2 show the IF-300 shipping package general arrangement[1]. The package consists of five major groups: 1) The cask, part 2; 2) The tiedowns, parts 3, 18, and 19; 3) The cooling system, parts 4, 5, and 6; 4) The skid, part 15; and, 5) The enclosure, parts 7 through 14. It should be noted that some of the drawing call-outs are for information purposes only rather than to denote a safety-related material, item or dimension.

4.1.1 Cask

4.1.1.1 Body

Drawing 159C4238 sheet 4 is a cross section of the cask. The inner cavity is encircled by a 317 or 216 stainless steel cylinder 37-1/2 inches inside diameter with a one-half inch thick wall. The bottom end of the cavity is sealed with a 1-1/4 inch thick 304 stainless steel plate. The upper end is welded to the 304 stainless steel forged closure flange.

Surrounding and shrink-fitted to the inner cavity is the depleted uranium shielding material. This heavy metal assembly consists of eight or ten annular castings, each with a 38-1/2 inch ID and a four inch thick wall. The segments are approximately 16 to 20 inches long. They fit end-to-end, using an overlapping joint to prevent irradiation streaming. The shrink-fitting of shielding to cavity assures good contact for the transmission of heat. Bottom end shielding uses a 3-3/4 inch thick uranium casting.

1 General Electric drawings, which are currently controlled by VECTRA, are located at the end of this section.

To prevent the formation of a low melting point alloy of steel and uranium, a minimum four mil thick copper diffusion barrier exists at every uranium-steel interface. The barrier is flame sprayed on the larger pieces. Copper foil or powder may be used in some of the smaller areas. In some welded areas a copper plated back-up strip is used.

The cask outer shell is a 317 stainless steel cylinder with a 46-1/2 inch ID and a 1-1/2 inch thick wall. This outer shell is shrink-fitted to the uranium castings thus forming a composite or laminated vessel.

The cylindrical portion of the cask is encircled by a thin-walled, corrugated, stainless steel water jacket. This jacket extends axially from the upper valve box to a point slightly above the cask bottom, thus masking the active fuel zone. The water or ethylene glycol and water mixture contained in this cavity functions as a neutron shield. The jacket surface is corrugated for heat transfer purposes. The use of continuous corrugation also provides a surface which is easily decontaminated. The jacket is partitioned at the cask midlength to form two independent cavities, each rated at 200 psig and equipped with a pressure relief valve and a fill, drain, and vent valve or blind flange. Each cavity has a pair of liquid expansion tanks mounted to assure that there will be no loss from the system under the most limiting conditions. N E

There are four large valve boxes on the exterior of the cask body, two for the neutron shielding cavity, two for the cask cavity. The latter two are nested in the upper and lower pairs of structural rings and have lids and side members which are finned for impact and puncture protection. The lids are removable to provide access.

The upper (head-end) cask cavity valve box contains a one-inch stainless steel bellows seal globe valve and a rupture disk E device for overpressure protection (see Section VI for details). The lower cavity valve box contains a one-inch stainless steel bellows globe valve. Both globe valves are equipped with quick-disconnect fittings or SST caps/plugs on their outlet sides.

The cask body and valve boxes are protected from side impact by four 1-1/4 inch thick 216 stainless steel structural rings. These members are also used to support the water jacket sections. The IF-300 cask is lifted by a set of trunnions located just below the closure head flange. These items are pinned to the upper pair of structural rings (see 159C5238 Sheet 5), and are designed to be removed for transit. The upper pair of lifting rings also act as the forward support/axial restraint when the cask is in the horizontal transport position.

The lower end of the cask is equipped with 32 radially mounted impact fins. These fins are 304 stainless steel, slightly more than 1" thick. Sixteen fins are 8-1/8 inches high. The remaining sixteen are 6 inches high. All fins are welded in place.

Temperature monitoring is performed with a chromel-alumel thermocouple contained in a well entering from the bottom of the cask, located equidistant from the ends of the cask body at what is expected to be the hottest axial point.

The overall length of the cask body from fins to flange face is 184-3/16 inches. The cask cavity depth from the flange face is 169-11/16 inches. The flange face contains 32 equally spaced studs each of which is 1-3/4 inches in diameter with 8 threads per inch. The studs protrude 6-1/2 inches from the face and are made of 17-4 PH H1075 stainless steel. the flange itself is an ASTM A-182 type 304 stainless steel machined forging. There are two stainless steel guide pins which protrude above the flange plane for initial head alignment.

4.1.1.2 Cask Heads

The IF 300 cask can be equipped with either of two heads. These heads provide two different cask cavity lengths to match the particular fuel being shipped. With the short head in place the overall cavity length is 169-1/2 inches. The long head increases the cavity to 180-1/4 inches. All PWR fuel to data can be shipped using the short (PWR) head. The longer BWR fuel necessitates the use of an extended (BWR) head.

Shielding in the heads consists of 3 inches of uranium. The outer shell and flange is a single 304 stainless steel machined casting. A circular 304 stainless steel plate is welded in place to form the inner liner and the head cover. As in the case of the body, each steel-uranium interface is isolated with a 4 mil (minimum) copper layer.

Each head has 32 radially mounted fins on the end. Sixteen fins protrude 9-1/2 inches from the surface, the remaining fins protrude 6 inches. These fins are designed to offer impact protection to the cask and contents. The fins are 304 stainless steel and are welded in place.

Due to variations in fuel lengths, it is necessary to provide some spacing scheme. There are a total of five spacer assemblies for the two heads. These spacers are mounted on circular plates which bolt to the top of the head cavity. Spacing is accomplished with struts and pads which protrude from the circular plate. Each plate is numbered and indexed to ensure proper installation. For certain BWR and PWR fuels a spacer is used in the bottom of each basket channel in addition to the head spacers. These bottom spacers elevate the fuel assemblies such that they can be easily engaged by the fuel handling grapple for removal. A special tool is used for bottom spacer installation and withdrawal. See drawing 159C5238, Sheet 7 for a description of the spacers.

4.1.1.3 Closure

The cask body and either head are joined together using the 32 studs in the body flange and an equal number of special sleeve nuts. When the PWR head is being mounted, the short, 3½ inch-length, nuts secure it. Due to its greater length, the BWR head must utilize 13-3/4-inch-long sleeve nuts. Using the sleeve nut approach makes it possible to inter change heads without changing the studs. Two guide pins provide alignment and orientation.

Cask sealing is accomplished using a Grayloc metallic ring as shown on 159C5238 Sheet 5. The head and body flanges interlock to provide shear steps to protect the seal during impact. The seal will sustain a minimum test pressure of 600 psi at room temperature.

4.1.1.4 Fuel Baskets

There are three different fuel baskets which can be used in the IF-300 cask: a 7-cell PWR unit, an 18-cell BWR unit, and a 17-cell channelled BWR fuel unit. The 7- and 18-cell baskets are discussed in this section and are illustrated on 159C5238 Sheet 6. The 17-cell basket with its borated stainless steel poison plate design is discussed in Volume 3, Appendix A. The 7-cell basket holds the various PWR bundles and the 18-cell basket holds the various BWR fuel bundles presented in Section III for fuel description. Poison rods containing B_4C effectively cover the 7-cell PWR length 159.8 inches and the 18-cell BWR length of 167.6 inches for criticality control.

Each basket "cell" is formed from sheet stainless steel. The walls of each cell are slotted to provided coolant flow to the contained fuel. The cells are positioned by nine circular spacers placed along the basket length. These spacers or "spacer disks," also center the basket in the cask cavity. The basket cells run the full length of the fuel. When the cask is horizontal, the weight of the fuel bundle is carried by the spacer disks. The cells are not principal load carrying members; they function as guides for ease in fuel loading.

As shown on 159C5238 sheets 6, 10 and 11 the BWR fuel basket is fitted with gamma shielding at its head-end. These assemblies consist of stainless steel-clad depleted uranium pieces and their supporting structures. The assemblies are permanently fixed to the basket. This shielding attenuates the cobalt-60 radiation from the BWR fuel bundle upper tie plates.

4.1.2 Cask Support and Tiedown

The cask support and tiedown arrangement is shown on 159C5238 sheet 5. It consists of a front saddle and a rear cradle/pedestal assembly.

4.1.2.1 Front Saddle

As illustrated (159C5238, Section B-Sheet 2, Sheet 5) the front saddle is a steel structure which is welded to the skid framing. When the cask is horizontal its upper pair of structural rings straddle the front support. There is approximately 45 degrees of circumferential contact between the cask body and the front support saddle.

This structure provides full axial restraint and partial vertical and lateral restraint of the cask. Axial restraint comes from the contact between the saddle and one of the two structural rings (depending on direction of movement). Vertical restraint is achieved by pinning the cask to the structure as illustrated on 159C5238 Sheet 5, Detail B. Lateral restraint comes from the cask-to-support contact and the tiedown pinning.

During cask rotation to the horizontal position, contact is made between the uppermost structural ring on the cask and the front edge of the saddle. The force generated by this contact draws the cask one-inch forward, out of the rear cradle. This capability of forward cooling system diesels are incorporated into the framing. Deck plate is provided for all accessible areas. The cooling system and cask support members are attached directly to the frame. The skid is 37½ feet long, 8 feet wide and is of high strength steel construction.

Both ends of the skid are designed to accept a hydraulic "gooseneck." For transporting the package by truck, wheeled assemblies can be attached to both ends of the skid. The "goosenecks" can be used to lift the unit to a minimum road clearance of 12 inches for highway transportation.

During rail shipment, the skid sits directly on the bed of a slightly modified standard 90 ton 4-axle flat car. The skid is restrained by a tiedown system designed to comply with the load requirements of rule 88A.1.d of the Association of American Railroads Field Manual.

4.1.5 Enclosure

Exclusion from the cask and cooling system is provided by an aluminum frame and expanded metal cage as shown on 159C5238 Sheet 2. This enclosure is in three sections; two sections are over the cask and the third covers the cooling system. The two cask enclosures move along rails and telescope over the third one, which is semi-permanent, to facilitate cask removal. The enclosure ends are also semi-permanently attached to the skid. When the movable sections are retracted, the rails form a sill which protects the bottom air ducting and provides a work platform along the cask. when the sections are in place over the cask, a locking device lifts them off of their tracks and secures their movement. This device is padlocked during transit.

The cooling equipment end wall has a lockable access door for inspection. In addition, there is one small removable panel on each side of the equipment enclosure which permits access to each of the engine/blower instrument consoles. The equipment enclosure and the end walls may be removed by unbolting.

All three enclosure sections have solid roofs for sun shading. The enclosure ends are also solid. This entire enclosure makes the nearest accessible external shipping package surface approximately four feet from the cask centerline.

V. STRUCTURAL INTEGRITY ANALYSIS

5.1 INTRODUCTION

The structural integrity analysis of the IF-300 cask design is described in this section. This includes the 7-cell PWR and 18-cell BWR fuel baskets licensed prior to 1991. The structural integrity analysis of the 17-cell channelled BWR fuel basket licensed in 1991 is described in Volume 3, Appendix A.

All design loads specified in 10 CFR Part 71 "Packaging of Radioactive Material for Transport", as amended, plus all loads imposed by the designer of the cask are accounted for in this analysis. The best available material properties and conservative assumptions have been used in the analysis so that the element being examined can not in actuality exhibit higher stresses than those of the analysis. The acceptance criteria used for the cask require factors of safety in excess of 1.0 when subjected to each design load. The safety factor is defined as:

$$SF = \frac{\text{Allowable (load, stress, displacement, etc.)}}{\text{Calculated (load, stress, displacement, etc.)}}$$

5.2 DESIGN LOADS

The following paragraphs describe the conditions to which the cask, fuel bundles, and fuel baskets have been analyzed for the highest G-loading the cask sustains in a 30-foot drop for a given orientation.

5.2.1 Regarded as a simple beam supported at its end along any major axis, the cask shall be capable of withstanding a static load, normal to and uniformly distributed along its length, equal to five times its fully loaded weight without generating stresses in any material in excess of its yield strength.

5.2.2 The cask inner cavity shall suffer no loss of contents if subjected to an external pressure of 25 psig.

5.2.3 The internal pressure of the cask in normal operation shall be less than 200 psig.

5.2.4 The maximum internal pressure shall be 400 psig. The rupture disk device shall be designed to burst within the range of 350-400 psig at 443°F.

5.2.5 The cask shall withstand a free drop through a distance of 30 feet onto a flat, essentially unyielding horizontal surface, striking the surface in a position for which the maximum damage is expected.

5.2.6 The cask shall withstand a free drop through a distance of 40 inches striking, in a position for which maximum damage is expected, the top end of a vertical cylindrical mild steel bar mounted on an essentially unyielding, horizontal surface. The bar shall be 6 inches in diameter, with the top horizontal and its edge rounded to a radius of not more than one-quarter (1/4) inch, and of such length as to cause maximum damage to the package, but not less than 8 inches in length. The long axis of the bar shall be perpendicular to the unyielding horizontal surface.

5.2.7 Cask lifting devices which are structural parts of the package shall support three times the weight of the loaded cask without exceeding the yield stress of any material.

5.2.8 Lifting devices which are part of the cask lid shall support three times the weight of the lid without exceeding the yield stress of any material.

5.2.9 The tiedown devices for attachment of the cask to the equipment skid shall be capable of withstanding a static force, applied at the center of gravity of the cask, having a vertical component of two times the weight of the package and contents, a horizontal component along the direction in which the vehicle travels of ten times the weight of the package contents, and a horizontal component in the transverse direction of five times the weight of the package and contents.

5.2.10 The cask body shall withstand the thermal stress conditions arising from: 1) normal cooling; 2) loss-of-mechanical cooling: 3) partial loss-of-shielding water; 4) 30-minute fire; and 5) post-fire equilibrium.

5.3 MATERIALS

Table V-1 presents the materials used in the cask, the 7-cell PWR and 18-cell BWR fuel assembly support baskets licensed prior to 1991, and miscellaneous attachments. Volume 3, Appendix A presents the materials used in the 17-cell channelled BWR fuel assembly support basket licensed in 1991.

5.3.1 Uranium Shielding Specification

The depleted uranium metal shielding material is in the form of annular castings, shrink-fitted together to form a continuous shield for the length of the cask. All casting, handling, testing and preparation for shipment are performed in accordance with General Electric Company approved specifications.

The cast material has a maximum U-235 content of 0.22%. The U-235 content of UF_6 tail material is nominally 0.20% with a ± 0.02% variation. Isotopic analysis has been performed on each casting to assure compliance with the aforementioned limit. Certified copies of the various analyses were originally provided to General Electric Company and are currently retained by VECTRA.

Table V-1

MATERIALS

Item	Materials
External water jacket	ASTM A240 Type 304
Inner shell	ASTM A296-65 CG-8M (317SST modified) or AISI 200 Type 216SST rolled plate
Shielding (casting)	Uranium, depleted metal
Outer body shell	ASTM A296-65 CG-8M (317SST modified)
Structural rings	AISI 200 Type 216
Valve box sides (castings)	ASTM A351-CF8 (304SST)
Valve box cover	AISI 200 Type 216
Bottom head outside shell	ASTM A240 Type 304
Bottom head inside shell	ASTM A240 Type 304
Top head outside shell	ASTM A240 Type 304
Top head inside shell	ASTM A240 Type 304
Top head flange (forging)	ASTM A182 304SST
Cask body flange (forging)	ASTM A182 304SST
Top head fins	ASTM A240 Type 304
Bottom head fins	ASTM A240 Type 304
Valve box fins	AISI 200 Type 216
Studs and nuts	17-4 PH H-1075/H-1025
Fuel element basket axial supports	AISI 200 Type 216
Fuel element basket channels	ASTM A240 Type 304
Basket support rings	AISI 200 Type 216
Support saddle	ASTM A516 Gr 70
Pivot cradle	ASTM A516 Gr 70
Cradle pedestals	ASTM A516 Gr 70
Block pin	AISI 4340 heat treated
Pivot cradle counter weight	Lead
Lifting trunnion blocks	AISI 4340, 304N or nitronic 40 stainless steel forgings.
Cooling ducts	6061/3003 aluminum
Enclosure	6061/6063/3003 aluminum
Skid	Tri-Ten steel

The shield material after completion of fabrication has the following nominal physical properties:

a.	Minimum Yield (0.2% offset):	35,000 psi
b.	Ultimate Tensile Strength:	60,000 psi
c.	Elongation:	6%
d.	Hardness:	Rockwell B-65
e.	Average Density:	18.82 $\pm$ 0.12 gm/cc

Samples from each heat are prepared and tested to demonstrate that the fabricated material meets the above listed physical properties. A density measurement is performed on each casting. Certified copies of each report were provided to General Electric and currently retained by VECTRA.

The porosity and soundness of all uranium castings are completely checked by coblt-60 gamma scanning. Strips of material having established thickness and density are placed at intervals on the casting surfaces to serve as reference points for checking the accuracy and sensitivity of the scanning equipment. Scanning follows procedures approved by General Electric Company.

Unacceptable porosity is defined as any area of the casting having deviation (increase) of gamma reading equivalent to a 5% decrease in the shielding thickness.

5.3.1.1 Uranium Properties

Uranium properties used in the calculations are contained in Reference 1, page 124, Figure 7.44.*

5.3.2 Fuel Basket Poison Material Specification

Criticality control in the 7-cell PWR and 18-cell BWR fuel baskets licensed prior to 1991 is provided by 0.5-inch diameter, boron carbide-filled, stainless steel tubes on 1.5-inch

* References are listed at the back of this chapter.

centers between adjacent fuel assemblies. These rods provide poison over the length of the basket and are fixed between the basket spacer disks.

Each poison tube is filled with chemically pure natural boron carbide consisting of 19.6 atomic percent B-10 and 80.4 atomic percent B-11. The minimum packed density of B_4C is 1.75 grams per cubic centimeter. The tubes may be either mechanically compacted or filled with prepressed B_4C pellets. A void space is provided in each tube to contain the small amount of helium produced in the boron-capture process.

The boron carbide columns extend well beyond the fuel active zone to compensate for any fractional settling which may occur with time. Each tube is loaded, backfilled with helium, seal welded and checked with a mass spectrometer.

The poison tube vendor is required to qualify fabrication and testing techniques prior to fabrication. Each batch of material is certified with copies of documentation retained by General Electric Company.

5.3.3 CG-8M (317 Modified) Stainless Steel

As indicated in Table V-1, the cask inner shell may be CG-8M but the outer shell must be a CG-8M centrifugal casting. The "317 modified" designation is placed on the material by the supplier (Sandusky Machine and Foundry) to indicate that a ferrite control process has been used to elevate the strength over those values tabulated in ASTM A296-65.

Properties have been derived from actual elevated temperature tests on CG-8M material as well as supplier data, as listed in Table V-2.

5.6.4 Fuel Basket Poison Rods - 30-Foot Drop

As described in Section IV, the poison rods for the 7-cell PWR and 18-cell BWR fuel baskets licensed prior to 1991 are 1/2-inch diameter by 0.020-inch wall 304 stainless steel tubes filled with boron carbide. These tubes are retained by cage plates attached to the spacer disks between adjacent elements. They run parallel to the axis of the fuel basket. (The borated stainless steel poison plates used in the 17-cell channelled BWR fuel basket licensed in 1991 are described in Volume 3, Appendix A.)

5.6.4.1 Side Drop

For the purpose of analysis, a single rod was examined since the failure of one rod will not lead to the failure of any other. The first 3/4 inch on each end of the rod is the supported length. Each rod is unsupported along its length. The maximum unsupported span is less than 20 inches. These tunes can be assumed to act as a beam built in at both ends. The span length (L) is assumed to be 20 inches for conservatism (see Figure V-31).

The maximum side drop deceleration is 122,3G's. The tube and born carbide act as a distributed loaf having a linear weight (W) of 1.92×10^{-2} pounds per inch. The maximum bending moment is given by:

$$M = \frac{WL^2}{12}$$

$$M = \frac{1.95 \times 10^{-2}}{12} (20)^2 = 0.65 \text{ inch-lb @ 1 G}$$

For 1/2 inch diameter tubing:

$$C = \frac{0.5}{2} = 0.25 \text{ inch}$$

Moment of inertia is given by the following:

$$I = \frac{\pi (r_o^4 - r_i^4)}{4}$$

$$I = \frac{\pi}{4}(0.25^4 - 0.23^4) = 8.78 \times 10^{-4} \text{ inch}^4$$

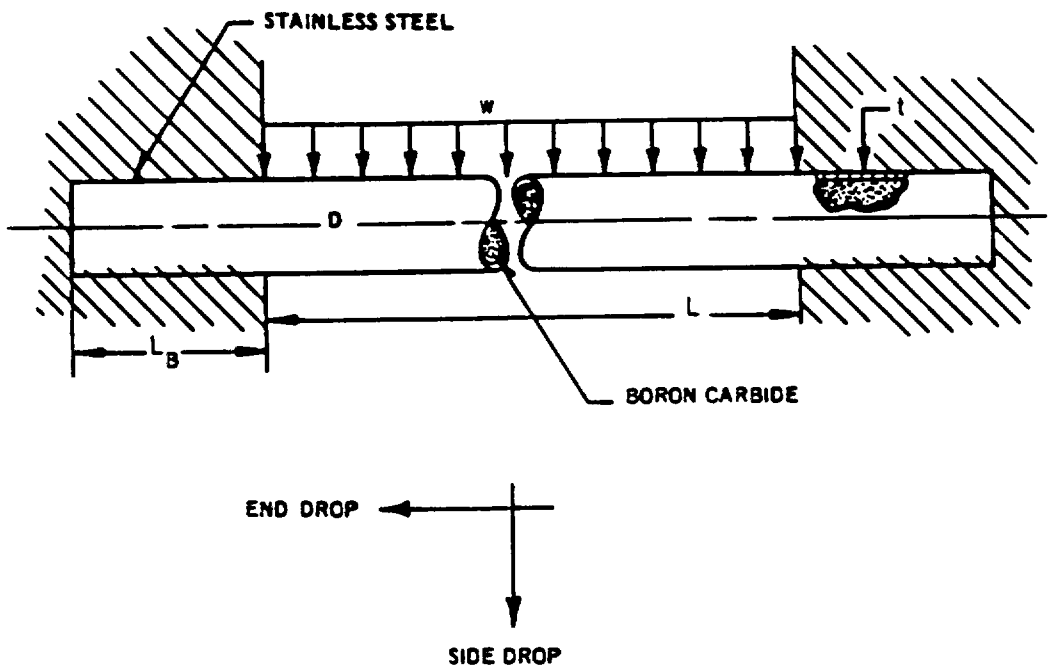

FIGURE V-31. POISON TUBE

The specification includes material, fabrication and equipment specifications. This document, along with the fabrication drawings forms a package which assures that the cask as fabricated and put into service has the highest level of quality assurance. In addition, those items which can be classified as containment are fabricated following a QA plan based on Appendix B of 10 CFR 50.

To provide further quality control, General Electric has an inspector in the fabricator's shop. General Electric also supplements the resident inspector with periodic audits conducted by personnel from the San Jose, California and/or Morris, Illinois facilities.

Prior to acceptance from the fabricator, the IF 300 cask is given a thermal demonstration test at design basis heat load conditions. These tests are used to verify the normal operating conditions described in Section VI. The thermal demonstration test sequence is described in Section VI. Each cask is functionally tested prior to acceptance by General Electric.

5.11.2.2 Maintenance and Inspection Procedures

The key to maintaining the initial quality and reliability of a system such as the IF 300 is periodic inspection and maintenance of functional areas. In the IF 300 system, these functional areas include:

a. Corrugated barrel and valving
b. Cavity closure flange and valving
c. Cask tiedown structure

The inspection and maintenance plan is discussed in Section X.

5.11.2.3 In-use Procedures

Section X discusses the activities which are performed prior to cask usage. A detailed operating manual exists and is periodically updated with emphasis on proper operation of the items listed in Section 5.11.2.2

For VECTRA-owned casks, VECTRA provides training to all individuals involved in the IF-300 handling operation. VECTRA conducts periodic audits on the condition of the equipment, to assure proper handling. Administrative controls such as sign-off sheets and similar documentation are used to confirm equipment functioning.

Specifically, these procedures require that prior to release to a carrier, the cask and components: (1) are operating properly; (2) temperatures and pressures are within limits for normal operation, and; (3) radiation and contamination levels are within limits. Further, where transshipments are performed, at the time of release to a railroad, the procedures will require that: (1) the mechanical cooling system, if desired, is operating properly, and (2) no package damage has been sustained since leaving the reactor site. These two criteria will assure that there has been no inadvertent damage during the over-the-road portion of the transport. It should be noted that no single failure of a component, including failure due to vandalism, will result in any release of radioactive material to the environment, and that the cooling system is not required for the safe use of the cask.

5.11.3 One-Foot Drop

The regulations include a one-foot drop as part of the conditions of normal transport. Since the IF-300 cask is transported solely in a horizontal orientation, only an analysis of the package (cask and skid) dropped one foot in horizontal orientation is necessary to satisfy the intent of the regulations.

VI. THERMAL ANALYSIS

6.1 INTRODUCTION

This section describes thermal analyses of the IF-300 shipping cask with the 7-cell PWR and 18-cell BWR baskets licensed prior to 1991. The characteristics of the types of fuels licensed prior to 1991 which may be transported in the cask are described in Section 3.0.

The analyses described in this section assume that the IF-300 cask is used in the "dry" shipping mode with a design basis heat load of 40,000 Btu/hr. The per bundle maximum decay heat rates are as follows:

- PWR fuel - 5725 Btu/hr
- BWR fuel - 2225 Btu/hr

Thermal analyses were originally done in 1973 for wet shipments of high heat load Group I BWR and PWR fuels. Additional analyses were completed for the Group I fuels in 1974 for low heat load dry shipments and in early 1980 for high pressure fuel pins. The newer Group II BWR and PWR fuels were analyzed in late 1980 with their results being similar to those of the Group I fuels. Fuel designs licensed since 1991 are presented in Volume 3, Appendices A and B.

Table VI-1 tabulates the characteristecs used in the thermal analyses of the Group I 7 x 7 BWR and 15 x 15 PWR fuels (14 x 14 PWR is bounded by the 15 x 15 fuel) and the Group II 8 x 8 BWR and 17 x 17 PWR fuels (16 x 16 PWR is bounded by the 17 x 17 fuel) licensed prior to 1991.

This section contains five "Design Basis Heat Load Conditions"; normal cooling, loss-of-mechanical cooling (LOMC), 50% shielding water loss, 30 minute fire, and post-fire equilibrium (PFE).

The mechanical cooling system is not required by the NRC. This system has been partially or completely removed from all four IF-300 casks. The thermal results are shown in this section. The LOMC results replace all normal cooling results.

Volume 3, Appendix A, Section A-3.0 has four design basis heat load conditions; normal cooling (NOC), 30 minute fire, 3-hour post fire, and post-fire equilbrium (PF3). NOC is natural convection in 130° ambient air, which is eqivalent to the LOMC in this section.

6.2 PROCEDURES AND CALCULATIONS

6.2.1 Introduction

The thermal analyses described in this section have been, with minor exceptions, calculated by computer. These calculations are based on parameters specified in Table VI-1. This section of the report describes the methodology incorporated in the various computer codes, discusses the bases for the procedures used, and details the calculations performed.

Table VI-1

CHARACTERISTICS OF CASK AND DESIGN BASIS FUELS

CASK

Type	BWR	PWR
Cavity Length, in.	180.25	169.50
Cavity Diameter, in.	37.5	37.5
Inner Shell Thickness, in.	0.5	0.5
Shielding Thickness, in.	4.0	4.0
Outer Shell Thickness, in.	1.5	1.5
Cask Linear Surface Area, ft^2/ft	39.2	32.2
Cask Length (Excluding Fins), in.	192.3	182.1
Shielding Water, lb	4,540	4,540
Cavity Relief Pressure, psig at 443°F	350 - 400	350 - 400
Neutron Shielding Relief Pressure, psig	200	200
Maximum Heat Load, Btu/hr - Air Filled	40,000	40,000

FUELS

Type	BWR	PWR
Number of Fuel-Bearing Rods/Bundle - Group I	49 (7 x 7)	208 (15 x 15)
Number of Fuel-Bearing Rods/Bundle - Group II	62 (8 x 8)	264 (17 x 17)
Exposure, GWd/MTU (average)	35.0	35.0*
Operating Power, kW/kgU	30.0	40.0
Assembly Decay Heat Rate, (max) BTU/hr, Air Filled	2,225	5,725
Assemblies per Cask Load	18	7
Uranium, kgU/Bundle	198	465

* See Volume 3, Appendix B for high burnup PWR fuel.

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6.4 FUEL CLADDING TEMPERATURES

This section presents analyses for fuels licensed prior to 1991. See Volume 3, Appendices A and B for fuels licensed since 1991.

6.4.1 Wooton-Epstein Correlation

Since there has been no work done in testing a configuration similar to the IF-300 cask, the work of Wooton and Epstein (Ref. 6.23) has been selected as the best available basis for predicting fuel rod temperatures in a "dry" If-300 cask. As a benchmark, this method was applied to the experimental work of Watson (Ref. 6.10) and succeeded in predicting hottest rod cladding temperature within 1% of the measured value.

The analytical approach of Wooton and Epstein is based primarily on radiation, with a convective term added to reflect heat removal from the fuel bundle exterior. The radiation term is derived by assuming that the bundle consists of a series of concentric surfaces where the area ratio between any two such surfaces is approximately one. This ratio makes the radiative exchange factor (F) between any two surfaces only emissivity dependent.

This correlation relates the cask cavity wall temperature to the hottest rod cladding temperature for any given heat load and fuel bundle configuration.

$$Q = \sigma C_1 F_1 A_1 (T_E^4 - T_C^4) + C_2 A_1 (T_E - T_C)^{4/3} \qquad (6.12)$$

where:

$$C_1 = \frac{4N}{(N+1)^2} (N \text{ odd})$$

or:

$$C_1 = \frac{4}{N+2} \quad (N \text{ even})$$

and:

N = number of rod rows in bundle

F_1 = exchange factor

$$= \left(\frac{1}{\varepsilon E} + \frac{1}{\varepsilon C} - 1\right)^{-1}$$

A_1 = bundle surface area

= 4HL

H = height of one side of bundle (ft)

L = length of bundle (ft)

T_E = hottest rod cladding temperature (°R)

T_C = cavity wall temperature (°R)

C_2 = convection constant

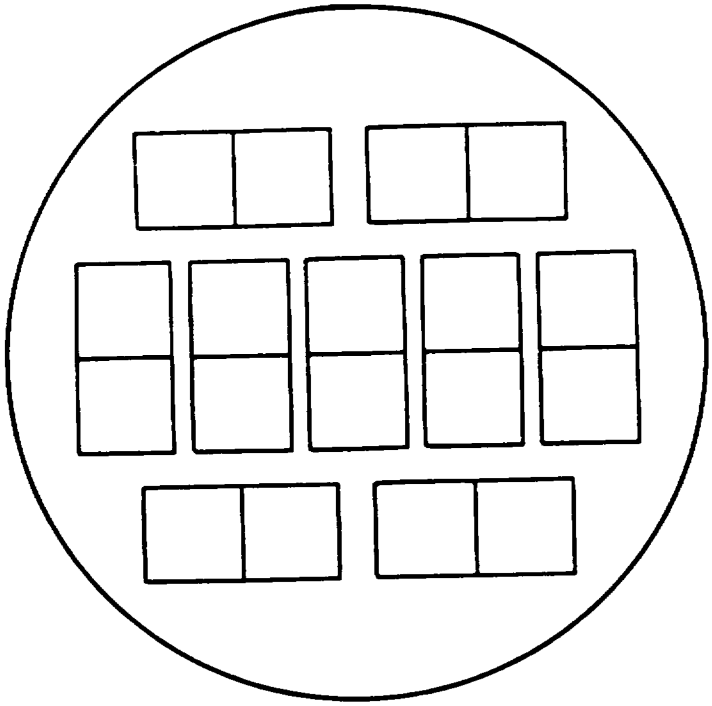
= 0.118 (air)

Q = decay heat rate (Btu/hr)

ε = material emissivity

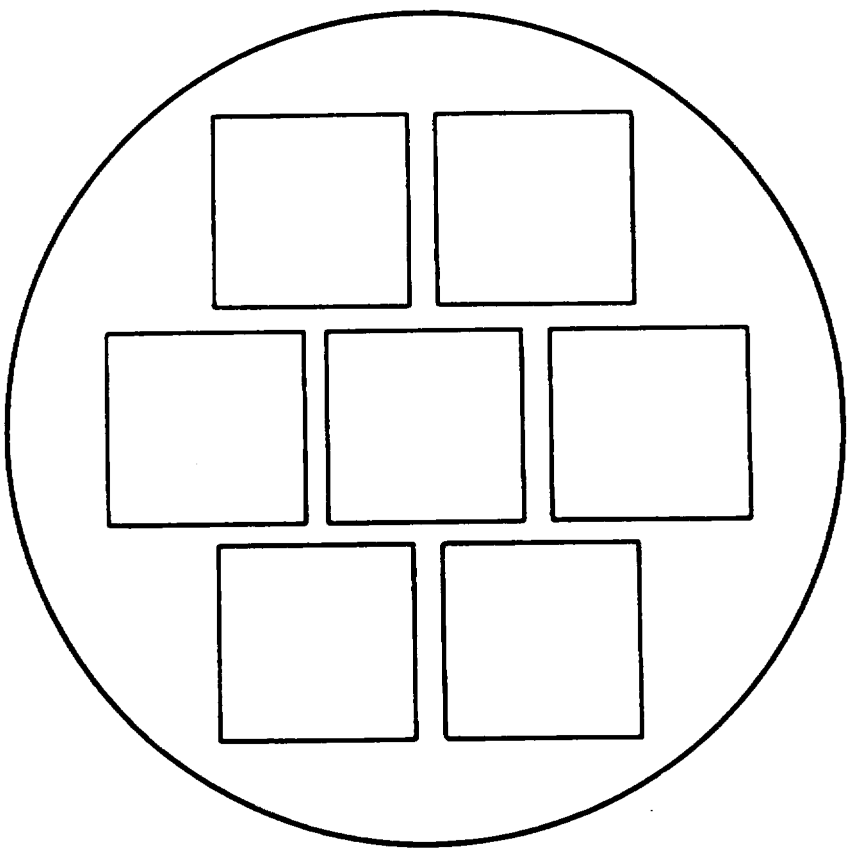
δ = Stefan - Boltzmann Constant

It is assumed that the above equation can be applied to both BWR and PWR fuel rod arrays within the IF-300 cask by modeling each with a square, NxN, uniformly spaced array having approximately the same number of fuel rods, the same heat load, and the same perimeter. The BWR & PWR fuel bundle arrays are shown in Figure VI-6.



18-CELL BWR CONFIGURATION*

(a)



7-CELL PWR CONFIGURATION*

(b)

Figure VI-6. Fuel Basket Configurations

* See Volume 3, Appendix A for 17-Cell Channelled BWR Fuel Basket

There are several features of this analytical method which make it conservative:

a. The Wootou-Epstein correlation is inherently conservative in that it is based on the assumption of concentric surfaces. In a fuel rod bundle radiation occurs through the spaces between rods thus reducing the effective number of rows. The actual test results conducted at BMI demonstrated that the correlation consistently overpredicted the hottest rod cladding temperature.

b. The cavity wall temperatures used in the analysis are maximums. Use of average cavity wall temperatures would result in lower fuel rod temperature predictions.

6.4.2 Analysis

6.4.2.1 Example Problem - PFE

As previously noted, both BWR and PWR configuration are analyzed using a direct application of the W-E equation. Table VI-18 shows the input parameters for PFE conditions for BWR 8x8 fuel and PWR 17x17 fuel.

Table VI-18

WOOTEN-EPSTEIN CORRELATION INPUT PARAMETERS - PFE

Parameter	BWR-8x8	PWR-17x17
C_1	0.111	0.085
N	34	45
H, ft	2.50	2.53
Q, Btu/hr	40,000	40,000
L, ft	12	12
C_2	0.118	0.118
T_c, °F	377	377
F_1	.539	.539
ε	0.7	0.7

VII. CRITICALITY ANALYSIS

7.1 INTRODUCTION

The IF-300 shipping cask has been designed to transport irradiated reactor fuel bundles from both pressurized water reactors (PWR) and boiling water reactors (BWR). The IF-300 cask utilizes interchangeable inserts or baskets in the cask cavity for fuel bundle support. There are three types of fuel baskets for 7 PWR, 18 BWR, and 17 BWR channelled fuel assemblies. The purpose of this chapter is to identify, describe, discuss and analyze the principle criticality engineering-physics design of the packaging, components and systems important to safety and necessary to comply with the performance requirements of 10 CFR Part 71 for the 7-cell PWR and 18-cell BWR baskets licensed prior to 1991. The 17-cell BWR channelled fuel basket design is addressed in Volume 3, Appendix A.

7.2 DISCUSSION AND RESULTS

Criticality control for the PWR and BWR fuel licensed prior to 1991 in the IF-300 cask is achieved through the use of boron carbide (B_4C) filled stainless steel tubes permanently affixed to the fuel baskets as opposed to borated stainless steel poison plates used in the 17-cell BWR channelled fuel basket. The IF-300 cask is shown in quarter symmetry in Figures VII-1 and VII-2, showing the PWR and BWR geometries licensed prior to 1991 and B_4C tube locations. These absorber rods are manufactured by the General Electric Company following the same standards, where applicable, used for BWR control blade absorber tubes. Quality control checks include B_4C density determinations, helium leak checking and material certifications on both tubing and end plugs.

The criticality analysis calculations were performed with the MERIT computer program, a Monte Carlo program which solves the neutron transport equation as an eigenvalue or a fixed source problem and includes the effects of neutron shielding. This program is especially written for the analysis of fuel lattices in thermal nuclear reactors. MERIT has the capability to perform calculations in up to three dimensions and with neutron energies between 0 and 10 MeV. MERIT uses cross sections processed from the ENDF/B-IV library tapes. The qualifications of MERIT is addressed in Section 7.5.

The IF-300 cask was shown to be critically safe for the transport of both PWR and BWR fuels supplied to domestically designed reactors. Both abnormal and accident conditions were considered. Detailed results of the analysis are contained in Section 7.4 and fuel descriptions are contained in Section 7.2. In summary, the maximum cask k-effective values for PWR and BWR fuels are in Table VII-1.

These values calculated for one sigma include MERIT calculation uncertainty and bias.

Table VII-1
K-EFFECTIVE VALUES

	PWR	BWR
k_{eff} (4.0% enrichment)	0.955 ± 0.004	0.880 ± 0.005

7.3 CASK FUEL LOADING

Prior to 1991, the IF-300 cask was designed to carry either 18 BWR fuel bundles or 7 PWR fuel bundles. As described in Volume 3, Appendix A, it was licensed to also carry 17 BWR channelled fuel bundles in 1991. This section addresses the former types of fuel.

BWR fuel bundles are primarily manufactured by General Electric and are square arrays of Zircaloy tubes containing the UO_2 fuel pellets. These arrays are either the earlier Group I 7 x 7 design or the current Group II 8 x 8 design which are held in position by a tie plate at the upper end and a nozzle at the lower end. Longitudinal spacing and support is provided by a series of spacer assemblies along the length of the fuel rods. The design basis BWR bundle in this analysis has the dimensions shown in Table VII-2.

The BWR design does not contain control rods within the bundles, but some BWR bundle designs may contain non-fueled rods, (water-holes) in one or two rod locations as shown in Figure VII-1.

VIII. SHIELDING

8.1 FUEL BASES AND SOURCE TERMS

Section III describes the BWR and PWR design basis fuels and Section IV indicates the maximum number of each type which the IF-300 cask will hold as licensed prior to 1991. Volume 3, Appendices A and B describe BWR and PWR fuels licensed since 1991. Considering 18 BWR bundles or 7 PWR bundles, the latter represents the more severe shielding problem because of its higher specific operating power and higher exposure potential due to greater enrichment. For this reason, the IF-300 cask shielding analysis is based on consideration of 7 PWR design basis bundles. Volume 3, Appendix A describes the shielding analysis for the IF-300 cask with 17 channelled BWR fuel assemblies. Table VIII-1 gives the parameters of both reference fuel loadings for comparison. The source term has two components, gamma and fast neutron.

8.1.1 Gamma Radiation

The gamma source comes from the decay of radioisotopes produced in the fuel during reactor operation. The gamma source strength is a function of fuel operating specific power, irradiation time and cooling time. Table VIII-2 shows a seven group distribution for fuels licensed prior to 1991 (see Volume 3, Appendices A and B for fuels licensed since 1991). The seven group distribution is based on 875 operating days at a specific power of 40 kW/kgU, followed by 120 days of cooling. This forms the shielding computer solution input.

8.1.2 Fast Neutron Radiation

Recent work indicates that light water reactor fuel with a burnup of greater than 20,000 MWd/T will contain sufficient concentrations of transplutonium isotopes to make neutron shielding in a shipping cask a necessity.

The isotopes that form the primary neutron source in high exposure fuel are Curium 242 and Curium 244. In a U-235 fueled reactor, the formation of one atom of CM-242 requires four neutron capture events, while Cm-244 requires six neutron captures. Thus the concentration of these isotopes will depend, roughly on the fuel exposure to the fourth and to the sixth power until the concentrations approach their

TABLE VIII-1
IRRADIATED FUEL PARAMETERS

PWR Parameters:

Specific Power	=	40 kWth/kgU
U/Assembly	=	465 kg
Average Power/Assembly	=	18.28 MWth
Peaking Factor	=	1.2
Peak Power/Assembly	=	21.94 MWth
Power/Basket*	=	153.6 MWth
Vol of 7 bundles	=	1.178×10^6 cm^3

BWR Parameters:

Specific Power	=	30 kWth/kgU
kgU/Assembly	=	198 kgU
Average Power/Assembly	=	5.85 MWth
Peaking Factor	=	1.2
Peak Power/Assembly	=	7.02 MWth
Power/Basket*	=	126.36 MWth
Vol of 18 bundles	=	1.616×10^6 cm^3

*Denotes power of fuel while in reactor

TABLE VIII-2
ENERGY GROUPS

Group	Energy Range	Effective Energy	MEV/Fission
I	>2.6 MeV	2.8 MeV	- NEG -
II	2.2 - 2.6	2.38	1.54×10^{-5}
III	1.8 - 2.2	1.97	4.22×10^{-4}
IV	1.35 - 1.80	1.54	2.42×10^{-4}
V	0.9 - 1.35	1.30	1.08×10^{-4}
VI	0.4 - 0.9	0.80	4.16×10^{-2}
VII	0.1 - 0.4	0.40	6.02×10^{-4}

The seven-group distribution is taken from data published by K. Shure in WAPD-BT-24.

equilibrium values. Because of this, the neutron dose will be small with exposures less than 20,000 MWd/T. Figure VIII-1 show the curium production reaction chain.

The Cm-242 and Cm-244 produce neutrons by two types of mechanisms: spontaneous fission and (α, n) reactions with the oxygen in the fuel. ORNL-4357, "Curium Data Sheets," gives the $^{242}CmO_2$ neutron emission rates as 2.34 x 107 n/sec-gm from (α, n) and 2.02 x 10^7 from spontaneous fission. The (α, n) and spontaneous fission yields of $^{244}CmO_2$ are 5.05 x 10^5 and 5.05 x 10^7 respectively. The document also indicates that the (α, n) and spontaneous fission neutron spectra are quite similar to the energy spectrum of neutrons from thermal fission of U-235.

The concentration of Cm-242 and Cm-244 in spent BWR and PWR fuels of exposures up to ~44,000 MWd/T has been measured and reported in WCAP-6085, BNWL-45 and GEAP-5746. In addition, calculations of the curium concentrations have been made using effective cross sections based on the measured data. These calculations are reported in GEAP-5355, BNWL-1010, and by E.D. Arnold of Oak Ridge National Laboratory. As expected, the Cm-242 and Cm-244 concentrations depend on the spectrum and total fluence seen by the fuel. Thus, for a specified exposure, the magnitude of the curium concentrations for various fuel types will cover a range which is determined by the enrichments and spectra considered. These various measurements and calculations have been combined to yield a band of probable values for neutron emission rate. Figure VIII-2 is a graph of neutron emission rate from Cm-242 and Cm-244 vs. fuel exposure. The upper limit of the band represents low enrichment fuels, the lower limit is for high enrichment fuels.

The design basis neutron source strength for the IF-300 cask is 3 x 10^9 neutrons per second for fuels licensed prior to 1991. Calculations show that the exposure which will yield this source is 35,000 MWd/T for a capacity loading of either BWR or PWR fuel. See Volume 3, Appendices A and B for fuels licensed since 1991.

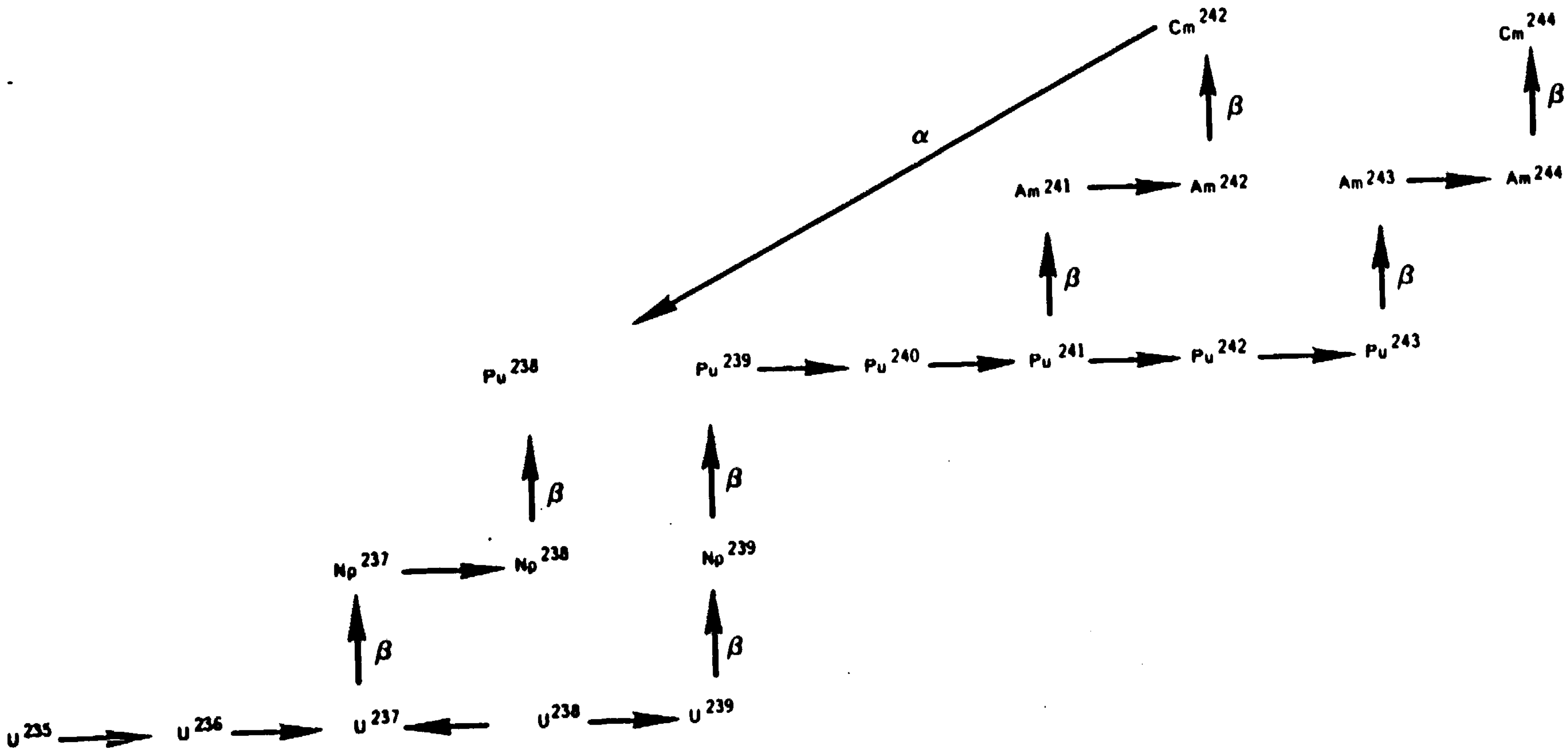


Figure VIII-1 Nuclear Reaction Sequence in UO_2 Fuel the cask axial centerline (six feet from the nearest accessible surface) is the side of the screened and locked enclosure. This enclosure extends to the edge of the eight-foot wide equipment skid. Since the cask centerline is on the skid centerline, a point six feet from the skid edge is also ten feet from the cask centerline (see Section IV - Equipment Description).

TABLE VIII-5

GAMMA AND NEUTRON SHIELDING RESULTS*

	R_{10} 10 ft from Cask Centerline	R_3 Accident 3 ft from Cask Surface	F_9 9 ft from Flange	T_9 9 ft from Top Head	B_9 9 ft from Bottom End
Gamma (mr/hr)	5.46	17.6	< 0.2	3.0	2.8
Neutron (mRem/hr)**	3.96	440.0	< 0.02	≤ 0.6	0.4
Total (mRem/hr)	9.42	457.6	< 0.22	≤ 3.6	3.2
Regulatory Limit (mRem/hr)***	10	1000	10	10	10

* Locations of R_{10}, R_3, F_9, T_9, and B_9 illustrated in Figure VIII-3 for fuels licensed prior to 1991. See Volume 3, Appendices A and B for fuels licensed since 1991.

** Includes fission in uranium shield.

*** 10CFR71 and 49CFR173.

8.2.4 Calculational Results:

49CFR173 prescribes the allowable dose rates as 10 mr/hr total radiation at a point 6 feet from the nearest accessible surface of the package equidistant from the ends, or 200 mr/hr at the cask surface, whichever is greater. The former pertains to the IF-300 cask. Furthermore, 10CFR71 specifies a limit of 1 R/hr three feet from the cask surface following the accident conditions. Table VIII-5 indicates that the cask shielding meets both normal and accident shielding requirements.

8.3 INTERNAL SHIELDING

Supplementary shielding has been added to the upper end of the BWR fuel basket. The computer analysis of these stainless steel-clad uranium metal components and their supporting structures is contained as an appendix to the structures analysis Section V of this SAR.

8.4 AIR-FILLED CAVITY SHIELDING

The IF-300 cask cavity may be air-filled rather than water-filled provided the heat load is less than 40,000 Btu/hr. This low decay heat rate can be produced by various combinations of fuel exposure and cooling time (i.e. high exposure - long cooled, low exposure - short cooled, etc.)

For dry shipments the reduced allowable heat load reduces the gamma and neutron source strengths. The expected dose rates under both accident and normal conditions are less than dose rates calculated for wet shipments.

8.5 DOSE-RATE ACCEPTANCE CRITERIA

10CFR, § 71.51(a)(2) limits the post-accident dose rate to 1,000 millirems per hour at 3 feet from the external surface of the package. The IF-300 cask contents must be so limited as to meet § 71.51(a)(2). This limitation is implemented by applying multipliers to the normal condition dose rate measurements which are taken prior to shipment. If the sum of the adjusted measurements exceeds an established value the shipment cannot be made.

8.5.1 The measurements, adjustments and limits are applied as follows:

$$(\text{Gamma D/R})(11.3) + (\text{Neutron D/R})(111.0) \leq 1{,}000 \text{ mr/hr}$$

IX. SAFETY COMPLIANCE

9.1 INTRODUCTION

This section is designed to recap and summate this report in light of the requirements of 10CFR71 - Subpart C, and 49CFR173.

9.2 10CFR71

9.2.1 General Standards for All Packaging

9.2.1.1 No Internal Reactions

The cask surfaces and the fuel baskets are stainless steel. This material does not react with steam or water either chemically or galvanically. The fuel is designed to be nonreactive in waterfilled systems. The uranium shield is totally clad in stainless steel. A copper diffusion barrier separates the stainless steel from the uranium to prevent the formation of an alloy under high temperature conditions. The entire shipping package is chemically and galvanically inert.

9.2.1.2 Positive Closure

The IF-300 cask head is held in place by 32 bolted studs. The mating flanges are designed to accept a Grayloc metallic gasket with a minimum design pressure of 600 psi. Shear steps are provided in the flange to prevent damage to the gasket under impact. Two tapered guide pins ensure proper head alignment during installation.

9.2.1.3 Lifting Devices

The analysis of Section V indicates that the lifting structures of both the cask and the lid are capable of supporting three times their respective weights without generating stresses in excess of their yield strengths (FS > 1.0).

The cask design is such that there are no possible lifting points other than those intended. In addition, the failure of any of the intended lifting structures will not result in a redistribution of shielding or a loss of cask integrity.

9.2.1.4 Tie Down Devices

Section V shows that both the front and rear cask supports are capable of sustaining the combined 10 g longitudinal, 5 g transverse and 2 g vertical forces without generating stresses in excess of their yield strengths (FS > 1.0).

The cask is designed to have only one tiedown method. The failure of either, or both, supports will not impair the ability of the package to meet other requirements. There will be no shielding redistribution or loss of cask integrity.

9.2.2 Structural Standards for Large Quantity Shipping

9.2.2.1 Load Resistance

With the package considered as a simple beam loaded with five times its own weight, the cask body outer shell safety factors in shear and bending are 20.4 and 8.6 respectively, based on allowable stresses.

9.2.2.2 External Pressure

When subjected to an external pressure of 25 psig, the package outer shell safety factors in elastic stability and axial failure exceed unity, based on allowable stresses.

9.2.3 Criticality Standards for Fissile Material Packages

This section addresses the 7-cell PWR and 18-cell BWR fuel baskets licensed prior to 1991. Volume 3, Appendices A and B address fuels and baskets licensed since 1991.

9.2.3.1 Maximum Credible Configuration

Fuel element spacing is provided by the stainless steel basket. The stress analysis of Section V shows that during accident conditions there is no redistribution of fuel. The normal transport arrangement is the maximum credible configuration.

All cask closure nuts will be safety wired prior to shipment. In addition, enclosure access doors and panels are locked during transit.

Under the normal shipping conditions, the nearest accessible surface temperature remains below the 180°F limit.

9.4 BASIC COMPONENTS (Safety Related) E

Certain components and structures of the IF-300 casks are safety related and as such are identified as Basic Components. Basic Components of the IF-300 are listed in Table IX-1 according to their nuclear functions which are a) containment of radioactive material within 10CFR71 limits, b) nuclear shielding, and c) criticality control. IF-300 Basic Components are designed, fabricated, assembled, tested, used and maintained under an NRC approved quality assurance program that satisfies the requirements in 10CFR71 Subpart E
H, "Quality Assurance".

Table IX-1
IF-300 BASIC COMPONENTS
(Safety Related)

I. CONTAINMENT

- Cavity End Plate
- Inner Shell
- Vent Pipe Assembly
- Locating Key
- Body Flange
- PWR Head Forging
- PWR Head Subassembly
- BWR Head Forging
- BWR Head Liner
- Trunnion Assembly
- Valve Boxes
- BWR Head End Plate
- BWR Head Liner Ring
- BWR Sleeve Nuts
- PWR Sleeve Nuts
- Studs
- Cavity Globe Valves
- Valve Pipe Cap or Plugs
- Valve Hardware
- Grayloc Seal Ring
- Fins
- Cavity Drain Line Assembly
- Rupture Disk Device

II. NUCLEAR SHIELDING

Uranium shield (cask barrel, closure head, bottom; basket shield), Neutron shield (corrugated barrel, valve boxes, expansion tank, piping, valves, blind flanges, liquid.)

III. CRITICALITY CONTROL

BWR Baskets
PWR Basket

c. The cask is slowly raised (while monitoring radiation levels) until the top of the cask reaches the level of the fuel pool curb.
d. Four cask closure head sleeve nuts are installed, hand tight.
e. The cask is removed from the pool (while again monitoring radiation levels), washed, and placed in the preparation area.
f. The yoke is removed and set aside.

10.1.1.9 Securing the Cask Closure Head

a. Parallelism of the head and cask flanges is tested and the head sleeve nuts are torqued to 370 ft-lbs minimum.
b. After metal-to-metal contact (.007 inch gap or less) is achieved between the head and cask flanges, the head sleeve nuts are lockwired for security.

10.1.1.10 Flushing of the Cask Inner Cavity

a. When desired, the cask inner cavity may be flushed with demineralized water until sample analysis conforms with pre-determined limits. This step is not mandatory.

10.1.1.11 Draining of the Cask Inner Cavity

N

a. A pressure regulated helium supply is connected to the cask cavity vent valve.
b. A drain hose is connected to the cask cavity fill/drain valve and directed into a radwaste drain or back into the pool.
c. After opening the cask cavity vent and fill/drain valves, helium is introduced through the vent valve at 15 psig.
d. When helium is observed to flow out of the cask cavity drain hose, the fill/drain valve is closed and the cask cavity pressurized to 15 psig.
e. The drain hose is removed.
f. The cask cavity vent valve is closed and the helium supply removed.

10.1.1.12 Assembly Verification Leakage Testing

a. Leakage testing of the cask closure seal, vent valve, fill/drain valve, and rupture disk device is performed with a thermal conductivity sensing instrument. This type of instrument is sensitive to any gas stream having a thermal conductivity different from the ambient air in which the instrument is being used.
b. The test instrument is set up and used according to written procedures and the manufacturer's instructions.
c. With the instrument calibrated to a sensitivity of at least 2×10^{-1} cm^3/sec (helium), the vent valve, fill/drain valve, and rupture disk device are checked for indications of leakage.

d. With the instrument calibrated to a sensitivity of at least 2 x 10^{-2} cm^3/sec (helium), the closure seal is checked for indications of leakage. (The sensitivity of this test is increased to account for the dilution which would occur between a potential point of closure seal leakage and the nearest point of measurement.)

e. If leakage is detected during either of the above checks, the offending components are repaired or replaced and then re-tested for leakage.

f. Valve must be checked to be open if pipe cap or plugs are used.

10.1.1.13 Preparing the cask for Transport of Irradiated Fuel

a. Steps 10.1.1.11a thru c are repeated. Nitrogen may be used to supply the third cask volume of inert gas.

b. The supply of helium (nitrogen) is discontinued when at least one additional cask volume has been supplied to the inner cavity. (One cask volume equals 83 cubic feet when shipping irradiated fuel.)

c. the excess helium (nitrogen) within the inner cavity is bled off thru the fill/drain valve until the cavity pressure has decayed to 0 psig. This completes the process of inerting the cask cavity.

d. The vent and fill/drain valve is closed and the connecting hoses and gages are removed.

e. The cask, skid, and rail car are decontaminated in accordance with regulatory requirements.

f. The cask is lifted with the yoke, positioned on the tilting cradle, and lowered to its horizontal position.

g. The yoke is removed.

h. The trunnions are removed and the cask tiedown pins installed.

i. The valve box covers are replaced.

j. The radiological survey of the cask and rail car is completed.

10.1.1.14 Preparing the Cask for Transport of Irradiated Hardware

a. A drain hose is connected to the cask cavity fill/drain valve and directed into a radwaste drain or back into the pool.

b. Steps 10.1.1.13c thru j are repeated.

10.1.1.15 Closing the Equipment Skid

a. The cask enclosures are closed, locked, and sealed.

10.1.2 Procedures for Unloading the Package

Operations at the unloading facility are largely the same as loading operations with the major exception being the increased radiological awarenèss required for receiving a loaded cask. Each unloading facility must provide fully trained personnel and detailed operating procedures to cover all activities.