NAC INTERNATIONAL

Atlanta Corporate Headquarters
655 Engineering Drive
Norcross, Georgia 30092
770-447-1144
Fax 770-447-1797
www.nacintl.com

June 29, 2000

U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attn: Document Control Desk

Subject: Submittal of UMS® Universal Transport System Safety Analysis Report, Revision UMST-00A - Docket 71-9270 (TAC No. L22452)

References:
1. Request for Additional Information (RAI-1) for the UMS® Universal Transport System Safety Analysis Report Application, U.S. Nuclear Regulatory Commission, August 30, 1999
2. Safety Analysis Report for the UMS® Universal Transport System, Docket 71-9270 (TAC No. L22452), submitted April 30, 1997, Supplemented: September 12 and December 24, 1997; April 23, 1998; and May 27, 1999

NAC International (NAC) herewith submits ten copies of the responses to the U.S. NRC Request for Additional Information (RAI-1) for NAC-UMS® Transport System Safety Analysis Report (SAR), Reference 1. This submittal includes the RAI comments and NAC's responses presented in the standard NAC RAI response format, followed by SAR Revision UMST-00A of the NAC-UMS® Transport SAR.

The revised pages have been prepared in accordance with the following conventions:

- Revision indicators (shading and revision bars) are used to highlight changes. Shading indicates a revision from SAR Revision 0; while a revision bar indicates a change in the SAR text flow from Revision 0 or a change from a previous revision other than Revision 0.
- The changed pages for this submittal are designated as Revision UMSS-00A to provide a unique identification of the pages and changes.
- The List of Effective Pages are all designated Revision UMSS-00A and no revision bars are used on those pages.
- The entire document is provided in two volumes for your review.

If you have any comments or questions, please contact me or Jim Ballowe at (770) 447-1144.

Sincerely,

for Thomas C. Thompson
Director, Licensing and Competitive Assessment
Engineering & Design Services

ED20000808

EA790-SAR-001

DOCKET No. 71-9270

UMS®

UNIVERSAL MPC SYSTEM®

SAFETY ANALYSIS REPORT

for the

UMS® Universal Transport Cask

MAY 2000 UMST-00A

VOLUME 2 OF 2

NAC INTERNATIONAL

List of Effective Pages

Master Table of Contents

Chapter 1

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

2.6-381	Revision UMST-00A
2.6-382	Revision UMST-00A
2.6-383	Revision UMST-00A
2.6-384	Revision UMST-00A
2.6-385	Revision UMST-00A
2.6-386	Revision UMST-00A
2.6-387	Revision UMST-00A
2.6-388	Revision UMST-00A
2.6-389	Revision UMST-00A
2.6-390	Revision UMST-00A
2.6-391	Revision UMST-00A
2.6-392	Revision UMST-00A
2.6-393	Revision UMST-00A
2.6-394	Revision UMST-00A
2.6-395	Revision UMST-00A
2.6-396	Revision UMST-00A
2.6-397	Revision UMST-00A
2.6-398	Revision UMST-00A
2.6-399	Revision UMST-00A
2.6-400	Revision UMST-00A
2.6-401	Revision UMST-00A
2.6-402	Revision UMST-00A
2.7-1	Revision UMST-00A
2.7-2	Revision 0
2.7-3	Revision 0
2.7-4	Revision UMST-00A
2.7-5	Revision 0
2.7-6	Revision 0
2.7-7	Revision UMST-00A
2.7-8	Revision UMST-00A
2.7-9	Revision UMST-00A
2.7-10	Revision UMST-00A
2.7-11	Revision 0
2.7-12	Revision 0
2.7-13	Revision 0
2.7-14	Revision UMST-00A
2.7-15	Revision 0
2.7-16	Revision 0
2.7-17	Revision 0
2.7-18	Revision UMST-00A
2.7-19	Revision 0
2.7-20	Revision 0
2.7-21	Revision UMST-00A
2.7-22	Revision 0
2.7-23	Revision 0
2.7-24	Revision 0
2.7-25	Revision UMST-00A
2.7-26	Revision 0
2.7-27	Revision 0
2.7-28	Revision UMST-00A
2.7-29	Revision UMST-00A
2.7-30	Revision UMST-00A
2.7-31	Revision UMST-00A
2.7-32	Revision UMST-00A
2.7-33	Revision UMST-00A
2.7-34	Revision UMST-00A
2.7-35	Revision UMST-00A
2.7-36	Revision UMST-00A
2.7-37	Revision UMST-00A
2.7-38	Revision UMST-00A
2.7-39	Revision UMST-00A
2.7-40	Revision UMST-00A
2.7-41	Revision UMST-00A
2.7-42	Revision UMST-00A
2.7-43	Revision UMST-00A
2.7-44	Revision UMST-00A
2.7-45	Revision UMST-00A
2.7-46	Revision UMST-00A

List of Effective Pages (Continued)

List of Effective Pages (Continued)

2.7-115..........Revision UMST-00A
2.7-116..........Revision UMST-00A
2.7-117..........Revision UMST-00A
2.7-118..........Revision UMST-00A
2.7-119..........Revision UMST-00A
2.7-120..........Revision UMST-00A
2.7-121..........Revision UMST-00A
2.7-122..........Revision UMST-00A
2.7-123..........Revision UMST-00A
2.7-124..........Revision UMST-00A
2.7-125..........Revision UMST-00A
2.7-126..........Revision UMST-00A
2.7-127..........Revision UMST-00A
2.7-128..........Revision UMST-00A
2.7-129..........Revision UMST-00A
2.7-130..........Revision UMST-00A
2.7-131..........Revision UMST-00A
2.7-132..........Revision UMST-00A
2.7-133..........Revision UMST-00A
2.7-134..........Revision UMST-00A
2.7-135..........Revision UMST-00A
2.7-136..........Revision UMST-00A
2.7-137..........Revision UMST-00A
2.7-138..........Revision UMST-00A
2.7-139..........Revision UMST-00A
2.7-140..........Revision UMST-00A
2.7-141..........Revision UMST-00A
2.7-142..........Revision UMST-00A
2.7-143..........Revision UMST-00A
2.7-144..........Revision UMST-00A
2.7-145..........Revision UMST-00A
2.7-146..........Revision UMST-00A
2.7-147..........Revision UMST-00A
2.7-148..........Revision UMST-00A
2.7-149..........Revision UMST-00A
2.7-150..........Revision UMST-00A
2.7-151..........Revision UMST-00A
2.7-152..........Revision UMST-00A
2.7-153..........Revision UMST-00A
2.7-154..........Revision UMST-00A
2.7-155..........Revision UMST-00A
2.7-156..........Revision UMST-00A
2.7-157..........Revision UMST-00A
2.7-158..........Revision UMST-00A
2.7-159..........Revision UMST-00A
2.7-160..........Revision UMST-00A
2.7-161..........Revision UMST-00A
2.7-162..........Revision UMST-00A
2.7-163..........Revision UMST-00A
2.7-164..........Revision UMST-00A
2.7-165..........Revision UMST-00A
2.7-166..........Revision UMST-00A
2.7-167..........Revision UMST-00A
2.7-168..........Revision UMST-00A
2.7-169..........Revision UMST-00A
2.7-170..........Revision UMST-00A
2.7-171..........Revision UMST-00A
2.7-172..........Revision UMST-00A
2.7-173..........Revision UMST-00A
2.7-174..........Revision UMST-00A
2.7-175..........Revision UMST-00A
2.7-176..........Revision UMST-00A
2.7-177..........Revision UMST-00A
2.7-178..........Revision UMST-00A
2.7-179..........Revision UMST-00A
2.7-180..........Revision UMST-00A
2.7-181..........Revision UMST-00A
2.7-182..........Revision UMST-00A

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

3.5-4 .... Revision 0
3.5-5 .... Revision 0
3.5-6 .... Revision UMST-99A
3.5-7 .... Revision UMST-00A
3.5-8 .... Revision UMST-00A
3.5-9 .... Revision 0
3.5-10 .... Revision 0
3.5-11 .... Revision 0
3.5-12 .... Revision 0
3.5-13 .... Revision 0
3.5-14 .... Revision 0
3.5-15 .... Revision 0
3.5-16 .... Revision 0
3.5-17 .... Revision 0
3.5-18 .... Revision 0
3.5-19 .... Revision 0
3.5-20 .... Revision 0
3.5-21 .... Revision 0
3.5-22 .... Revision 0
3.5-23 .... Revision 0
3.5-24 .... Revision 0
3.5-25 .... Revision 0
3.5-26 .... Revision 0
3.5-27 .... Revision 0
3.5-28 .... Revision 0
3.5-29 .... Revision UMST-00A
3.5-30 .... Revision UMST-00A
3.5-31 .... Revision UMST-99A
3.6-1 .... Revision UMST-99A
3.6-2 .... Revision UMST-99A
3.6-3 .... Revision UMST-99A
3.6-4 .... Revision UMST-00A
3.6-5 .... Revision UMST-00A
3.6-6 .... Revision UMST-00A
3.6-7 .... Revision UMST-99A
3.6-8 .... Revision UMST-99A
3.6-9 .... Revision UMST-99A
3.6-10 .... Revision UMST-99A
3.6-11 .... Revision UMST-99A
3.6-12 .... Revision UMST-99A
3.6-13 .... Revision UMST-99A
3.7-1 .... Revision UMST-00A
3.7-2 .... Revision UMST-00A
3.7-3 .... Revision UMST-00A

Chapter 4

4-i .... Revision UMST-00A
4-ii .... Revision UMST-00A
4-iii .... Revision UMST-00A
4-iv .... Revision UMST-00A
4-1 .... Revision UMST-00A
4-2 .... Revision UMST-00A
4.1-1 .... Revision UMST-00A
4.1-2 .... Revision UMST-00A
4.1-3 .... Revision UMST-00A
4.1-4 .... Revision UMST-00A
4.1-5 .... Revision UMST-00A
4.2-1 .... Revision UMST-00A
4.2-2 .... Revision UMST-00A
4.2-3 .... Revision 0
4.2-4 .... Revision UMST-00A
4.2-5 .... Revision UMST-00A
4.2-6 .... Revision UMST-00A
4.2-7 .... Revision UMST-00A
4.2-8 .... Revision UMST-00A
4.2-9 .... Revision UMST-00A
4.2-10 .... Revision UMST-00A

List of Effective Pages (Continued)

4.2-11	Revision UMST-00A
4.3-1	Revision UMST-00A
4.3-2	Revision UMST-00A
4.3-3	Revision UMST-00A
4.3-4	Revision UMST-00A
4.3-5	Revision UMST-00A
4.4-1	Revision 0
4.5-1	Revision UMST-00A
4.5.1-1	Revision UMST-99A
4.5.2-1	Revision UMST-99A
4.5.2-2	Revision UMST-99A
4.5.2-3	Revision UMST-99A
4.5.2-4	Revision UMST-99A
4.5.2-5	Revision UMST-99A
4.5.2-6	Revision UMST-99A
4.5.2-7	Revision UMST-99A
4.5.2-8	Revision UMST-99A
4.5.2-9	Revision UMST-99A
4.5.2-10	Revision UMST-99A
4.5.2-11	Revision UMST-99A
4.5.3-1	Revision UMST-00A
4.5.3-2	Revision UMST-00A
4.5.3-3	Revision UMST-00A
4.5.3-4	Revision UMST-00A
4.5.3-5	Revision UMST-00A
4.5.3-6	Revision UMST-00A
4.5.3-7	Revision UMST-00A
4.5.3-8	Revision UMST-00A
4.5.3-9	Revision UMST-00A
4.5.3-10	Revision UMST-00A
4.5.3-11	Revision UMST-00A
4.5.3-12	Revision UMST-00A
4.5.3-13	Revision UMST-00A
4.6-1	Revision UMST-00A

Chapter 5

5-i	Revision UMST-99A
5-ii	Revision UMST-00A
5-iii	Revision UMST-99A
5-iv	Revision UMST-00A
5-v	Revision UMST-00A
5-vi	Revision UMST-97A
5-vii	Revision UMST-00A
5-viii	Revision UMST-00A
5-ix	Revision UMST-00A
5-1	Revision UMST-97A
5-2	Revision UMST-99A
5.1-1	Revision 0
5.1-2	Revision 0
5.1-3	Revision UMST-97A
5.1-4	Revision UMST-97A
5.1-5	Revision UMST-00A
5.1-6	Revision UMST-00A
5.1-7	Revision UMST-00A
5.1-8	Revision UMST-00A
5.1-9	Revision UMST-00A
5.1-10	Revision UMST-00A
5.2-1	Revision UMST-97A
5.2-2	Revision UMST-97A
5.2-3	Revision UMST-97A
5.2-4	Revision 0
5.2-5	Revision 0
5.2-6	Revision UMST-97A
5.2-7	Revision 0
5.2-8	Revision UMST-97A
5.2-9	Revision UMST-97A
5.2-10	Revision 0
5.2-11	Revision 0

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

List of Effective Pages (Continued)

6.6.1-1	Revision UMST-00A
6.6.1-2	Revision UMST-00A
6.6.1-3	Revision UMST-00A
6.6.1-4	Revision UMST-00A
6.6.1-5	Revision UMST-00A
6.6.1-6	Revision UMST-00A
6.6.1-7	Revision UMST-00A
6.6.1-8	Revision UMST-00A
6.6.1-9	Revision UMST-00A
6.6.1-10	Revision UMST-00A
6.6.1-11	Revision UMST-00A
6.6.1-12	Revision UMST-00A
6.6.1-13	Revision UMST-00A
6.6.1-14	Revision UMST-00A
6.6.1-15	Revision UMST-00A
6.6.2-1	Revision UMST-00A
6.6.2-2	Revision UMST-99A
6.6.2-3	Revision UMST-00A
6.6.2-4	Revision UMST-00A
6.6.2-5	Revision UMST-00A
6.6.2-6	Revision UMST-00A
6.6.2-7	Revision UMST-00A
6.6.2-8	Revision UMST-00A
6.6.2-9	Revision UMST-00A
6.6.2-10	Revision UMST-00A
6.6.2-11	Revision UMST-00A
6.6.2-12	Revision UMST-00A
6.6.2-13	Revision UMST-00A
6.6.2-14	Revision UMST-00A
6.6.2-15	Revision UMST-00A
6.6.2-16	Revision UMST-00A
6.6.2-17	Revision UMST-00A
6.6.2-18	Revision UMST-00A
6.6.2-19	Revision UMST-00A
6.6.2-20	Revision UMST-00A
6.6.2-21	Revision UMST-00A
6.6.2-22	Revision UMST-00A
6.6.2-23	Revision UMST-00A
6.6.2-24	Revision UMST-00A
6.6.2-25	Revision UMST-00A
6.6.2-26	Revision UMST-00A
6.6.2-27	Revision UMST-00A
6.6.2-28	Revision UMST-00A
6.6.2-29	Revision UMST-00A
6.6.2-30	Revision UMST-00A
6.6.2-31	Revision UMST-00A
6.6.2-32	Revision UMST-00A
6.6.2-33	Revision UMST-00A
6.6.2-34	Revision UMST-00A
6.6.2-35	Revision UMST-00A
6.6.2-36	Revision UMST-00A
6.6.2-37	Revision UMST-00A
6.6.2-38	Revision UMST-00A
6.6.2-39	Revision UMST-00A
6.6.2-40	Revision UMST-00A
6.6.2-41	Revision UMST-00A
6.6.2-42	Revision UMST-00A
6.6.2-43	Revision UMST-00A
6.6.2-44	Revision UMST-00A
6.6.2-45	Revision UMST-00A
6.6.2-46	Revision UMST-00A
6.6.2-47	Revision UMST-00A
6.6.2-48	Revision UMST-00A
6.6.2-49	Revision UMST-00A
6.6.2-50	Revision UMST-00A
6.6.2-51	Revision UMST-00A
6.6.2-52	Revision UMST-00A
6.6.2-53	Revision UMST-00A

List of Effective Pages (Continued)

Page	Revision
6.6.2-54	Revision UMST-00A
6.6.2-55	Revision UMST-00A
6.6.2-56	Revision UMST-00A
6.6.2-57	Revision UMST-00A
6.6.2-58	Revision UMST-00A
6.6.2-59	Revision UMST-00A
6.6.2-60	Revision UMST-00A
6.6.2-61	Revision UMST-00A
6.6.2-62	Revision UMST-00A
6.6.2-63	Revision UMST-00A
6.6.2-64	Revision UMST-00A
6.6.2-65	Revision UMST-00A
6.6.2-66	Revision UMST-00A
6.6.2-67	Revision UMST-00A
6.6.2-68	Revision UMST-00A
6.6.2-69	Revision UMST-00A
6.6.2-70	Revision UMST-00A
6.6.2-71	Revision UMST-00A
6.6.2-72	Revision UMST-00A
6.6.2-73	Revision UMST-00A
6.6.2-74	Revision UMST-00A
6.6.2-75	Revision UMST-00A
6.6.2-76	Revision UMST-00A
6.6.2-77	Revision UMST-00A
6.6.2-78	Revision UMST-00A
6.6.2-79	Revision UMST-00A
6.6.2-80	Revision UMST-00A
6.6.2-81	Revision UMST-00A
6.6.2-82	Revision UMST-00A
6.6.2-83	Revision UMST-00A
6.6.2-84	Revision UMST-00A
6.6.2-85	Revision UMST-00A
6.6.2-86	Revision UMST-00A
6.6.2-87	Revision UMST-00A
6.6.2-88	Revision UMST-00A
6.6.2-89	Revision UMST-00A
6.6.2-90	Revision UMST-00A
6.6.2-91	Revision UMST-00A
6.6.2-92	Revision UMST-00A
6.6.2-93	Revision UMST-00A
6.6.2-94	Revision UMST-00A
6.6.3-1	Revision UMST-00A
6.6.3-2	Revision UMST-00A
6.6.3-3	Revision UMST-00A
6.6.3-4	Revision UMST-00A
6.6.3-5	Revision UMST-00A
6.6.3-6	Revision UMST-00A
6.6.3-7	Revision UMST-00A
6.6.3-8	Revision UMST-00A
6.6.3-9	Revision UMST-00A
6.6.3-10	Revision UMST-00A
6.6.3-11	Revision UMST-00A
6.6.3-12	Revision UMST-00A
6.6.3-13	Revision UMST-00A
6.6.3-14	Revision UMST-00A
6.6.3-15	Revision UMST-00A
6.6.3-16	Revision UMST-00A
6.6.3-17	Revision UMST-00A
6.6.3-18	Revision UMST-00A
6.6.3-19	Revision UMST-00A
6.6.3-20	Revision UMST-00A
6.6.3-21	Revision UMST-00A
6.6.3-22	Revision UMST-00A
6.6.3-23	Revision UMST-00A
6.6.3-24	Revision UMST-00A
6.6.3-25	Revision UMST-00A
6.6.3-26	Revision UMST-00A
6.6.3-27	Revision UMST-00A

List of Effective Pages (Continued)

List of Effective Pages (Continued)

MASTER TABLE OF CONTENTS

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

Master Table of Contents (Continued)

THIS PAGE INTENTIONALLY LEFT BLANK

Table of Contents

Table of Contents (Continued)

List of Figures

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Tables

List of Tables (Continued)

3.0 THERMAL EVALUATION

This chapter presents the thermal design and analyses of the Universal Transport Cask for the 10 CFR 71 normal conditions of transport and hypothetical accident conditions. The analyses include consideration of design basis PWR and BWR fuel. Results of the analyses demonstrate that with the design basis payloads, the Universal Transport Cask meets the thermal performance requirements of 10 CFR 71 [1] and IAEA Safety Series No. 6 [2].

3.1 Discussion

The Universal Transport Cask is designed to transport one of three classes of PWR fuel or one of two classes of BWR fuel. Only the bounding evaluation for the PWR and BWR classes of fuel is reported herein. The bounding case is represented by a configuration consisting of the shortest Transportable Storage Canister, shortest fuel tube, and shortest fuel assemblies with the lowest effective thermal conductivity. The fuel assemblies are confined within the fuel basket. The shortest fuel basket contains the fewest support disks and longest space in the bottom of the cask cavity. The result is greater concentration of heat and maximized thermal resistance for rejection of heat through the cavity top and bottom. The shorter fuel tube results in reduced axial conductance.

The design basis heat loads are 20 kW for up to 24 PWR assemblies and 16 kW for up to 56 BWR fuel assemblies. The individual PWR assembly decay heat is limited to 0.83 kW and the individual BWR assembly decay heat load is limited to 0.29 kW. As shown in Section 3.4.6, the thermal analysis considers a range of fuel assembly burnup and cool times for both fuel types to establish the allowable cladding temperatures. These limits are used to establish the allowable decay heat loads for fuel having cooling times of 5 years or more.

The thermal analyses presented in the following sections use helium as the cover gas in the cask cavity. The analyses are performed with both air and helium as the cover gas in the canister. The canister is backfilled with helium in the design basis; therefore, the results with air as the canister cover gas are conservatively presented for use in the structural evaluations presented in Section 2.0.

Heat transfer from the Universal Transport Cask to the environment is by passive means only. No forced cooling is necessary. Conduction and radiation are the means by which heat is transferred from the fuel assemblies to the fuel tubes and through the tubes to the support disks and heat transfer disks. Heat is transferred through the support disks and heat transfer disks by conduction and radiation. Radiation and conduction are the means by which heat is transferred from the support disks and heat transfer disks to the canister wall and then to the cask cavity inner wall. From the Universal Transport cask cavity inner shell surface, heat is conducted through the lead (gamma shield) and then through the cask outer shell.

The neutron shield region surrounding the outer shell along most of the cask's length conducts heat to the neutron shield shell, primarily through the Cu/SS fins located within the NS-4-FR radial neutron shield material. The stainless steel shell that encloses the radial neutron shield is exposed to the environmental ambient temperature. Heat is removed from the surface of the neutron shield shell by convection and radiation. Heat transfer through the cask lid at the top of the cask and through the bottom forging and enclosed neutron shield material at the bottom of the cask is by conduction. Because of the insulating characteristics of the impact limiters, essentially no heat is removed from the ends of the cask. The bounding thermal conditions for the analysis required by 10 CFR 71 and IAEA Safety Series No. 6 under normal conditions of transport are presented in Table 3.1-1.

During normal conditions of transport and hypothetical accident conditions, the cask must reject the fuel decay heat to the environment without exceeding the operational temperature ranges of the cask seals or other components important to safety. In addition, to maintain fuel rod integrity for normal conditions of transport the fuel must be maintained at a sufficiently low temperature in an inert atmosphere that thermally induced fuel rod cladding deterioration is precluded. To preclude fuel degradation, a maximum allowable cladding temperature of 716°F (380°C) is used for normal conditions of transport for 5-year cooled PWR and BWR fuel. Finally, the thermally induced stresses, in combination with pressure and mechanical load stresses, must be below allowable stress levels.

The temperatures for the various components of the fuel, canister, basket, and cask during normal conditions of transport and hypothetical accident condition fire are calculated by using finite element methods. For both normal conditions and the hypothetical accident conditions, the cask

loaded with PWR fuel and the cask loaded with BWR fuel are analyzed by using separate finite element models. For each fuel configuration, the thermal analyses of the cask for normal conditions of transport are performed by using three-dimensional finite element models of the loaded cask. The cask is transported in a horizontal orientation - Figures 3.1-1 and 3.1-2 show the gaps in the cask for PWR and BWR configurations, respectively. These models are described in Section 3.4.1.1 and 3.4.1.2. The thermal analyses of the cask for the hypothetical accident fire condition are performed by using two-dimensional models of the cask. These models are described in Section 3.5.1.1.

Results of thermal analysis of the package are presented in Sections 3.4.7 and 3.5.6. The results demonstrate that the maximum fuel rod cladding temperatures remain below 1,058°F for normal conditions of transport and hypothetical accident conditions. The thermally induced stresses, combined with pressure and mechanical load stresses, are within the allowable levels, as demonstrated in Chapter 2.0. Therefore, the cask design and operation are in conformance with temperature and thermal stress criteria.

The thermal results presented in this chapter, and other properties evaluated herein, are used in other analyses included in this Safety Analysis Report. The material properties and allowable stresses at the corresponding temperatures are used in the structural calculations presented in Chapter 2.0. The structural evaluation of components also incorporates stresses resulting from differential thermal expansion and temperature effects on the cask internal pressure as applicable.

Figure 3.1-1 Definition of Gap Between Basket, Canister, and Inner Shell for Horizontal Position of the Universal Transport Cask Containing PWR Fuel

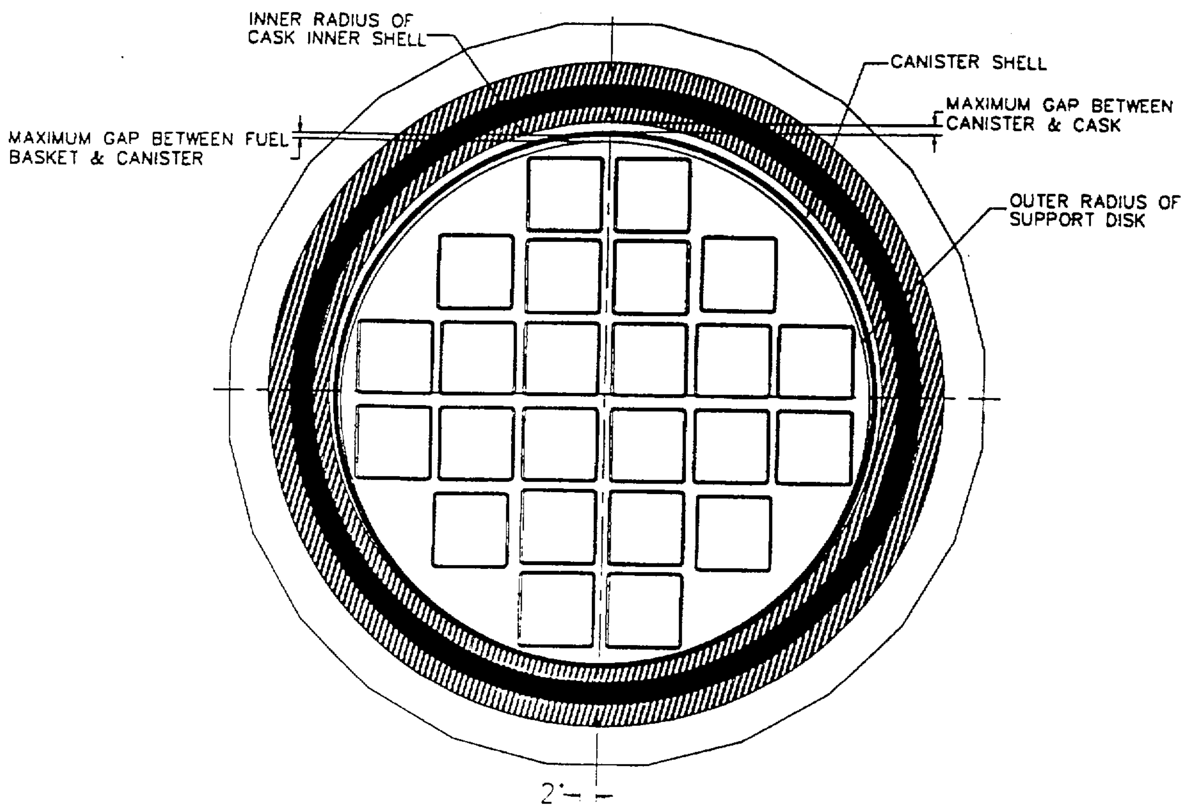


Figure 3.1-2 Definition of Gap Between Basket, Canister, and Inner Shell for Horizontal Position of the Universal Transport Cask Containing BWR Fuel

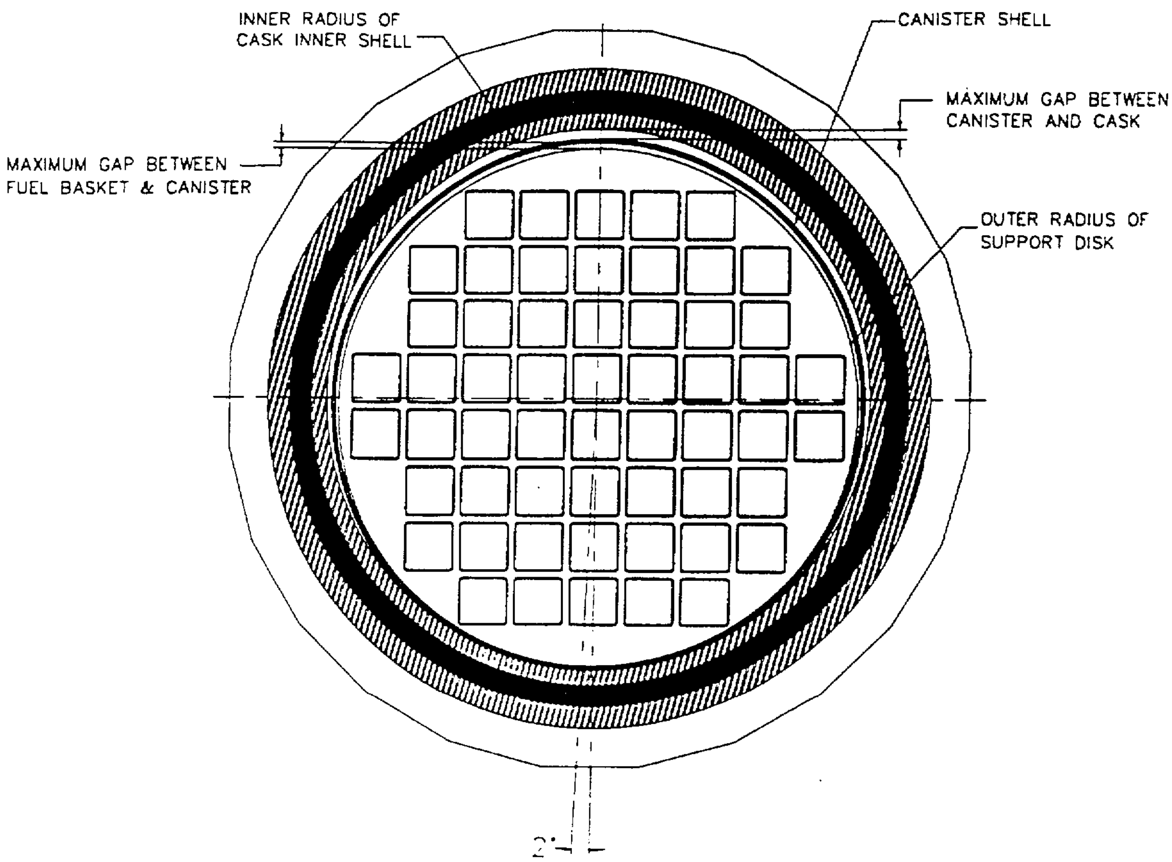


Table 3.1-1 Thermal Analysis Bounding Conditions - Normal Conditions of Transport

Condition	**Value**
Ambient Temperature per 10CFR71:	
Maximum (hot conditions)	100°F
Minimum (cold conditions and minimum temperature)	-40°F
Insolance (for 12 hr per day) per 10CFR71:	
Horizontal Flat Surfaces (facing up)	2950 Btu/ft^2
Curved Surfaces	1475 Btu/ft^2
Vertical Flat Surfaces	737 Btu/ft^2
PWR Fuel Assembly Decay Heat, Total:	20 kW
PWR Fuel Peaking Factor	1.1
BWR Fuel Assembly Decay Heat, Total:	16 kW
BWR Fuel Peaking Factor	1.22

3.2 Summary of Thermal Properties of Materials

The transfer of heat within the cask is primarily accomplished by conduction and radiation. The thermal conductivities and emissivities of the materials of construction are required for the thermal analysis of the cask. In addition, certain convective properties are required for modeling of convective heat transfer at the cask exterior surface. These properties are presented in Tables 3.2-1 through 3.2-13. These tables include only the properties of materials that form the heat transfer pathways employed in the finite element models. Materials for small components, such as valves and trunnions, which are not directly modeled are not included in the property tabulation.

3.2.1 Conductive Properties

The values for the conductivities of the materials are given in Tables 3.2-1 through 3.2-13.

3.2.2 Radiative Properties

3.2.2.1 Governing Radiation Principle

Radiation heat transfer between two nodes, i (hotter node) and j (colder node), is accounted for by the expression:

$$q_r = \sigma \varepsilon A F \left(T_i^4 - T_j^4 \right)$$

where

σ = Stefan-Boltzman constant
= 1.19×10^{-11} Btu/hr-in^2-°R^4
ε = emissivity
A = surface area
F = shape factor for the surfaces
T_i = temperature of i-th node
T_j = temperature of j-th node.

3.2.2.2 Radiation from Cask Surface

The expression shown previously is considered to be the governing equation for radiation within the cask and from the cask to the environment. Radiation heat transfer from the surface of the cask can be incorporated in the model by modifying the convection coefficient as follows:

$$Q_t = q_r + q_c$$

where q_r is specified as shown for the radiation heat transfer and q_c, which is the heat transfer by convection, is expressed as

$$q_c = h_c A(T_i - T_j)$$

where h_c = film coefficient (Btu/hr - in^2 - °F).

The q_r can be rewritten as

$$q_r = \sigma\varepsilon AF(T_i^2 + T_j^2)(T_i + T_j)(T_i - T_j)\ .$$

By combining both expressions,

$$Q_t = (\sigma\varepsilon F(T_i^2 + T_j^2)(T_i+T_j) + h_c)A(T_i - T_j)$$

or

$$Q_t = h_{eff} A(T_i - T_j)$$

where $h_{eff} = \sigma\varepsilon F(T_i^2 + T_j^2)(T_i+T_j) + h_c$

The effective convection coefficient used for the cask surface (h_{eff}) now includes the radiation heat transfer. In this application, the form factor (F) is taken to be unity.

3.2.2.3 Radiation Across Gaps Within the Cask

The gaps represented in the cask model are small compared with the surfaces separated by the gap. These gaps for both the PWR and the BWR casks are provided in Table 3.2-14.

The total heat transfer can be expressed as the sum of the radiation and the conduction processes.

$$Q_t = q_r + q_k$$

where q_r is specified as shown for the radiation heat transfer and q_k, which is the heat transfer by conduction, is expressed as

$$q_k = \frac{KA}{g}(T_i - T_j)$$

where:

g = gap distance (between two surfaces defined by nodes i and j)
K = conductivity of gas in gap
A = cross sectional area for heat conduction

By combining the two expressions (for q_k and q_r) and factoring out the term $A(T_i - T_j)/g$,

$$Q_t = [g\sigma\varepsilon F(T_i^2 + T_j^2)(T_i+T_j) + K][A(T_i - T_j)/g]$$

or

$$Q_t = K_{eff}A(T_i - T_j)/g$$

where $K_{eff} = g\sigma\varepsilon F(T_i^2 + T_j^2)(T_i+T_j) + K$.

The material conductivity used in the analysis for the elements that constitute the gap includes the heat transfer by both conduction and radiation. Because the gap is small compared with the disk thickness, the form factor (F) is taken to be unity.

3.2.2.4 Radiation from the Top of the Canister

The radiation heat transfer from the top of the canister is based on the expression:

$$q_r = \sigma \varepsilon AF(T_i^4 - T_j^4)$$

where the area (A) corresponds to the basket, lids, and spacer areas, and (ε) corresponds to the emissivities.

On the basis of the preceding equation, the radiation heat transfer is modeled by using radiation link elements in the cask three-dimensional model for the following locations:

1. From top of fuel region to bottom surface of canister shield lid;
2. From bottom of fuel region to top surface of canister bottom plate; and,
3. From exterior surfaces of the fuel tubes to the inner surface of the canister shell.

3.2.3 Convective Properties

A convective heat transfer coefficient, h_c, is associated with each surface where convection operates. Several surfaces must be considered. Surfaces vary by shape and orientation. Only the cylindrical surface of the cask takes part in the heat removal process, because the ends of the cask are thermally "insulated" from the environmental ambient thermal sink by the impact limiters.

The cask body surface is represented by a horizontal cylinder in air. From the Standard Handbook for Mechanical Engineers [5], the heat transfer coefficient, h_c, is:

$$h_c = 0.19\ \Delta T^{0.33}\ \text{BTU/hr-ft}^2\text{-°F, for } D^3\Delta T > 100$$

where:

ΔT = temperature difference between surface and air, °F
D = cylinder diameter, ft

For D = 7.667 ft and $\Delta T > 100°F$, the value of $D^3\Delta T > 45{,}000$ is significantly larger than 100. The expression can be converted into

$$h_c = 0.00132\ \Delta T^{0.33}\ \text{Btu/hr-in}^2\text{-°F}\ .$$

Table 3.2-1 Thermal Properties of Solid Neutron Shield (NS-4-FR)

Property (units)	**Value**
Conductivity (Btu/hr-in-°F) [6]	0.0311
Density (Borated) (lbm/in^3) [6]	0.0589
Specific heat (Btu/lbm-°F) [6]	0.39

Table 3.2-2 Thermal Properties of Stainless Steel

Property	Type 304 and 304L Temperature 100°F	200°F	400°F	550°F	750°F
Conductivity (Btu/hr–in-°F) [14]	0.7250	0.7750	0.8667	0.9250	1.0000
Density (lbm/in^3) [14]	0.2896	0.2888	0.2872	0.2855	0.2839
Specific Heat (Btu/lbm-°F) [14]	0.1158	0.1207	0.1272	0.1320	0.135
Emissivity [23]	← 0.36 (300°F) →				

Property	17-4PH, Type 630 Temperature 100°F	200°F	500°F	700°F
Conductivity (Btu/hr -in-°F) [14]	0.8417	0.8833	1.0167	1.1000
Emissivity [15]	← 0.58 →			

Table 3.2-3 Thermal Properties of Carbon Steel

Property	Temperature				
	100°F	**200°F**	**400°F**	**500°F**	**700°F**
Conductivity (Btu/hr–in-°F) [14]	1.992	2.033	2.017	1.975	1.867
Density (lbm/in^3) [24]	← 0.284 →				
Specific Heat (Btu/lbm-°F) [9]	← 0.113 →				
Emissivity [16]	← 0.80 →				

Table 3.2-4 Thermal Properties of Chemical Copper Lead

Property	Temperature			
	209°F	**400°F**	**581°F**	**630°F**
Conductivity (Btu/hr -in-°F) [18]	1.6308	1.5260	1.2095	1.0079
Density (lbm/in^3) [18]	0.411			
Specific Heat (Btu/lbm-°F) [18]	0.03			
Emissivity [16]	0.38 (75°F)			

Table 3.2-5 Thermal Properties of Type 6061-T651 Aluminum Alloy

Property (units)	Temperature				
	200°F	**300°F**	**400°F**	**500°F**	**600°F**
Conductivity (Btu/hr-in-°F) [14]	8.25	8.38	8.49	8.49	8.49
Emissivity [15]	← 0.22 →				

Table 3.2-6 Thermal Properties of Helium

Property	Temperature			
	200°F	400°F	600°F	800°F
Conductivity (Btu/hr -in-°F) [13]	0.00808	0.00942	0.01075	0.01150
Specific Heat (Btu/lbm-°F) [13]	← 1.24 →			
Density (lbm/in^3) [13]	4.83E-06	3.70E-06	3.01E-06	2.52E-06

Table 3.2-7 Thermal Properties of Dry Air

Property	Temperature			
	100°F	**300°F**	**500°F**	**700°F**
Conductivity (Btu/hr -in-°F) [13]	0.00128	0.00161	0.00193	0.00223
Density (lbm/in^3) [13]	4.11E-05	3.25E-05	2.38E-05	1.97E-05
Specific Heat (Btu/lbm-°F) [13]	0.240	0.244	0.247	0.253

Table 3.2-8 Thermal Properties of Copper

Property	Temperature		
	32°F	**212°F**	**392°F**
Conductivity (Btu/hr -in-°F) [17]	18.58	18.25	18.00
Density (lbm/in^3) [17]	← 0.32 →		
Specific Heat (Btu/lbm-°F) [17]	← 0.09 →		

Table 3.2-9 Thermal Properties of Zircaloy and Zircaloy-4 Cladding

Property	Temperature			
	392°F	**572°F**	**752°F**	**932°F**
Conductivity (Btu/hr -in-°F) [25]	0.69	0.73	0.80	0.87
Density (lbm/in^3) [26]	← 0.237 →			
Specific Heat (Btu/lbm-°F) [25]	0.072	0.074	0.076	0.079
Emissivity [25]	← 0.75 →			

Table 3.2-10 Thermal Properties of Fuel (UO_2)

Property	Temperature				
	100°F	**257°F**	**482°F**	**707°F**	**932°F**
Conductivity (Btu/hr–in-°F) [25]	0.38	0.347	0.277	0.236	0.212
Density (lbm/in^3) [26]	← 0.396 →				
Specific Heat (Btu/lbm-°F) [25]	0.057	0.062	0.067	0.071	0.073
Emissivity [25]	← 0.85 →				

Table 3.2-11 Thermal Properties of BORAL Composite Sheet

Property	Temperature	
	100°F	**500°F**
Conductivity (Btu/hr –in-°F)		
Aluminum Clad [19]	7.805	8.976
Core Matrix: PWR (calculated) BWR (calculated)	 3.45 6.60	 3.05 7.23
Emissivity [20]	← 0.15 →	

Table 3.2-12 Thermal Properties of Redwood (Air Dry)

Property (units)	**Value**
Conductivity (Btu/hr-in-°F):	
Parallel to Grain [23]	0.012
Transverse to Grain [23]	0.005
Density (lbm/in^3) [23]	0.014

Table 3.2-13 Thermal Properties of Fiberfrax Ceramic Fiber Paper

Property[1]	**Value**
Thermal Conductivity [21]	0.40 Btu in/hr ft^2 °F
Temperature Use Limit	≥ 2300°F

1. Grades 550, 880 and 970.

Table 3.2-14 Gaps Within the Universal Transport Cask

Gap Location	**Gap (in.)**	
	Cask with PWR Fuel Canister	**Cask with BWR Fuel Canister**
Gap between support disk and canister shell	0.155	0.155
Gap between canister and inner shell	0.275	0.275
Gap between lead gamma shield and inner shell	0.015	0.015
Gap between heat transfer disk and canister shell	0.280	0.315
Gap between canister bottom plate and top of spacer	0.125	0.25
Gap between bottom forging and spacer	*	0.25
Gap between spacer and canister bottom plate	0.125	0.25

* Only the base disk of the spacer is modeled in the PWR cask analysis. The cylindrical shells attached to the base disk are neglected.

THIS PAGE INTENTIONALLY LEFT BLANK

3.3 Technical Specifications for Components

This section provides a discussion of the major components that provide radiation protection and of the other cask components that must be maintained within their safe operating temperature ranges.

3.3.1 Radiation Protection Components

Radiation protection in the Universal Transport Cask is provided by gamma and neutron shielding. The primary gamma radiation shielding components are the materials used in fabricating the multiwalled body, the end forgings of the cask body, and the cask lid. The multiwalled body consists of the lead cast in place between the inner and outer stainless steel shells. Neutron shielding is provided in the radial direction by solid neutron shield material and in the bottom end direction by a 1-inch-thick disk in the bottom of the cask. The neutron shields are borated to suppress secondary gamma generation. The capture of neutrons by many materials produces a secondary gamma ray that must also be shielded; however, when ^{10}B absorbs a neutron, the alpha particle emitted is stopped locally. Thus, the secondary gamma dose rate is minimized. The radiation protection components are analyzed for normal conditions of transport in Section 3.4 and for hypothetical accident event conditions in Section 3.5.

3.3.2 Safe Operating Ranges

Six major components must be maintained within their safe operating temperature ranges: the cask lid O-rings, the lower drain port O-rings, the lead gamma shield, the NS-4-FR solid neutron shield, the aluminum heat transfer disks, and the support disks. The support disks for the PWR basket are fabricated from SA 693, 17-4 PH, Type 630 stainless steel. For the BWR basket, the support disks are fabricated from SA 533, Type B carbon steel. The safe operating ranges for these components are as follows:

Component	Safe Operating Range
Cask Lid and Lower Drain Port O-rings (EPDM)	-65°F to +300°F (up to +375°F for 10 hours for accident conditions)
Lead gamma shield	-40°F to +600°F
Radial NS-4-FR neutron shield	-40°F to +300°F
Aluminum heat transfer disks	-40°F to +700°F
PWR basket support disks	-40°F to +650°F
BWR basket support disk	-40°F to +700°F

The safe operating range of the EPDM O-rings, obtained from the technical information presented in Section 4.5.2, ensures that the O-rings maintain their ability to perform their sealing function. The analyses of Sections 3.4 and 3.5 show that the temperatures of the O-rings are maintained within the safe operating range during normal conditions of transport and hypothetical accident conditions.

The safe operating range of the lead gamma shield is based on preventing the lead from reaching its melting point of 620°F [5]. To preclude localized lead temperatures from exceeding their safe operating range, Fireblock Protective Coating (FPC) is used to insulate the lead from the high temperatures that occur during the 10 CFR 71 hypothetical fire accident. The FPC is included for additional assurance of safety. A 0.125-inch layer of the material is located around the top corner of the lead gamma shield and at the bottom area where the lead is adjacent to the bottom ring, i.e., above and below the coverage provided by the radial neutron shield.

The maximum operating temperature limit of the NS-4-FR solid neutron shield material to ensure sufficient neutron shielding capacity is determined by the manufacturer to be 330°F [6]. The peak calculated temperatures experienced in the Universal Transport Cask neutron shield, with helium as the cover gas in the cask cavity and helium inside the canister (which is the design basis configuration), are 293°F for the cask containing PWR fuel and, 286°F for the cask containing BWR fuel. The peak temperatures occur only at localized areas. For the remainder of the neutron shield material, temperatures are well below the 293°F and 286°F values for the casks containing the PWR and BWR fuels, respectively.

Because the temperature of the radial neutron shield exceeds the temperature allowable during the 30-min hypothetical accident fire, the radial neutron shield is considered lost after the fire accident for shielding purposes. The necessity for the neutron shield material to remain within its safe operating range is thus eliminated (See Chapter 5.0 for a discussion of the effect of a loss of the neutron shield on the cask dose rates). The radial neutron shield is assumed to remain intact throughout the hypothetical fire and to be removed and replaced with air at the end of the fire for the fire analysis. A thermal transient sensitivity analysis was performed using the finite element model described in Section 3.5.1. The analysis was performed with the radial neutron shield (NS-4-FR) removed during the 30-minute fire. The analysis results indicate a slight reduction in the maximum temperature for all components. The maximum reduction occurs at the lead (5°F) and the inner shell (4°F). Therefore, it is conservative to perform the thermal transient analysis with the radial neutron shield in place during the 30-minute fire and remove it at the end of the fire event. This shows that larger quantities of energy are transferred into the cask during the fire accident and lesser quantities are rejected from the cask after the 30-minute fire.

The primary consideration in establishing the safe operating range of the aluminum heat transfer disk is maintaining the integrity of the aluminum. According to the MIL-HDBK-5F [7], it can be shown that aluminum at 700°F retains component performance.

The support disks must support, and control the geometry of, the stored spent fuel in transport. The thermal performance properties of the 17-4 PH stainless steel PWR support disk, and the SA 533, Type B carbon steel BWR support disk, are taken from the ASME Code, Section II, Part D, "Properties" [14].

THIS PAGE INTENTIONALLY LEFT BLANK

3.4 Thermal Evaluation for Normal Conditions of Transport

The finite element method is used to evaluate the thermal performance of the Universal Transport Cask for normal conditions of transport as specified in 10 CFR 71. The general-purpose finite element analysis program ANSYS Revision 5.5 [5] is used to perform the finite element evaluations.

The normal conditions of transport used in the thermal evaluation of the cask are as follows:

1. Hot Conditions: maximum decay heat generation, ambient temperature = 100°F, solar insolance (solar insolance applied according to Table 3.1-1)

2. Cold Conditions: maximum decay heat generation, ambient temperature = -40°F, no solar insolance

3. Minimum Temperature Conditions: no decay heat generation, ambient temperature = -40°F, no solar insolance (no analysis is performed for this condition because all component temperatures will be -40°F for steady state conditions).

The objectives of the cask thermal analyses under normal conditions of transport are as follows:

1. Demonstrate that the cask can safely maintain the design basis temperatures required for fuel cladding integrity under the range of thermal conditions expected during normal conditions

2. Demonstrate that cask components important to safety are maintained within their safe operating temperature ranges

3. Provide thermal input to the structural analyses.

The first objective is met by demonstrating that the cask maintains maximum fuel rod cladding temperatures below 716°F (380°C) during normal conditions.

The second objective is met by comparing the results of the analyses with the safe operating ranges established in Section 3.3.

The third objective is met by using the results of the thermal analyses (as direct import of ANSYS temperature data, as maximum and minimum component temperatures, or as allowable look-up temperatures) as input to the structural analyses, which demonstrate that the combined load stresses are within allowable limits.

3.4.1 Thermal Models

The finite element method is used to evaluate the Universal Transport Cask for exposure to normal conditions of transport as specified in 10 CFR 71. This section describes the finite element models used in the thermal evaluation of the cask under normal conditions of transport. Separate three-dimensional finite element models are used to evaluate the cask loaded with PWR fuel and the cask loaded with BWR fuel. In addition, a separate model is used to determine the volumetric average temperature of the cask impact limiter for normal conditions. The analyses for normal conditions of transport consider the transport cask oriented horizontally.

For each fuel-loading configuration, the cask is evaluated for normal conditions of transport using a three-dimensional half-symmetry (180°) finite element model of the loaded cask including internal components. The three-dimensional finite element models of the cask/internal components both comprise five parts: basket with fuel tube and fuel assembly; canister; spacer (between canister bottom and shell bottom forging); transport cask body; and gases between components. To model the cask in a horizontal orientation, the fuel basket in each model is modeled in contact with the canister on one side which, in turn, is in contact with the inner shell of the cask on one side—thus simulating no gap on one side of the basket and canister and a maximum gap at the opposite side (see Figure 3.1-1 for the PWR fuel configuration and Figure 3.1-2 for the BWR fuel configuration).

Gaps within the models are adjusted to account for differential expansion on the basis of thermal and defined physical contact conditions. Solar insolance, natural convection and thermal radiation boundary conditions based on ambient temperature are applied to the outer surface of

the cask (the sections of the cask body covered by the impact limiters are modeled as adiabatic). The three-dimensional finite element model for the cask loaded with PWR fuel is described in Section 3.4.1.1.1. The three-dimensional finite element model for the cask loaded with BWR fuel is described in Section 3.4.1.2.1.

The models of the cask/internal components (both PWR and BWR) are constructed of ANSYS three-dimensional, solid brick, thermal conduction elements (SOLID70) to model heat conduction/combined conduction and thermal radiation, as well as two-node thermal radiation link elements (LINK31) to model thermal radiation. The analyses of the cask models correspond to steady-state conditions.

In the three-dimensional cask models, the fuel assemblies are modeled as homogeneous regions with effective temperature-dependent thermal conductivity. The effective thermal conductivity of the fuel region in the plane perpendicular to the major axis of the cask is determined for each fuel (PWR and BWR) by using two-dimensional finite element models representing the cross-section of a single fuel assembly. The two-dimensional finite element models of the fuel assemblies consist of the UO_2 fuel pellets; Zircaloy cladding; and gas between the fuel pellets and cladding and between the fuel rods (fuel pellet/cladding). Heat generation rates (multiplied by the respective peaking factors for each fuel) are applied to the elements representing the UO_2 and an isothermal temperature condition is applied to the edges of the model representing the outer surfaces of the fuel assembly. The effective conductivity of the fuel assembly is then calculated by determining the maximum temperature in the fuel and using a closed form expression for a square with uniform heat generation. The two-dimensional finite element model of the PWR fuel is also described in Section 3.4.1.1.2. The two-dimensional finite element model of the BWR fuel is also described in Section 3.4.1.2.2.

The models of the fuel assemblies are constructed of ANSYS two-dimensional thermal elements (PLANE55) to model heat conduction and two-node thermal radiation link elements (LINK31) to model thermal radiation. The analyses of the fuel assemblies models are steady-state.

Additionally, the fuel tube walls and BORAL plate are modeled in the three-dimensional cask models as homogeneous regions by using effective thermal conductivity properties. The

effective thermal conductivity of the fuel tube walls and BORAL plate is determined for each fuel tube (PWR and BWR) by using two-dimensional finite element models representing the cross-section of a typical fuel tube. The two dimensional models of the fuel tube walls and BORAL plate consist of the stainless steel tube wall; the BORAL sheet, which is composed of a sheet of boron sandwiched between aluminum sheets; the stainless steel sheet covering the BORAL plate; and the gaps separating these components. A heat flux is applied to the inner face of the composite tube wall while a temperature is applied to the outer face. The change in temperature is then used to calculate the effective thermal conductivity. This method treats the thermal resistance of the different layers as being in series. The effective thermal conductivity for heat condition parallel to the axis of the cask is computed as a weighted average based on the thickness of each layer. The two-dimensional finite element model of the PWR fuel tube is described in Section 3.4.1.1.3. The two-dimensional finite element model of the BWR fuel tube is described in Section 3.4.1.2.3.

The models of the tube wall and BORAL plate are constructed of ANSYS two-dimensional thermal elements (PLANE55) to model heat conduction and two-node thermal radiation link elements (LINK31) to model thermal radiation. The analyses of the fuel tube and BORAL plate models are steady-state.

A separate thermal analysis from the cask models is performed to determine the volumetric average temperature of the cask impact limiters. The impact limiters are not explicitly modeled in the cask thermal analyses previously discussed—the cask surfaces covered by the impact limiters are modeled as adiabatic. The impact limiter thermal model consists of an axis-symmetric finite element model of one impact limiter, the cask lid, the cask upper forging, the fire block inside the impact limiter shell, and the air gap between the cask upper forging and the impact limiter.

3.4.1.1 Analytical Models: Cask with PWR Fuel Canister

The thermal analysis of the cask transporting PWR fuel uses three finite element ANSYS models as previously described. A three-dimensional model is employed to evaluate the cask in a horizontal position with the basket in contact with the canister, which, in turn, is in contact with

the cask inner shell. The fuel regions and the fuel tubes with BORAL plates in this model are modeled by using effective conductivities. The effective conductivity of the fuel is determined by a second model, which is a detailed two-dimensional thermal model of the fuel assembly. The effective conductivities of the fuel tube wall and BORAL plate are calculated by using a third model, which is a two-dimensional thermal model of the fuel tube. The three ANSYS thermal models are described in the following paragraphs.

3.4.1.1.1 Three-Dimensional Cask Model: Cask with PWR Fuel Canister

The three dimensional Universal Transport Cask model is a half-symmetry finite element model constructed by using ANSYS Revision 5.5. The model considers the fuel assemblies, fuel tubes, stainless steel support disks, aluminum heat transfer disks, canister shell, lids and bottom plate, spacers at the bottom of the canister, cask inner shell, lead, outer shell, neutron shield, and neutron shield shell. The gaps between the individual components are also considered. The ANSYS model is shown in Figure 3.4-1. As shown in Figure 3.4-1, the internal cavity of the canister contains the active fuel region: the top and bottom end fittings of the fuel assemblies, fuel tubes enclosing the fuel assemblies and the top and bottom end fittings, and the bottom weldment.

For the PWR configuration, two analyses are performed. Gas inside the canister is modeled as helium in one model and air in another model. Gas inside the cask cavity is modeled as helium, because the cavity will be backfilled with helium following fuel loading prior to transport. The finite element model is constructed of ANSYS three-dimensional, solid brick, thermal conduction elements (SOLID70) to model heat conduction/combined conduction and thermal radiation and two-node thermal radiation link elements (LINK31) to model thermal radiation. The principal gaps applied to the model are shown in Figure 3.1-1 and described in Section 3.2.2.3. In establishing these gaps, the differential thermal expansion between the components is considered. The gap values selected are conservative.

Because the canister is in the horizontal position during transport, the elements for the canister shell are shifted downwards to simulate contact with the inner shell of the cask. Similarly, the support disks and the heat transfer disks are shifted downward to simulate contact with the canister shell. As shown in Figure 3.1-1, a 2-degree contact is considered for the gaps between

the canister shell and the cask inner shell and between the support disk and the canister shell. At the 2-degree contact region in the model, an element 0.005-inch thick (in the radial direction) is modeled between the elements of the canister shell and cask inner shell, and between the elements for the support disk and canister shell. To simulate the contact condition, a conductivity of 100 Btu/hr-in-°F is assumed for the element. The value of conductivity used has a negligible effect on the thermal analysis results, since the thermal resistance across the element is negligible compared to the thermal resistance of the canister shell or the cask inner shell because the thickness of the element is only 0.005 inch. The aluminum heat transfer disks are assumed to have only a line contact with the canister shell because the heat transfer disks are not subjected to any loads other than their own weight.

To account for differential thermal expansion, gaps within the model are adjusted on the basis of temperature and defined physical contact conditions. Solar insolance and ambient temperature conditions are applied to the neutron shield shell when appropriate. Insolance is used at the exterior surface of the cask and is based on the amount of insolation required by 10 CFR 71 to be applied over a 12-hr period evaluated in the steady state (applied over 24 hr simulating 12-hr period of solar exposure and 12-hr period of no solar exposure). The heat flux resulting from insolation on a curved surface is calculated as follows:

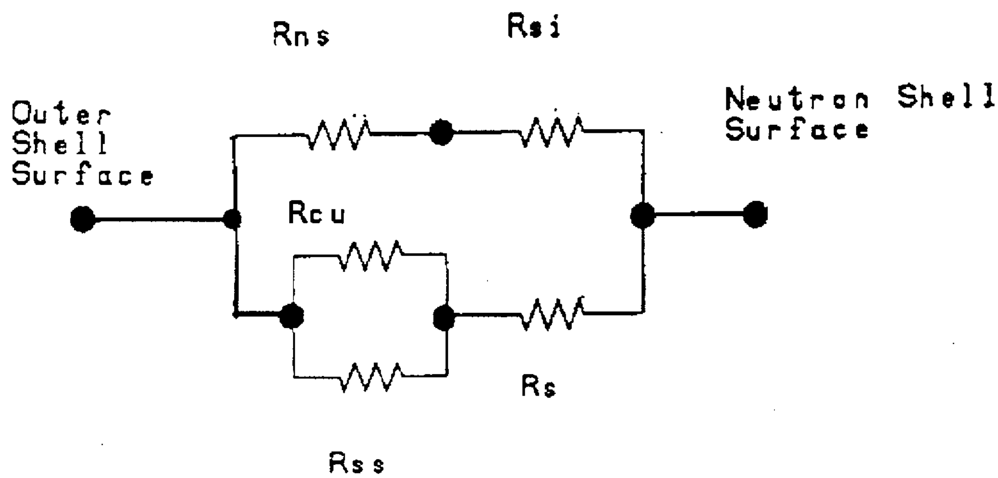
$$1475 \frac{\text{Btu}}{12\text{ hr - ft}^2} \times \frac{12\text{ hr}}{24\text{ hr}} \times \frac{1\text{ ft}^2}{144\text{ in}^2} = 0.427\text{ Btu/hr-in}^2.$$

Multiplying this value by the emissivity of the cask surface, $\varepsilon = 0.36$, gives a heat flux resulting from insolance on curved surfaces of 0.154 Btu/hr-in^2. Using the same method and a heat flux of 2,950 Btu/12 hr-ft^2 (0.853 Btu/hr-in^2) gives a heat flux resulting from insolance on flat surfaces of 0.307 Btu/hr-in^2. Applying one-half of the required 12-hr insolance over a 24-hr period to achieve a steady state solution, as has been done previously in transport cask licensing, is conservative.

The model is analyzed to determine the maximum temperatures for the basket, canister, cask shells, radial shielding, and surface conditions under normal conditions of transport. All material properties are shown in Tables 3.2-1 through 3.2-13.

The fuel regions (inside tubes) are modeled as homogeneous regions with effective conductivities, determined by the two dimensional fuel model as described in Section 3.4.1.1.2. The fuel assembly tube and the BORAL plate, including air gaps on both sides of the BORAL sheet and the gap between the stainless steel cladding for BORAL and disk, are modeled as one element thick with effective conductivities, as established by using the two-dimensional tube model discussed in Section 3.4.1.1.3. Only the spacer plate is modeled with the spacer concentric cylinders. Therefore, only conduction through the helium (modeled using SOLID70 elements) and radiation from the spacer plate to the cask bottom (modeled using LINK31 thermal radiation links) are conservatively modeled.

The neutron shield of the Universal Transport Cask, consisting of NS-4-FR, steel, and Cu/SS fins, is also modeled with effective conductivities. The radial neutron shield effective conductivity is calculated using an electrical resistance analogy. The equivalent circuit corresponding to Cu/SS, fin, NS-4-FR and Silicon foam is shown below.



Rns = NS-4-FR material
Rsi = silicon foam between NS-4-FR and neutron shield shell
Rcu = 6 mm copper plate
Rss = 8 mm stainless steel
Rs = stainless steel connection to neutron shield shell

The axial conductivity, specific heat, and density are calculated on the basis of a weighted average of the axial cross sectional area and property. Conductivity of the neutron shield material NS-4-FR (0.031 Btu/hr-inch-°F) is used as the conductivity in the circumferential direction. The effective thermal conductivities for the neutron shield are :

Temperature	**100°F**	**225°F**	**350°F**
Radial Conductivity (Btu/hr in °F)	0.380	0.382	0.383
Axial Conductivity (Btu/hr in °F)	0.425	0.424	0.421
Specific heat (Btu/hr in °F)	0.39	0.39	0.39
Density lbm/in^3	0.0589	0.0589	0.0589

In the model, radiation heat transfer is considered from the top of the fuel region to the bottom surface of the canister shield lid, from the bottom of the fuel region to the top surface of the canister bottom plate, and from the exterior surfaces of the fuel tubes to the inner surface of the canister shell. This radiation is modeled by using LINK31 radiation elements. Radiation across gaps in the model is described in Section 3.2.2.3 and 3.2.2.4.

Radiation at the neutron shield shell surface to ambient is combined with the convection effect by using the method described in Section 3.2.2.2. The convection heat transfer coefficient is calculated on the basis of the formula shown in Section 3.2.3. Effective emissivities are used for all radiation calculations, with the form factor taken to be unity. Effective emissivity is computed by using the following formula [9] based on corresponding material emissivities:

$$\varepsilon_{eff} = 1/(1/\varepsilon_1 + 1/\varepsilon_2 - 1)$$

Solar insolance is applied to the neutron shield shell surface for the "Hot" condition (ambient temperature = 100°F). A cosine distribution is considered for the heat flux since the cask side surface is subjected to maximum insolation at the top and minimum (zero) insolation at the bottom while in the horizontal position. The heat flux is determined based on the average value of 0.154 Btu/hr-in^2 for the curved surface discussed previously.

Volumetric heat generation (Btu/hr-inch^3) is applied to the active fuel region on the basis of a total heat load of 20 kW, with an active fuel rod length of 144 inches, and an axial power distribution as shown in Figure 3.4-2. While CE 14x14 has a shorter fuel length of 128 inches, the corresponding total heat is only 15.7 kW and the heat density is 88% of the 20 kW over 144 inches. The 20kW over 144 inches is considered to be controlling.

3.4.1.1.2 Two-Dimensional Fuel Assembly Model: PWR Fuel

The effective conductivity of the fuel is determined by a detailed two-dimensional finite element thermal model of the PWR 14x14 fuel assembly. Taking advantage of the symmetry of the cross-section of the fuel, the finite element model represents a one-quarter section of the fuel. The model includes the fuel pellets, cladding, gas between the fuel rods, and gas occupying the gap between the fuel pellets and cladding. Modes of heat transfer modeled include conduction and radiation between individual fuel rods for the steady-state condition. The model is shown in Figure 3.4-3. Thermal analyses of the other PWR fuel assemblies (i.e., 17x17, 16x16, and 15x15) are performed; however, because the PWR 14x14 fuel assembly results in the lowest effective thermal conductivities, only the analysis of that fuel assembly is presented in this section.

ANSYS PLANE 55 conduction elements and LINK31 radiation elements are used in the model, which includes a total of 49 fuel rods (representing a total of 196 fuel rods for the full cross-section). Each fuel rod consists of the pellet, Zircaloy cladding, and a gap between the pellet and clad. The gas in the gap between the pellet and clad, as well as the gas between the fuel rods, is modeled as air in one analysis and helium in another analysis. Radiation elements are defined

between rods and from rods to the boundary of the model (inside surface of the fuel tube). Radiation across the gap between the pellet and clad is conservatively ignored. Effective emissivities are determined by using the formula shown in Section 3.4.1.1.1.

The effective conductivity for the fuel is determined by using a two-step procedure. Using the fuel assembly model, a uniform temperature is applied to the exterior of the model (see Figure 3.4-3) in conjunction with the volumetric heat generation. From this analysis, the maximum temperature located at the center of the fuel assembly is determined. This maximum temperature occurs at the corner of the model, which represents the center of the entire fuel assembly.

A Sandia National Laboratory Report [10] defines an expression for use in determining the maximum temperature of a square cross section of an isotropic homogeneous fuel with uniform volumetric heat generation. At the boundary of this square cross section, the temperature is constrained to be uniform. The expression for the maximum temperature is given by:

$$T_c = T_e + 0.29468 \frac{Q a^2}{K_{eff}} 1$$

where:

T_c = temperature at center of fuel (°F)
T_e = temperature applied at exterior of fuel (°F)
Q = volumetric heat generation rate (Btu/hr-in^3)
a = half-length of square cross section of fuel (inch)
K_{eff} = effective thermal conductivity for isotropic homogeneous fuel material (Btu/hr-in-°F).

Using the maximum temperature, located at the center of the fuel, from the detailed fuel assembly model, the preceding expression is used to determine the K_{eff} for an isotropic homogeneous representation of the fuel assembly.

Volumetric heat generation based on the design heat load of 20 kW with a peaking factor of 1.1 is applied to the fuel pellets. The temperature at the boundary of the model is constrained to be uniform. The effective conductivity is determined on the basis of the heat generated and the

temperature difference from the center of the model to its edge. The temperature-dependent effective properties are established by using different boundary temperatures. The effective conductivity in the axial direction of the fuel assembly is calculated on the basis of a weighted average of the axial cross sectional area.

3.4.1.1.3 Two-Dimensional Fuel Tube Model: PWR Fuel

The effective conductivity of the fuel tube and BORAL plate, which is used in the three-dimensional canister model, is determined by the two-dimensional fuel tube model. As shown in Figure 3.4-4, this model includes the fuel tube, the BORAL plate (including the core matrix sandwiched by aluminum claddings), air gaps on both sides of the BORAL plate, and an air gap between the stainless steel cladding for the BORAL plate and the support disk or heat transfer disk. The BORAL plate in the PWR fuel tube is composed of 62.34% B_4C and 37.66% aluminum.

ANSYS PLANE55 conduction elements and LINK31 radiation elements are used to construct the model, which consists of eight layers of conduction elements and six radiation elements that are defined at the air gaps (two per gap). The thickness of the model (x-direction) is the distance measured from the inside dimension of the fuel tube to the inside dimension of the slot in the support disk (assuming that the fuel tube is located at the center of the disk slot). The tolerance of the BORAL plate core thickness, 0.003 inch, is used as the gap size for both sides of the BORAL plate. The model height is defined to be the same dimension as the model thickness.

A heat flux is applied at the left side of the model and the temperature at the right boundary of the model is constrained. The heat flux is determined on the basis of design heat load of 20 kW with a peaking factor of 1.1. The maximum temperature of the model (at the left boundary where the heat flux is applied) is calculated by using ANSYS. The effective conductivity through the thickness of the tube is determined by using the following equation:

$$q = K_{eff}(A/L)\,\Delta T$$

or $K_{eff} = qL/(A\ \Delta T)$

where:

q = heat rate applied to inner surface of fuel tube (Btu/hr)
A = area (in^2)
L = thickness of composite tube model (in)
ΔT = temperature difference across the model (°F)
K_{eff} = effective conductivity (Btu/hr-in-°F).

The temperature-dependent conductivity for heat conduction through the wall (K_{eff}) is determined by varying the temperature constraint at the boundary of the model and then re-solving for the temperature difference. The effective conductivity for heat conduction parallel to the axis of the cask body or in the plane of the tube wall is calculated on the basis of the weighted average of the thickness and conductivity of the individual layers.

3.4.1.2 Analytical Models: Cask with BWR Fuel Canister

The finite element ANSYS models used in the thermal analysis of the cask transporting BWR fuel are similar to those used in the thermal analysis of the cask with PWR fuel canister discussed in previous sections. A three-dimensional model is employed to evaluate the cask in a horizontal position with the basket in contact with the canister, which, in turn, is in contact with the cask inner shell. The fuel regions and the fuel tubes with BORAL plates are modeled by using effective conductivities. A detailed two-dimensional thermal model of the fuel assembly is used to determine the effective conductivity of the fuel. A two-dimensional thermal model of the fuel tube is used to calculate the effective conductivities of the fuel tube wall and BORAL plate. Another two-dimensional thermal model for the fuel tube is used to calculate the effective conductivity of the fuel tube wall with no BORAL plate present. These four ANSYS thermal models are described in the following sections.

3.4.1.2.1 Three-Dimensional Cask Model: Cask with BWR Fuel Canister

The three dimensional Universal Transport Cask model is a half-symmetry finite element model constructed by using ANSYS Revision 5.5. The model considers the fuel assemblies, fuel tubes, stainless steel support disks, aluminum heat transfer disks, canister shell, lids and bottom plate, spacers at the bottom of the canister, cask inner shell, lead, outer shell, neutron shield, and neutron shield shell. The ANSYS model is shown in Figure 3.4-5. As shown in the figure, the internal cavity of the canister contains the active fuel region: the top and bottom fittings of the fuel assemblies, fuel tubes enclosing the top and bottom fittings, and the first stainless steel support.

For the BWR configuration, two analyses are performed. Gas inside the canister is modeled as air in one analysis and helium in another analysis. Gas inside the cask cavity is modeled as helium, because the cavity will be backfilled with helium prior to transport. Conduction and radiation are modeled by using ANSYS "SOLID70" and "LINK31" elements, respectively. The principal gaps applied to the model are shown in Figure 3.1-2 and are described in Section 3.2.2.3. In establishing these gaps, the differential thermal expansion between the components is considered.

Because the canister is in horizontal position during transport, the elements for the canister shell are shifted downwards to simulate contact with the inner shell of the cask. Similarly, the support disks and the heat transfer disks are shifted downward to simulate contact with the canister shell. As shown in Figure 3.1-2, a 2-degree contact is considered for the gaps between the canister shell and the cask inner shell and between the support disk and the canister shell. This contact is simulated by using appropriate conductivity (100 Btu/hr-inch-°F) for elements at the contact locations. The aluminum heat transfer disks are assumed to have only a line contact with the canister shell because the heat transfer disks are not subjected to any loads other than their own weight.

To account for differential expansion, gaps within the model are adjusted on the basis of temperature and defined physical contact conditions. Solar insolance and ambient temperature conditions are applied to the neutron shield shell when appropriate. Insolance is used at the

exterior surface of the cask and is based on the amount of insolation required by 10 CFR 71 to be applied over a 12-hr period evaluated in the steady state (applied over 24 hr simulating 12-hr period of solar exposure and 12-hr period of no solar exposure). The heat flux resulting from insolation on a curved surface is calculated as follows:

$$1475\frac{\text{Btu}}{12\text{ hr - ft}^2} \times \frac{12\text{ hr}}{24\text{ hr}} \times \frac{1\text{ ft}^2}{144\text{ in}^2}3 = 0.427\text{ Btu/hr-in}^2$$

Multiplying this value by the emissivity of the cask surface, $\in = 0.36$, gives a heat flux resulting from insolance on curved surfaces of 0.154 Btu/hr-in^2. Using the same method and a heat flux of 2,950 Btu/12 hr-ft^2 (0.853 Btu/hr-in^2), gives a heat flux resulting from insolance on flat surfaces of 0.307 Btu/hr-in^2.

The model is analyzed to determine the maximum temperatures for the basket, canister, cask shells, radial shielding, and surface conditions under normal conditions of transport. All material properties are shown in Tables 3.2-1 through 3.2-13.

The fuel regions (inside tubes) are modeled as homogeneous regions with effective conductivities determined by the two dimensional fuel model as described in Section 3.4.1.2.2. All sides of the BWR fuel tubes do not contain the BORAL plate. Therefore, two different two-dimensional BWR fuel tube models are analyzed to establish the effective conductivities used in the three dimensional analysis of the cask with BWR fuel. The models consist of the BORAL plate (where applicable), including gas gaps on both sides of the BORAL sheet (where applicable), and the gap between the stainless steel cladding for the BORAL and the support disks and heat transfer disks. These models are discussed in Section 3.4.1.2.3.

The radial neutron shield of the transport cask for the BWR configuration is identical to PWR configuration. The modeling of the radial neutron shield is described in Section 3.4.1.1.

In the model, radiation heat transfer is considered from the top of the fuel region to the bottom surface of the canister shield lid, from the bottom of the fuel region to the top surface of the canister bottom plate, and from the exterior surfaces of the fuel tubes to the inner surface of the canister shell. This radiation is modeled by using LINK31 radiation elements. Radiation across gaps in the model is described in Sections 3.2.2.3 and 3.2.2.4.

Radiation at the neutron shield shell surface to ambient is combined with the convection effect by using the method described in Section 3.2.2.2. The convection heat transfer coefficient is calculated on the basis of the formula shown in Section 3.2.3. Effective emissivities are used for all radiation calculations, with the form factor taken to be unity. Effective emissivity is computed by using the following formula [9] based on corresponding material emissivities:

$$\varepsilon_{eff} = 1/\,(1/\varepsilon_1 + 1/\,\varepsilon_2 - 1)$$

Solar insolance is applied to the neutron shield shell surface for the "Hot" condition (ambient temperature = 100°F). A value of 0.154 Btu/hr-inch^2 is used as the heat flux at the neutron shield shell surface on the basis of the 1,475 Btu/hr-ft^2 heat flux for a curved surface. Calculation of the heat flux resulting from insolation on a curved surface is discussed earlier in this section.

Volumetric heat generation (Btu/hr-inch^3) is applied to the active fuel region on the basis of a total heat load of 16 kW, a shortest active fuel rod length of 144 inches, and an axial power with a peaking factor of 1.22 as shown in Figure 3.4-6.

3.4.1.2.2 Two-Dimensional Fuel Assembly Model: BWR Fuel

The effective conductivity of the fuel is determined by a detailed two-dimensional finite element thermal model of the BWR 9x9 fuel assembly. Taking advantage of the symmetry of the cross-section of the fuel, the finite element model represents a one-quarter section of the fuel. The model includes the fuel pellets, cladding, gas between the fuel rods, and gas occupying the gap between the fuel pellets and cladding. Modes of heat transfer modeled include conduction and radiation between individual fuel rods for the steady-state condition. The model is shown in

Figure 3.4-7. Thermal analyses of the other BWR fuel assemblies (i.e., 7x7 and 8x8) are performed; however, because the BWR 9x9 fuel assembly results in the lowest effective thermal conductivities, only the analysis of that fuel assembly is presented in this section.

ANSYS PLANE55 conduction elements and LINK31 radiation elements are used in the model, which includes a total of 20.25 fuel rods (representing a total of 81 fuel rods for the full cross-section). Each fuel rod consists of the pellet, Zircaloy cladding, and a gap between the pellet and clad. The gas in the gap between the pellet and clad, as well as the gas between the fuel rods, is modeled as air in one analysis and helium in another analysis. Radiation elements are defined between rods and from rods to the boundary of the model (inside surface of the fuel tube). Radiation effect at the gaps between the pellet and clad is conservatively ignored. Effective emissivities are determined by using the formula shown in Section 3.4.1.1.1.

The effective conductivity for the fuel is determined by using a two-step procedure. Using the fuel assembly model, a uniform temperature is applied to the exterior of the model (see Figure 3.4-7) in conjunction with the volumetric heat generation. From this analysis, the maximum temperature located at the center of the fuel assembly is determined. This maximum temperature occurs at the corner of the model, which represents the center of the entire fuel assembly.

A Sandia National Laboratory Report [10] defines an expression for use in determining the maximum temperature of a square cross section of an isotropic homogeneous fuel with uniform volumetric heat generation. At the boundary of this square cross section, the temperature is constrained to be uniform. The expression for the maximum temperature is given by:

$$T_c = T_e + 0.29468 \frac{Q a^2}{K_{eff}}$$

where:

T_c = temperature at center of fuel (°F)

T_e = temperature applied at exterior of fuel (°F)

Q = volumetric heat generation rate ($Btu/hr\text{-}in^3$)

a = half-length of square cross section of fuel (inch)

K_{eff} = effective thermal conductivity for isotropic homogeneous fuel material (Btu/hr-in-°F).

Using the maximum temperature, located at the center of the fuel, from the detailed fuel assembly model, the preceding expression is used to determine the K_{eff} for an isotropic homogeneous representation of the fuel assembly.

Volumetric heat generation based on the design heat load of 16 kW with a peaking factor of 1.22 is applied to the fuel pellets. The temperature at the boundary of the model is constrained to be uniform. The effective conductivity is determined on the basis of the heat generated and the temperature difference from the center of the model to its edge. The temperature-dependent effective properties are established by using different boundary temperatures. The effective conductivity in the axial direction of the fuel assembly is calculated on the basis of the material area ratio.

3.4.1.2.3 Two-Dimensional Fuel Tube Models: BWR Fuel

The fuel tubes in the BWR fuel basket differ from those in the PWR fuel basket in that not all sides of the fuel tubes contain BORAL. Therefore, two effective conductivity models are necessary—one fuel tube model with the BORAL plate (a total of 10 layers of materials) and another fuel tube model with a gas gap replacing the BORAL plate (a total of 4 layers of materials). Additionally, the BORAL plate in the BWR fuel tube is composed of 16.46% B_4C and 83.54% aluminum, whereas the BORAL plate in the PWR fuel tube is composed of a 62.34%—37.66% composition of B_4C and aluminum.

The effective conductivity of the fuel tube and BORAL plate, which is used in the three-dimensional canister model, is determined by a two-dimensional fuel tube model. As shown in Figure 3.4-8, this model includes the fuel channel, gas gaps between the fuel channel and fuel tube. the fuel tube, the BORAL plate (including the core matrix sandwiched by aluminum claddings), gas gaps on both sides of the BORAL plate, and a gas gap between the stainless steel cladding for the BORAL plate and the support disk or heat transfer disk.

Additionally, the effective conductivity of the fuel tube without the BORAL plate, which is used in the three-dimensional canister model, is determined by another two-dimensional fuel tube model. As shown in Figure 3.4-9, this model includes the fuel channel, gas gaps between the fuel channel and stainless steel fuel tube, the fuel tube, and a gas gap between the stainless steel cladding and the support disk or heat transfer disk.

ANSYS PLANE55 conduction elements and LINK31 radiation elements are used to construct the models. The model with the BORAL plate consists of 10 layers of conduction elements and 8 radiation elements that are defined at the gas gaps (two per gap). The model without the BORAL plate consists of four layers of conduction elements and four radiation elements that are defined at the gas gaps (two per gap). The thickness of the models (x-direction) is the distance measured from the inside dimension of the fuel channel to the inside dimension of the slot in the support disk (assuming that the fuel tube is located at the center of the disk slot). In the model containing the BORAL plate, the tolerance of the BORAL plate core thickness, 0.0045 inch, is used as the gap size for both sides of the BORAL plate. The height of the models is defined to be the same dimension as the thickness of the models.

In each analysis, a heat flux is applied at the left side of the model and the temperature at the right boundary of the model is constrained. The heat flux is determined on the basis of the design heat load of 16 kW with a peaking factor of 1.22. The maximum temperature of the model (at the left boundary) and the temperature difference (ΔT) across the model are calculated by using ANSYS. The effective conductivity is determined by using the following formula:

$$q = K_{eff}(A/L)\,\Delta T$$

or

$$K_{eff} = qL/(A\,\Delta T)$$

where:

q = heat rate applied to inner surface of fuel tube (Btu/hr)
A = area (in^2)

L = hickness of composite tube model (in)

ΔT = temperature difference across the model (°F)

K_{eff} = effective conductivity (Btu/hr-in-°F).

The temperature-dependent conductivity (K_{eff}) in each analysis is determined by varying the temperature constraint at the boundary of the model and then re-solving for the temperature difference. The effective conductivity for the parallel path is calculated on the basis of area ratio of material.

3.4.1.3 Cask Impact Limiter Thermal Model

As described in Sections 3.4.1.1 and 3.4.1.2, the cask impact limiters are not explicitly modeled in the 3D cask models. In these models, the cask ends enclosed by the impact limiters are modeled as being adiabatic surfaces. The cask impact limiters are evaluated thermally for normal operating conditions in this section. Specifically, the volumetric average temperature of the redwood material in the cask impact limiters is calculated using an ANSYS finite element model. Taking advantage of the symmetrical geometry of the cask impact limiters about the major axis of the cask, the finite element model is an axisymmetric representation of one of the impact limiters with the cask oriented in a horizontal position. This represents the orientation of the impact limiters during normal transport. The cask impact limiter thermal model is shown in Figure 3.4-10.

The finite element model of the cask impact limiter is constructed of PLANE55 axisymmetric thermal elements, and radiation and conduction heat transfer across air gaps within the model are accounted for using effective thermal conductivity properties for air using the method described in Section 3.2.2.3. Air gaps are modeled between the cask and impact limiter based upon nominal dimensions. Additionally, a 0.125-in. thick layer of Fiberfrax® Ceramic Fiber Paper is modeled between the impact limiter redwood and the cask mating surface of the impact limiter. A heat flux of 0.13 Btu/h-in², which represents the package contents, is applied to the interior surface of the cask lid. This heat flux is obtained from the thermal results for the 3D cask model with the PWR canister and air as the canister cover gas (described in Section 3.4.1.1) by

conservatively assuming the heat transfer rate to the cask lid is equal to the heat transfer rate to the canister shield lid.

Heat fluxes representing the normal conditions solar heat loads are applied to the cylindrical and vertical flat end surfaces of the impact limiter as shown in Figure 3.4-10. The solar heat flux applied to the vertical flat surfaces of the impact limiter 0.0769 Btu/h-in² model (which is in the normal transport orientation) are calculated in the same manner described in Section 3.4.1.1.1 using the prescribed solar heat flux value of 737 Btu/12-hr-ft². A solar heat flux of 0.154 Btu/hr-in² is applied to the cylindrical portions of the cask and impact limiter modeled.

A steady-state heat transfer analysis is performed using the ANSYS model described in this Section. The volumetric average temperature of the cask impact limiter redwood material (T_{avg}) is calculated from the results of the thermal steady state analysis.

3.4.1.4 Personnel Barrier Thermal Model

According to 10 CFR 71.43(g), a package must be designed, constructed, and prepared for transport such that in still air at 100°F and shade, no accessible surface of the package has a temperature exceeding 185°F in an exclusive use shipment. Compliance with 10 CFR 71.43(g) is demonstrated by performing a computational fluid dynamics (CFD) analysis on a finite element model of the air between the cask surface (i.e., neutron shield shell) and the personnel barrier using ANSYS/FLOTRAN. The finite element model is constructed of two-dimensional FLUID141 elements and is presented in Figure 3.4-11.

Because of geometrical symmetry, only one-half of the cask and the air around the cask is modeled. In addition to the natural convection of the air, thermal radiation heat transfer from the cask outer surface to the personnel barrier is considered in this model. It is conservative to only model the air between the cask surface and the personnel barrier because it results in a higher air velocity and more heat is carried to the top of the personnel barrier. Along the centerline of the model, the horizontal velocity component is specified to be zero. The nodes at the location of the personnel barrier (except the top side) are conservatively defined as wall conditions (Velocity = 0) to force all of the heat out from the top of the barrier. At the inlet (bottom side of the model), the pressure is set to atmospheric pressure with the temperature constrained to 100°F. The

portion of the model corresponding to the cask surface constrains both the horizontal and vertical components of the velocity to be zero.

The cask and personnel barrier are not explicitly modeled in this analysis—only the air surrounding the cask is modeled. It is conservative that the personnel barrier is not explicitly modeled because it will not have a temperature greater than the temperature of the air in contact with it. The temperatures of nodes in the model that correspond to the air adjacent to the cask surface are constrained as boundary conditions of the model. The temperature is considered to be linearly distributed, with the bottom and top temperatures equal to 267°F and 244°F, respectively.

Since the personnel barrier is not explicitly modeled, its temperature is considered to be the temperature of the air at coordinates that correspond the location of the personnel barrier surface. The maximum temperature of the personnel barrier occurs at the top most location at the centerline of the model. The temperatures at key points from the analysis using the model described above are shown in Figure 3.4-12.

3.4.1.5 Test Model

The methods previously described have been used in previous transport cask licensing and are sufficient to show that the Universal Transport Cask meets the criteria set forth in Section 3.4. Therefore, no thermal test model is created.

3.4.2 Maximum Temperatures

Using the thermal models described in Sections 3.4.1.1 and 3.4.1.2, temperatures for the PWR and BWR cask body, canister, basket, and fuel rod cladding are determined for three normal conditions of transport: (1) maximum decay heat, 100°F ambient temperature, and solar insolance; (2) maximum decay heat, -40°F ambient temperature, and no insolance; and (3) no decay heat, -40°F ambient temperature, and no insolance. The maximum temperatures of the principal PWR and BWR cask components, canister, basket components, and fuel rod cladding are shown in Tables 3.4-1 and 3.4-2 for the first two environmental conditions listed above. For the third environmental condition (i.e., no decay heat, -40°F ambient temperature, and no insolance), no analysis is necessary because all package temperatures will equilibrate to -40°F. The cask body maximum allowable component temperatures are shown in Section 3.3.2 and Table 3.4-3.

Using the thermal model described in Section 3.4.1.3, the volumetric average temperature of the redwood in the impact limiters is 135°F.

3.4.3 Minimum Temperatures

The minimum temperatures of the cask and components occur with no heat load and -40°F. These conditions yield a uniform -40°F temperature throughout the Universal Transport Cask package. All package components are capable.

3.4.4 Maximum Internal Pressures

In the following sections, the maximum internal operating pressures for normal conditions of transport are calculated for the PWR and BWR Transportable Storage Canisters and for the Universal Transport Cask cavity. The maximum internal operating pressures for the canisters and cask cavity are summarized in Table 3.4-4.

3.4.4.1 Maximum Internal Pressure for PWR Fuel Canister and Cask

The internal pressures within the cask and PWR fuel canister are a function of rod-fill, fission, and backfill gases. The design basis PWR fuel assembly for the internal pressure calculations is the Westinghouse 17x17 fuel assembly. This assembly has the highest fuel rod backfill pressure (500 psig) and burnup (45,000 MWD/MTU). However, a burnup of 50,000 MWD/MTU is conservatively used for this evaluation. The PWR internal pressure calculations are performed on a PWR Class 1 System. This is conservative because the high quantity of fission gas is coupled with the smallest free gas volume, which represents the bounding analysis with respect to internal pressure. There are three different gases that contribute to the PWR fuel canister internal pressure and four gases that contribute to the cask cavity internal pressure. The canister gases are the fuel rod backfill gas, fission gas, and canister backfill gas. Since the canister is not the containment boundary of the shipping container, the cask cavity gases are the same gases as those listed for the canister plus the cask cavity backfill gas. All of the gases except the fission gas are assumed to be helium which is the design basis backfill gas. The internal pressure within the cask cavity is conservatively calculated by assuming the backfill gas within the cask is at the same temperature as the gas in the PWR fuel canister.

The maximum calculated average temperature of the gases within the PWR fuel canister with air as the backfill gas in the canister is 492°F. The average gas temperature with air as the cover gas within the canister is greater than the average gas temperature with helium as the cover gas within the canister. This average temperature of the PWR fuel canister cover gas is determined using the three-dimensional cask model described in Section 3.4.1.1. The backfill gases are assumed to be at an initial temperature of 68°F and an initial pressure of 1 atm. The total pressure for each volume is found by explicitly calculating the molar quantity of each gas and summing the quantities directly. The quantity of fission gas is derived using $0.3125 \frac{\text{Atoms of Gas}}{\text{Fission}}$

Additionally, the internal pressures within the PWR fuel canister and cask cavity are calculated assuming both 3% and 100% of the fuel rods fail. In both cases, it is assumed that the failed fuel rods release 30% of their total fission gas and all of the rod fill gas.

3.4.4.1.1 Maximum Internal Pressure for PWR Fuel Canister

The maximum internal pressure in the PWR fuel canister is calculated in the following two sections for 3% fuel rod failure and 100% fuel rod failure, respectively. The internal pressure is calculated for each condition using the ideal gas law (i.e., PV=NRT). The total number of moles of gas (N) used in the ideal gas law is the sum of the number of moles of canister backfill gas, fuel rod backfill gas, and fission gas. The calculation of the number of moles of: (1) fuel rod backfill gas; (2) fission gas; and (3) canister backfill gas are calculated in the following equations.

(1) Moles of fuel rod backfill gas:

Conservatively, the plenum volume is calculated neglecting the plenum spring.

$$V_1 = \pi r^2 L$$

$$V_1 = \pi \times \left\{ \left(\left(\frac{0.374\text{ inches}}{2} \right) - 0.0225\text{ inches} \right)^2 \times 6.3\text{ inches} \right\} = 0.5356\text{ inches}^3$$

The pellet clad gap volume is calculated as:

$$V_2 = \pi L \left(r^2_{\text{Clad ID}} - r^2_{\text{Pellet OD}} \right)$$

$$V_2 = \pi \times (144\text{ inches}) \times \left(\left(\frac{(0.374\text{ inches})}{2} - 0.0225\text{ inches} \right)^2 - \frac{(0.3225\text{ inches})^2}{4} \right) = 0.4789\text{ inches}^3$$

The fuel rod backfill volume is calculated as:

$$V_{\text{Rod Backfill}} = V_1 + V_2$$

$$V_{\text{Rod Backfill}} = 0.5356 \text{ inches}^3 + 0.4789 \text{ inches}^3 = 1.0145 \text{ inches}^3$$

$$1.0145 \text{ inches}^3 \times 264 \frac{\text{rods}}{\text{assembly}} \times 24 \frac{\text{assemblies}}{\text{Canister}} \times \left(2.54 \frac{\text{cm}}{\text{inch}}\right)^3 \times \frac{0.001\,\ell}{\text{cm}^3}$$

= 105.33 liters/Canister

Using the ideal gas law, the quantity of fuel rod backfill gas is calculated as:

$$N = \frac{\left\{(500\text{ psig} + 14.7) \times \frac{1\text{ atm}}{14.7\text{ psia}}\right\} \times 105.33 \frac{\ell}{\text{Canister}}}{0.0821 \frac{\text{atm}\,\ell}{\text{Mole K}} \times 293\text{ K}} = 153.31 \frac{\text{Total Moles of Rod Fill Gas}}{\text{Canister}}$$

(2) Moles of Fission Gas:

$$N = 50{,}000 \frac{\text{MWd}}{\text{MTU}} \times 1.0 \times 10^6 \frac{\text{W}}{\text{MW}} \times 86{,}400 \frac{\text{sec}}{\text{d}} \times \frac{1\text{ MeV}}{1.602 \times 10^{-13}\text{ J}} \times \frac{1\text{ Fission}}{200\text{ MeV}}$$
$$\times 0.3125 \frac{\text{Atoms of Gas}}{\text{Fission}} \times \frac{1\text{ Mole}}{6.02 \times 10^{23}\text{ Atoms}} \times 0.4807 \frac{\text{MTU}}{\text{Assembly}} \times 24 \frac{\text{Assemblies}}{\text{Canister}}$$

$$N = 808 \frac{\text{Total Moles of Fission Gas}}{\text{Canister}}$$

(3) Moles of Canister Backfill Gas:

The canister free gas volume is calculated using the formula shown below.

$$V^{TSC}_{\text{Free Gas Volume}} = V_{\text{Canister}} - \left(V_{\text{Shield Lid}} + V_{\text{Structural Lid}} + V_{\text{Lid support ring}} + V_{\text{Basket}} + V_{\text{Fuel}}\right)$$

$$V_{\text{Canister}} = \pi\frac{d^2}{4}\left(L_{\text{Canister}} - L_{\text{Bottom Plate}}\right) = \pi \times \frac{(65.81\,\text{inches})^2}{4} \times (175.05\,\text{inches} - 1.75\,\text{inch})$$

$$V_{\text{Canister}} = 589{,}484.3\,\text{inches}^3$$

$$V_{\text{Basket}} = V_{\text{BORAL}} + \frac{M_{\text{Aluminum}}}{\rho_{\text{Aluminum}}} + \frac{M_{\text{Stainless Steel}}}{\rho_{\text{Stainless Steel}}}$$

$$V_{\text{Basket}} = 8{,}826\,\text{inches}^3 + \frac{1{,}760\,\text{lb}}{0.0980\,\frac{\text{lb}}{\text{in}^3}} + \frac{12{,}405\,\text{lb}}{0.2910\,\frac{\text{lb}}{\text{in}^3}} = 69{,}414\,\text{inches}^3$$

$$V_{\text{Fuel}} = V_{\text{Assembly}} + V_{\text{Poison Rods}} + V_{\text{Poison Rod Spider}}$$

$$V_{\text{Assembly}} = \pi\frac{(\text{Rod}_{\text{OD}})^2}{4} * L_{\text{rod}} * N_{\text{rods}} + V_{\text{assembly hardware}}$$

$$V_{\text{Assembly}} = \pi\frac{(0.374)^2}{4} * 151.635 * 264 + 240$$

$$V_{\text{Assembly}} = 4637.81\,\text{in}^3 \text{ per assembly}$$

$$V_{\text{PoisonRods}} = 156.1\text{ inch x } \frac{(0.385\text{ inch})^2}{4} \text{ x } \pi \text{ x } 24\,\frac{\text{Guide Tubes}}{\text{Assembly}}$$

$$= 436.14\,\frac{\text{inch}^3}{\text{Poison Rods}}$$

$$V_{\text{Poison Rod Spider}} = \left[\left(5.776\ \frac{\text{lb}_{\text{SS304}}}{\text{spider}} \middle/ 0.2910\ \frac{\text{lb}_{\text{SS304}}}{\text{inch}^3}\right) + \left(0.926\ \frac{\text{lb}_{\text{Inconel}}}{\text{spider}} \middle/ 0.2992\ \frac{\text{lb}_{\text{Inconel}}}{\text{inch}^3}\right)\right]$$

$$= 22.94\ \frac{\text{inch}^3}{\text{Spider}}$$

$$V_{\text{Fuel}} = \left(4{,}637.81\frac{\text{inch}^3}{\text{Assembly}} + 436.14\frac{\text{inch}^3}{\text{Poison Rods}} + 22.94\frac{\text{inch}^3}{\text{Spider}}\right) \times 24\frac{\text{Assemblies}}{\text{Canister}}$$

$$= 122{,}325.36\,\text{inch}^3$$

$$V^{\text{TSC}}_{\text{Free Gas Volume}} = V_{\text{Canister}} - \left(V_{\text{Shield Lid}} + V_{\text{Structural Lid}} + V_{\text{Lid support ring}} + V_{\text{Basket}} + V_{\text{Fuel}}\right)$$

$$V^{\text{TSC}}_{\text{Free Gas Volume}} = 589{,}484.3 - \left(\begin{array}{l} 23{,}452\,\text{inches}^3 + 10{,}060\,\text{inches}^3 \\ +69{,}414\,\text{inches}^3 + 51\,\text{inches}^3 + 122{,}325.36\,\text{inch}^3 \end{array}\right)$$

$$V^{\text{TSC}}_{\text{Free Gas Volume}} = 364{,}181.94\frac{\text{inches}^3}{\text{Canister}}$$

$$V^{\text{TSC}}_{\text{Free Gas Volume}} = 364{,}181.94\frac{\text{inches}^3}{\text{Canister}} \times \frac{1\,\ell}{61.02\,\text{inches}^3} = 5968\frac{\ell}{\text{Canister}}$$

$$V^{\text{TSC}}_{\text{Free Gas Volume}} = 5968\frac{\ell}{\text{Canister}}$$

Using the ideal gas law, the quantity of canister backfill gas is calculated as:

$$N = \frac{1\,\text{atm} \times 5{,}968\,\dfrac{\ell}{\text{Canister}}}{0.0821\dfrac{\text{atm}\,\ell}{\text{Mole K}} \times 390.92\,\text{K}} = 185.96\frac{\text{Moles of Canister Backfill Gas}}{\text{Canister}}$$

3.4.4.1.1.1 Maximum Internal Pressure Calculations for PWR Fuel Canister (3% Fuel Rod Failure)

Using the number of moles calculated for the fuel rod backfill gas, the fission gas, and the canister backfill gas in Section 3.4.4.1.1, the ideal gas law, and assuming the failed fuel rods release 30% of their total fission gas and all of the rod backfill gas, the PWR fuel canister internal pressure for 3% failed fuel rods is calculated.

The total quantity of gas in the canister for 3% failed fuel rods:

$$N = N_{\text{TSC Backfill}} + 0.03\left(N_{\text{Rod Backfill}}\right) + 0.3(0.03)\left(N_{\text{Fission Gas}}\right)$$

$$N = 185.96\,\frac{\text{Moles}}{\text{Canister}} + 0.03\left(153.31\,\frac{\text{Moles}}{\text{Canister}}\right) + 0.3(0.03)\left(808\,\frac{\text{Moles}}{\text{Canister}}\right)$$

$$N = 198\,\frac{\text{Moles}}{\text{Canister}}$$

Using the ideal gas law, the internal pressure in the PWR canister with 3% failed fuel rods is:

$$P = \frac{\left(198\,\frac{\text{Moles}}{\text{Canister}}\right)\times\left(0.0821\,\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 528.71\,\text{K}}{\left(5{,}968\,\frac{\ell}{\text{Canister}}\right)} = 1.44\,\text{atm} \approx 21.2\,\text{psia} \approx 6.5\,\text{psig}$$

3.4.4.1.1.2 Maximum Internal Pressure Calculations for PWR Fuel Canister (100% Fuel Rod Failure)

Using the number of moles calculated for the fuel rod backfill gas, the fission gas, and the canister backfill gas in Section 3.4.4.1.1, the ideal gas law, and assuming the failed fuel rods release 30% of their total fission gas and all of the rod backfill gas, the PWR fuel canister internal pressure for 100% failed fuel rods is calculated. The number of moles of gas in the canister is used in the calculation of the accident condition pressure presented in Section 3.5.4.1.1.

The total quantity of gas in the canister for 100% failed fuel rods:

$$N = N_{\text{TSC Backfill}} + N_{\text{Rod Backfill}} + 0.3\left(N_{\text{Fission Gas}}\right)$$

$$N = 185.96\,\frac{\text{Moles}}{\text{Canister}} + 153.31\,\frac{\text{Moles}}{\text{Canister}} + 0.3\left(808\,\frac{\text{Moles}}{\text{Canister}}\right)$$

$$N = 582\,\frac{\text{Moles}}{\text{Canister}}$$

Using the ideal gas law, the internal pressure in the PWR canister with 100% failed fuel rods is:

$$P = \frac{\left(582\,\frac{\text{Moles}}{\text{Canister}}\right)\times\left(0.0821\,\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 528.71\,\text{K}}{\left(5{,}968\,\frac{\ell}{\text{Canister}}\right)} = 4.2\ \text{atm} \approx 62.2\ \text{psia} \approx 47.5\ \text{psig}$$

3.4.4.1.2 Maximum Internal Pressure for Cask with PWR Fuel Canister

The maximum internal pressure in the cask with PWR fuel canister is calculated in the following two sections for 3% fuel rod failure and 100% fuel rod failure respectively. The internal pressure is calculated for each condition using the ideal gas law (i.e., PV=NRT). The total number of moles of gas (N) used in the ideal gas law is the molar quantity of gas in the cask cavity combined with the molar quantity of gas in the PWR fuel canister calculated in Section 3.4.4.1.1. The calculation of the molar quantity of gas in the cask cavity is calculated in the following equations.

The cask cavity free gas volume is:

$$V^{UTC}_{\text{Free Gas Volume}} = V^{UTC}_{\text{Cavity}} - \left(V_{TSC} + V_{\text{Spacers}}\right)$$

The canister spacer volume in the above equation is calculated.

$$V_{\substack{\text{Cover}\\\text{Plate}}} = \frac{\pi\, D^2 L}{4} = \frac{\pi \times (67\ \text{inches})^2 \times \left(\frac{3}{8}\ \text{inches}\right)}{4} = 1322.12\ \text{inches}^3$$

Calculating the volume of the bottom plates starting with the smallest diameter first.

$$V = \pi \times D \times W \times THK$$

$$V_{Plate\ 1} = \pi \times 12 \text{ inches} \times \left(16.75 \text{ inches} - \frac{3}{8} \text{ inches}\right) \times \frac{3}{8} \text{ inches} = 231.50 \text{ inches}^3$$

$$V_{Plate\ 2} = \pi \times 24 \text{ inches} \times \left(16.75 \text{ inches} - \frac{3}{8} \text{ inches}\right) \times \frac{3}{8} \text{ inches} = 463.00 \text{ inches}^3$$

$$V_{Plate\ 3} = \pi \times 32 \text{ inches} \times \left(16.75 \text{ inches} - \frac{3}{8} \text{ inches}\right) \times \frac{3}{8} \text{ inches} = 617.33 \text{ inches}^3$$

$$V_{Plate\ 4} = \pi \times 50 \text{ inches} \times \left(16.75 \text{ inches} - \frac{3}{8} \text{ inches}\right) \times \frac{3}{8} \text{ inches} = 964.58 \text{ inches}^3$$

$$V_{Plate\ 5} = \pi \times 56 \text{ inches} \times \left(16.75 \text{ inches} - \frac{3}{8} \text{ inches}\right) \times \frac{3}{8} \text{ inches} = 1{,}080.33 \text{ inches}^3$$

$$V_{Plate\ 6} = \pi \times 65 \text{ inches} \times \left(16.75 \text{ inches} - \frac{3}{8} \text{ inches}\right) \times \frac{3}{8} \text{ inches} = 1{,}258.96 \text{ inches}^3$$

$$V_{Spacer} = V_{Cover\ Plate} + V_{Plate\ 1} + V_{Plate\ 2} + V_{Plate\ 3} + V_{Plate\ 4} + V_{Plate\ 5} + V_{Plate\ 6}$$

$$V_{Spacer} = \begin{pmatrix} 1{,}322.12 + 231.50 + 463.00 + 617.33 \\ + 964.58 + 1{,}080.33 + 1{,}258.96 \end{pmatrix} \text{inches}^3 = 5{,}937.82 \text{ inches}^3$$

The cask cavity total volume is:

$$V^{UTC}_{Cavity} = \frac{\pi\, D^2_{\substack{Cav.\\ I.D}} L_{Cav.}}{4} = \frac{\pi \times (67.61 \text{ inches})^2 \times (192.5 \text{ inches})}{4} = 691{,}102.54 \text{ inches}^3$$

$$V_{TSC} = \frac{\pi D^2_{\substack{\text{Canister}\\ \text{OD}}} L_{\text{Canister}}}{4} = \frac{\pi \times (67.06\ \text{inches})^2 \times (175.05\ \text{inches})}{4} = 618{,}271.31\ \text{inches}^3$$

$$V^{\text{UTC}}_{\text{Free Gas Volume}} = 691{,}102.54\ \text{inches}^3 - \left(618{,}271.3\ \text{inches}^3 + 5{,}937.82\ \text{inches}^3\right) = 66{,}893.41\ \text{inches}^3$$

Therefore, the free volume in the cask cavity is:

$$V_{\text{UTC Free Gas Volume}} = 66{,}893.41 \frac{\text{inches}^3}{\text{cask}} \times \frac{1\,\ell}{61.02\ \text{inches}^3} = 1{,}096.25 \frac{\ell}{\text{cask}}$$

$$V_{\text{UTC Free Gas Volume}} = 1{,}096.25 \frac{\ell}{\text{cask}}$$

Using the ideal gas law, the molar quantity of gas in the cask is:

$$N = \frac{1\ \text{atm} \times 1{,}096.25 \dfrac{\ell}{\text{cask}}}{0.0821 \dfrac{\text{atm} - \ell}{\text{Mole} - \text{K}} \times 293\ \text{K}} = 45.57 \frac{\text{Moles of Cask Backfill Gas}}{\text{Cask}}$$

3.4.4.1.2.1 Maximum Internal Pressure Calculations for Cask with PWR Fuel Canister (3% Fuel Rod Failure)

Since the PWR fuel canister is not considered to be the containment boundary for the shipping container, the molar quantities of the gases within the PWR fuel container are assumed to contribute to the maximum internal pressure in the cask cavity. Therefore, using the molar quantities calculated for the fuel rod backfill gas, the fission gas, and the canister backfill gas in Section 3.4.4.1.1 plus the molar quantity of gas in the cask, the maximum internal pressure within the cask with PWR fuel canister for 3% failed fuel rods is calculated using the ideal gas law. Additionally, it is assumed that the failed fuel rods release 30% of their total fission gas and all of the rod backfill gas.

The total quantity of gas in the cask with PWR canister for 3% failed fuel rods is:

$$N = N_{\text{TSC Backfill}} + N_{\text{UTC Backfilll}} + 0.03\left(N_{\text{Rod Backfill}}\right) + 0.3(0.03)\left(N_{\text{Fission Gas}}\right)$$

$$N = 185.96\frac{\text{Moles}}{\text{Cask}} + 45.57\frac{\text{Moles}}{\text{Cask}} + 0.03\left(153.31\frac{\text{Moles}}{\text{Cask}}\right) + 0.3(0.03)\left(808\frac{\text{Moles}}{\text{Cask}}\right)$$

$$N = 243.4\frac{\text{Moles}}{\text{Cask}}$$

Thus, using the ideal gas law, the internal pressure in the cask with PWR fuel canister for 3% failed fuel rods is:

$$P = \frac{\left(243.40\frac{\text{Moles}}{\text{Cask}}\right)\times\left(0.0821\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 528.71\text{ K}}{\left(7{,}064.25\frac{\ell}{\text{Cask}}\right)} = 1.50\text{ atm} \approx 21.98\text{ psia} \approx 7.3\text{ psig}$$

3.4.4.1.2.2 Maximum Internal Pressure Calculations for Cask with PWR Fuel Canister (100% Fuel Rod Failure)

Since the PWR fuel canister is not considered to be the containment boundary for the shipping container, the molar quantities of the gases within the PWR fuel container are assumed to contribute to the maximum internal pressure in the cask cavity. Therefore, using the molar quantities calculated for the fuel rod backfill gas, the fission gas, and the canister backfill gas in Section 3.4.4.1.1 plus the molar quantity of gas in the cask, the maximum internal pressure within the cask with PWR fuel canister for 100% failed fuel rods is calculated using the ideal gas law. Additionally, it is assumed that the failed fuel rods release 30% of their total fission gas and all of the rod backfill gas. The number of moles of gas is used in the accident conditions evaluation presented in Section 3.5.4.1.2.

The total quantity of gas in the cask with PWR canister for 100% failed fuel rods is:

$$N = N_{\text{TSC Backfill}} + N_{\text{UTC Backfill}} + N_{\text{Rod Backfill}} + 0.3\left(N_{\text{Fission Gas}}\right)$$

$$N = 185.96\frac{\text{Moles}}{\text{Cask}} + 45.57\frac{\text{Moles}}{\text{Cask}} + 153.31\frac{\text{Moles}}{\text{Cask}} + 0.3\left(808\frac{\text{Moles}}{\text{Cask}}\right)$$

$$N = 627.24 \frac{\text{Moles}}{\text{Cask}}$$

Thus, using the ideal gas law, the internal pressure in the cask with PWR fuel canister for 100% failed fuel rods is:

$$P = \frac{\left(627.24 \frac{\text{Moles}}{\text{Cask}}\right) \times \left(0.0821 \frac{\text{atm}\,\ell}{\text{mole K}}\right) \times 528.71\,\text{K}}{\left(7{,}064.25 \frac{\ell}{\text{Cask}}\right)} = 3.854\,\text{atm} \approx 56.66\,\text{psia} \approx 41.96\,\text{psig}$$

3.4.4.2 Maximum Internal Pressure for BWR Fuel Canister and Cask

The internal pressures within the cask and BWR fuel canister are a function of rod-fill, fission, and backfill gases. The design basis BWR fuel assembly for the internal pressure calculations is the Exxon-ANF 9x9 fuel assembly. This assembly has the highest fuel rod backfill pressure (60 psig) and burnup (40,000 MWD/MTU). However, a burnup of 50,000 MWD/MTU is conservatively used for this evaluation. It should be noted that the design basis BWR fuel assembly for the internal pressure calculations represents an impossible configuration since the Exxon-ANF 9x9 fuel assembly will not fit in the Class 4 UMS® TSC; however, this represents the case which would maximize the internal pressures. There are three different gases that contribute to the BWR fuel canister internal pressure and four gases that contribute to the cask cavity internal pressure. The canister gases are the fuel rod backfill gas, fission gas, and canister backfill gas. Since the canister is not the containment boundary of the shipping container, he cask cavity gases are the same gases as those listed for the canister plus the cask cavity backfill gas. All of the gases except the fission gas are assumed to be helium which is the design basis backfill gas. The internal pressure within the cask cavity is conservatively calculated by assuming the backfill gas within the cask is at the same temperature as the gas in the BWR fuel canister.

The maximum calculated average temperature of the gases within the BWR fuel canister with air as the backfill gas in the canister is 432°F. The average gas temperature with air as the cover gas within the canister is greater than the average gas temperature with helium as the cover gas within the canister. This average temperature of the BWR fuel canister cover gas is determined using the three-dimensional cask model described in Section 3.4.2.1. The backfill gases are assumed to be at an initial temperature of 68°F and an initial pressure of 1 atm. The total pressure for each volume is found by calculating the molar quantity of each gas and summing the quantities directly. The quantity of fission gas is derived using $0.3125 \dfrac{\text{Atoms of Gas}}{\text{Fission}}$.

Additionally, the internal pressures within the BWR fuel canister and cask cavity are calculated assuming both 3% and 100% of the fuel rods fail. In both cases, it is assumed that the failed fuel rods release 30% of their total fission gas and all of the rod fill gas.

3.4.4.2.1 Maximum Internal Pressure for BWR Fuel Canister

The maximum internal pressure in the BWR fuel canister is calculated in the following two sections for 3% fuel rod failure and 100% fuel rod failure, respectively. The internal pressure is calculated for each condition using the ideal gas law (i.e., PV=NRT). The total number of moles of gas (N) used in the ideal gas law is the sum of the number of moles of canister backfill gas, fuel rod backfill gas, and fission gas. The calculation of the number of moles of: (1) fuel rod backfill gas; (2) fission gas; and (3) canister backfill gas are calculated in the following equations.

(1) Moles of fuel rod backfill gas:

Conservatively, the plenum volume is calculated neglecting the plenum spring.

$$V_1 = \pi r^2 L$$

$$V_1 = \pi \times \left\{\left(\left(\frac{0.424 \text{ inches}}{2}\right) - 0.03 \text{ inches}\right)^2 \times 9.578 \text{ inches}\right\} = 0.9967 \text{ inches}^3$$

The pellet clad gap is calculated as:

$$V_2 = \pi L\left(r^2_{\text{Clad ID}} - r^2_{\text{Pellet OD}}\right)$$

$$V_2 = \pi \times (150 \text{ inches}) \times \left(\left(\frac{(0.424 \text{ inches})}{2} - 0.03 \text{ inches}\right)^2 - \frac{(0.3565 \text{ inches})^2}{4}\right) = 0.6366 \text{ inches}^3$$

The fuel rod backfill volume is calculated as:

$$V_{\text{Rod Backfill}} = V_1 + V_2$$

$$V_{\text{Rod Backfill}} = 0.9967\ \text{inches}^3 + 0.6366\ \text{inches}^3 = 1.6333\ \text{inches}^3$$

$$1.6333\,\text{inches}^3 \times 79\,\frac{\text{rods}}{\text{assembly}} \times 56\,\frac{\text{assemblies}}{\text{Canister}} \times \left(2.54\,\frac{\text{cm}}{\text{inch}}\right)^3 \times \frac{0.001\,\ell}{\text{cm}^3} = 118.41\,\frac{\ell}{\text{Canister}}$$

Using the ideal gas law, the quantity of fuel rod backfill gas is calculated as:

$$N = \frac{\left\{(60\ \text{psig} + 14.7) \times \dfrac{1\ \text{atm}}{14.7\ \text{psia}}\right\} \times 118.41\,\dfrac{\ell}{\text{Canister}}}{0.0821\,\dfrac{\text{atm}\ \ell}{\text{Mole K}} \times 293\ \text{K}} = 25.01\,\frac{\text{Total Moles of Rod Fill Gas}}{\text{Canister}}$$

(2) Moles of Fission Gas:

$$N = 50{,}000\,\frac{\text{MWd}}{\text{MTU}} \times 1.0 \times 10^6\,\frac{\text{W}}{\text{MW}} \times 86{,}400\,\frac{\text{sec}}{\text{d}} \times \frac{1\ \text{MeV}}{1.602 \times 10^{-13}\ \text{J}} \times \frac{1\ \text{Fission}}{200\ \text{MeV}}$$

$$\times 0.3125\,\frac{\text{Atoms of Gas}}{\text{Fission}} \times \frac{1\ \text{Mole}}{6.02 \times 10^{23}\ \text{Atoms}} \times 0.1979\,\frac{\text{MTU}}{\text{Assembly}} \times 56\,\frac{\text{Assemblies}}{\text{Canister}}$$

$$N = 775.67\,\frac{\text{Total Moles of Fission Gas}}{\text{Canister}}$$

(3) Moles of Canister Backfill Gas:

The canister free gas volume is calculated using the formula shown below.

$$V^{\text{TSC}}_{\text{Free Gas Volume}} = V_{\text{Canister}} - \left(V_{\text{Shield Lid}} + V_{\text{Structural Lid}} + V_{\text{Basket}} + V_{\text{Fuel}}\right)$$

$$V_{\text{Canister}} = \pi\frac{d^2}{4}\left(L_{\text{Canister}} - L_{\text{Bottom Plate}}\right) = \pi \times \frac{(65.81\ \text{inches})^2}{4} \times (185.55\ \text{inches} - 1.75\ \text{inch})$$

$$V_{\text{Canister}} = 625{,}199.8\ \text{inches}^3$$

$$V_{Basket} = V_{BORAL} + \frac{M_{Aluminum}}{\rho_{Aluminum}} + \frac{M_{Stainless\ Steel}}{\rho_{Stainless\ Steel}} + \frac{M_{Carbon\ Steel}}{\rho_{Carbon\ Steel}}$$

$$V_{Basket} = 12{,}085\ \text{inches}^3 + \frac{1{,}472\ \text{lb}}{0.0980\ \frac{\text{lb}}{\text{in}^3}} + \frac{7{,}187\ \text{lb}}{0.2910\ \frac{\text{lb}}{\text{in}^3}} + \frac{8{,}102\ \text{lb}}{0.2837\ \frac{\text{lb}}{\text{in}^3}} = 80{,}361.34\ \text{inches}^3$$

$$V^{TSC}_{Free\ Gas\ Volume} = V_{Canister} - \left(V_{Shield\ Lid} + V_{Structural\ Lid} + V_{Lid\ Support\ ring} + V_{Basket} + V_{Fuel}\right)$$

$$V^{TSC}_{Free\ Gas\ Volume} = 625{,}199.8 - \left(\begin{array}{l} 23{,}452\ \text{inches}^3 + 10{,}060\ \text{inches}^3 + 51\ \text{inches}^3 \\ + 80{,}361.34\ \text{inches}^3 + \left(1{,}884 \frac{\text{inches}^3}{\text{Assemblies}} \times 56 \frac{\text{Assemblies}}{\text{Canister}}\right) \end{array}\right)$$

$$V^{TSC}_{Free\ Gas\ Volume} = 405{,}771.46 \frac{\text{inches}^3}{\text{Canister}}$$

$$V^{TSC}_{Free\ Gas\ Volume} = 405{,}771.46 \frac{\text{inches}^3}{\text{Canister}} \times \frac{1\ \ell}{61.02\ \text{inches}^3} = 6{,}649.81 \frac{\ell}{\text{Canister}}$$

Using the ideal gas law, the quantity of canister backfill gas is calculated as:

$$N = \frac{1\ \text{atm} \times 6{,}649.81 \frac{\ell}{\text{Canister}}}{0.0821 \frac{\text{atm}\ \ell}{\text{Mole K}} \times 385.93\ \text{K}} = 209.87 \frac{\text{Moles of Canister Backfill Gas}}{\text{Canister}}$$

3.4.4.2.1.1 Maximum Internal Pressure Calculations for BWR Fuel Canister (3% Fuel Rod Failure)

Using the number of moles calculated for the fuel rod backfill gas, the fission gas, and the canister backfill gas in Section 3.4.4.2.1, the ideal gas law, and assuming the failed fuel rods release 30% of their total fission gas and all of the rod backfill gas, the BWR fuel canister internal pressure for 3% failed fuel rods is calculated.

The total quantity of gas in the canister for 3% failed fuel rods:

$$N = N_{\text{TSC Back-Fill}} + 0.03\left(N_{\text{Rod Back-Fill}}\right) + 0.3(0.03)\left(N_{\text{Fission Gas}}\right)$$

$$N = 209.87\frac{\text{Moles}}{\text{Canister}} + 0.03\left(25.01\frac{\text{Moles}}{\text{Canister}}\right) + 0.3(0.03)\left(775.67\frac{\text{Moles}}{\text{Canister}}\right)$$

$$N = 217.60\frac{\text{Moles}}{\text{Canister}}$$

Thus, using the ideal gas law, the internal pressure in the BWR fuel canister with 3% failed fuel rods is:

$$P = \frac{\left(217.60\frac{\text{Moles}}{\text{Canister}}\right)\times\left(0.0821\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 495.4\,\text{K}}{\left(6{,}649.81\frac{\ell}{\text{Canister}}\right)} = 1.33\,\text{atm} \approx 19.56\,\text{psia} \approx 4.87\,\text{psig}$$

3.4.4.2.1.2 Maximum Internal Pressure Calculations for BWR Fuel Canister (100% Fuel Rod Failure)

Using the number of moles calculated for the fuel rod backfill gas, the fission gas, and the canister backfill gas in Section 3.4.4.2.1, the ideal gas law, and assuming the failed fuel rods

release 30% of their total fission gas and all of the rod backfill gas, the BWR fuel canister internal pressure for 100% failed fuel rods is calculated. The number of moles of gas is used in the BWR accident condition analysis presented in Section 3.5.4.2.1

The total quantity of gas in the canister for 100% failed fuel rods:

$$N = N_{\text{TSC Back-Fill}} + N_{\text{Rod Back-Fill}} + 0.3\left(N_{\text{Fission Gas}}\right)$$

$$N = 209.87\frac{\text{Moles}}{\text{Canister}} + 25.01\frac{\text{Moles}}{\text{Canister}} + 0.3\left(775.67\frac{\text{Moles}}{\text{Canister}}\right)$$

$$N = 467.58\frac{\text{Moles}}{\text{Canister}}$$

Thus, using the ideal gas law, the internal pressure in the BWR fuel canister with 100% failed fuel rods is:

$$P = \frac{\left(467.58\frac{\text{Moles}}{\text{Canister}}\right)\times\left(0.0821\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 495.4\,\text{K}}{\left(6{,}649.81\frac{\ell}{\text{Canister}}\right)} = 2.86\,\text{atm} \approx 42.04\,\text{psia} \approx 27.34\,\text{psig}$$

3.4.4.2.2 Maximum Internal Pressure for Cask with BWR Fuel Canister

The maximum internal pressure in the cask with BWR fuel canister is calculated in the following two sections for 3% fuel rod failure and 100% fuel rod failure respectively. The internal pressure is calculated for each condition using the ideal gas law (i.e., PV=NRT). The total number of moles of gas (N) used in the ideal gas law is the molar quantity of gas in the cask cavity combined with the molar quantity of gas in the PWR fuel canister calculated in Section 3.4.4.2.1. The calculation of the molar quantity of gas in the cask cavity is calculated in the following equations.

The cask cavity free gas volume is:

$$V_{\substack{\text{Total}\\ \text{Free Gas}}} = V_{\text{UTC Cavity}} - \left(V_{\text{TSC}} + V_{\text{Spacers}}\right)$$

$$V_{\text{UTC Cavity}} = \frac{\pi \times D^2_{\text{UTC ID}} \times L_{\text{UTC Lenght}}}{4} = \frac{\pi \times (67.61 \text{ inches})^2 \times 192.5 \text{ inches}}{4} = 691{,}102.54 \text{ inches}^3$$

$$V_{\text{TSC}} = \frac{\hat{}\ \pi \times \hat{}D\ ^2_{\text{TSC OD}} \times L_{\text{Canister}}}{4} = \frac{\pi \times (67.06 \text{ inches})^2 \times 185.55 \text{ inches}}{4} = 655{,}356.99 \text{ inches}^3$$

$$V_{\text{Spacer}} = \frac{\pi \times D^2_{\text{Spacer}} \times H_{\text{Spacer}}}{4} + \left(\pi \times \left(\left(\frac{D_{\text{Major}}}{2}\right)^2 + \left(\frac{D_{\text{Mnor}}}{2}\right)^2\right) \times \frac{H}{2}\right)$$

$$V_{\text{Spacer}} = \left\{ \begin{array}{l} \dfrac{\pi \times (67.0 \text{ inches})^2 \times \frac{1}{2} \text{ inch}}{4} + \\ \left(\pi \times \left(\left(\dfrac{67.0 \text{ inches}}{2}\right)^2 + \left(\dfrac{65.0 \text{ inches}}{2}\right)^2\right) \times \dfrac{1}{2}\right) \end{array} \right\} \times 4 = 20{,}739.22 \text{ inches}^3$$

$$V_{\substack{\text{Total}\\ \text{Free Gas}}} = 691{,}102.54 \text{ inches}^3 - \left(655{,}356.99 \text{ inches}^3 + 20{,}739.22\right) = 15{,}006.33 \text{ inches}^3$$

$$V_{\substack{\text{Total}\\ \text{Free Gas}}} = 15{,}006.33 \frac{\text{inches}^3}{\text{Cask}} \times \frac{1\,\ell}{61.02 \text{ inches}^3} = 245.92 \frac{\ell}{\text{Cask}}$$

Using the ideal gas law, the molar quantity of gas in the cask is:

$$N = \frac{1\,\text{atm} \times 245.92 \dfrac{\ell}{\text{Cask}}}{0.0821 \dfrac{\text{atm - }\ell}{\text{Cask}} \times 293\,\text{K}} = 10.22 \frac{\text{Moles of Cask Backfill Gas}}{\text{Cask}}$$

3.4.4.2.2.1 Maximum Internal Pressure Calculations for Cask with BWR Fuel Canister (3% Fuel Rod Failure)

Since the BWR fuel canister is not considered to be the containment boundary for the shipping container, the molar quantities of the gases within the BWR fuel container are assumed to contribute to the maximum internal pressure in the cask cavity. Therefore, using the molar quantities calculated for the fuel rod backfill gas, the fission gas, and the canister backfill gas in Section 3.4.4.2.1 plus the molar quantity of gas in the cask, the maximum internal pressure within the cask with BWR fuel canister for 3% failed fuel rods is calculated using the ideal gas law. Additionally, it is assumed that the failed fuel rods release 30% of their total fission gas and all of the rod backfill gas.

The total quantity of gas in the cask with BWR canister for 3% failed fuel rods is:

$$N = N_{\text{TSC Backfill}} + N_{\text{UTC Backfilll}} + 0.03\left(N_{\text{Rod Backfill}}\right) + 0.3(0.03)\left(N_{\text{Fission Gas}}\right)$$

$$N = 209.87 \frac{\text{Moles}}{\text{Cask}} + 10.22 \frac{\text{Moles}}{\text{Cask}} + 0.03\left(25.01 \frac{\text{Moles}}{\text{Cask}}\right) + 0.3(0.03)\left(775.67 \frac{\text{Moles}}{\text{Cask}}\right)$$

$$N = 227.82 \frac{\text{Moles}}{\text{Cask}}$$

Thus, using the ideal gas law, the internal pressure in the cask with BWR fuel canister for 3% failed fuel rods is:

$$P = \frac{\left(227.82\,\frac{\text{Moles}}{\text{Cask}}\right)\times\left(0.0821\,\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 495.28\,\text{K}}{\left(6{,}895.73\,\frac{\ell}{\text{Cask}}\right)} = 1.34\,\text{atm} \approx 19.74\,\text{psia} \approx 5.04\,\text{psig}$$

3.4.4.2.2.2 <u>Maximum Internal Pressure Calculations for Cask with BWR Fuel Canister (100% Fuel Rod Failure)</u>

Since the BWR fuel canister is not considered to be the containment boundary for the shipping container, the molar quantities of the gases within the BWR fuel container are assumed to contribute to the maximum internal pressure in the cask cavity. Therefore, using the molar quantities calculated for the fuel rod backfill gas, the fission gas, and the canister backfill gas in Section 3.4.4.2.1 plus the molar quantity of gas in the cask, the maximum internal pressure within the cask with BWR fuel canister for 100% failed fuel rods is calculated using the ideal gas law. Additionally, it is assumed that the failed fuel rods release 30% of their total fission gas and all of the rod backfill gas. The number of moles of gas in the cask is used in the calculation of the accident condition pressure presented in Section 3.5.4.2.2.

The total quantity of gas in the cask with BWR canister for 100% failed fuel rods is:

$$N = N_{\text{TSC Backfill}} + N_{\text{UTC Backfill}} + N_{\text{Rod Backfill}} + 0.3\left(N_{\text{Fission Gas}}\right)$$

$$N = 209.87\,\frac{\text{Moles}}{\text{Cask}} + 10.22\,\frac{\text{Moles}}{\text{Cask}} + 25.01\,\frac{\text{Moles}}{\text{Cask}} + 0.3\left(775.67\,\frac{\text{Moles}}{\text{Cask}}\right)$$

$$N = 477.80\,\frac{\text{Moles}}{\text{Cask}}$$

Thus, using the ideal gas law, the internal pressure in the cask with BWR fuel canister for 100% failed fuel rods is:

$$P = \frac{\left(477.80\,\frac{\text{Moles}}{\text{Cask}}\right)\times\left(0.0821\,\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 495.28\text{ K}}{\left(6{,}895.73\,\frac{\ell}{\text{Cask}}\right)} = 2.82\text{ atm} \approx 41.44\text{ psia} \approx 26.74\text{ psig}$$

3.4.5 Maximum Thermal Stresses

The ANSYS computer code is used to obtain temperatures for use in the structural analyses of Chapter 2.0. These temperatures are presented in Tables 3.4-1 and 3.4-2. The thermal stress calculations for normal conditions of transport are performed in Sections 2.6.1 and 2.6.2.

3.4.6 Maximum Allowable Cladding Temperature and Canister Heat Load

The maximum allowable cladding temperatures are calculated for PWR and BWR systems based on fuel assembly type, maximum burnup, and minimum initial cool time. Allowable heat loads are determined by relating cladding temperature to canister heat load.

Cladding stresses are calculated for a set of representative PWR and BWR assemblies at 40,000 MWD/MTU and 380°C. The limiting, highest stress assemblies, the Westinghouse 14x14 and GE 8x8 (150-inch fuel region), are then evaluated at various burnups to determine the maximum allowable fuel cladding temperature based on PNL-6364 criteria [28]. Maximum allowable cladding temperatures are generically calculated for PWR and BWR burnups ranging from 35,000 MWD/MTU to 45,000 MWD/MTU. PWR burnups are extended to 50,000 MWD/MTU to envelop the Maine Yankee specific inventory. After applying a 5% design margin to the maximum allowable cladding temperatures, the maximum allowable heat load is calculated as a function of burnup and minimum initial cool time.

3.4.6.1 Maximum Allowable Cladding Temperature

Based on PNL-6364, the cladding temperature limit is expressed as a function of initial dry storage temperature, initial cladding stress at the dry storage temperature, and initial storage time. For this evaluation, the transport temperatures and transport times are applied.

The initial cladding stress is a function of the rod internal pressure, temperature, diameter of the fuel rod, and fuel cladding thickness. The initial cladding stress (σ_{mhoop}) for a particular assembly is calculated as [28]:

$$\sigma_{mhoop} = \frac{(P)(D_{mid})}{2t} \times \alpha \times \frac{T_2}{T_1} \times \frac{69.684}{10{,}000}$$

where:

σ_{mhoop} = dry storage cladding hoop stress, MPa
P = internal gas pressure of the rod, psi
T_1 = temperature at which P was determined, K
t = cladding wall thickness, in.
D_{mid} =cladding midwall diameter, in.
α = a factor, 0.95 for PWR rods or 0.90 for BWR rods
T_2 = allowable storage temperature for σ_{mhoop}, K

To account for cladding oxidation during in-core fuel assembly operation and storage of the fuel in the spent fuel pool, the nominal cladding thickness is reduced by 0.06 mm and 0.125 mm for PWR and BWR fuel rods, respectively [3]. For higher burnup PWR fuels (i.e., rod peak burnup up to 50,000 MWD/MTU), Maine Yankee experience is that the maximum oxide layer thickness on the fuel cladding is 120 microns [30]. The allowable cladding temperature calculations at 50,000 MWD/MTU therefore employ an oxide layer thickness of 0.012 cm.

The pressure in the fuel assembly rods is produced by the combination of fill gas and fission gas.

For a given fuel assembly design, the fill gas quantity is fixed and does not vary with discharge burnup. Based on the initial pressure and temperature of the fill gas, the number of moles of gas are calculated using the ideal gas law:

$$PV = NRT$$

where:

P = Pressure
V = Volume (free volume inside fuel rod)
N = Number of moles of gas
R = Universal gas constant
T = Temperature of the gas

The number of moles of fill gas are added to the fission gas quantity and converted to a cladding internal pressure at storage conditions.

The fission gas quantity pressurizing the fuel cladding is calculated on the basis of the burnup and a fission gas release fraction. While the amount of fission gas produced is a predictable quantity (directly correlated to the number of fissions required to produce the desired burnup), the release fraction of the gas from the pellet into the pellet-cladding void depends on fill gas pressure and reactor operating conditions.

The number of fissions (Z) is related to the burnup by:

$$Z = X\ \text{Burnup}\ \frac{\text{MWd}}{\text{MTU}} \times 1.0 \times 10^{6}\ \frac{\text{W}}{\text{MW}} \times 86{,}400\ \frac{\text{sec}}{\text{d}} \times \frac{1\ \text{MeV}}{1.602 \times 10^{-13}\ \text{J}} \times \frac{1\ \text{Fission}}{200\ \text{MeV}}$$
$$\times \frac{1\ \text{Mole}}{6.02 \times 10^{23}\ \text{Atoms}} \times \text{Mass}\ \frac{\text{MTU}}{\text{Assembly}} \times \frac{\text{Assembly}}{\#\ \text{Rods}}$$

Multiplying the number of fissions by 0.3125 (0.25 x 1.25) atoms/fission then derives the quantity of fission gas produced. Olander's "Fundamental Aspects of Nuclear Reactor Fuel Elements" [11] lists the number of gas atoms from a single fission as 0.25. Based on a detailed SAS2H isotope generated fission gas inventory, this fraction is increased by 25% to account for decay chains not included in Olander (particularly those leading to ^{136}Xe). By employing a conservative fission gas fraction rather than the SAS2H output itself, the allowable cladding temperature calculation is decoupled from source term calculations.

Based on Sandia Report 90-2406, "A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements" [10], gas release fractions from the fuel pellets are assumed to be 12% for PWR fuel rods and 25% for BWR fuel rods. Relying on a gas diffusion model (as applied to pre-pressurized light water reactor fuel rods), the Sandia report indicates a release fraction of approximately 1% for PWR rods and approximately 2% for BWR rods [10]. Experimental release fractions reach as high as 16% for PWR rods and 25 % for BWR rods [10]. The higher release fractions are associated with unpressurized fuel rods or those rods run at uncharacteristically high temperatures and linear heat generation rates. While these rods show higher release rates, they are not expected to produce higher "burned fuel" pressures, since the

partial pressure of the fill gas is not present, thereby allowing a larger number of fission gas molecules to accumulate before reaching limiting cladding pressure. The 12% PWR fission gas release fraction excludes the unpressurized Maine Yankee rod data while including the 43,000 MWD/MTU Calvert Cliff data to approximate the upper bound 45,000 MWD/MTU burnup. An additional analysis is performed comparing the 12% PWR and 25% BWR release fractions to the element specific release fractions in Reg. Guide 1.25 [29]. The 12% PWR release fraction results in gas releases similar to those indicated by the Regulatory Guide, while the BWR 25% release fraction is twice the Regulatory Guide indicated gas release. Note that both the Sandia report and the Regulatory Guide release fractions are for punctured fuel rods where the release of the pressurizing gas allows additional gaseous isotopes to migrate from the fuel matrix. Using the 12% PWR and 25% BWR fuel rod release fractions, therefore, results in a conservative cladding pressurization assumption for the intact rod analysis. For higher burnup PWR fuels (i.e., rod peak burnup up to 50,000 MWD/MTU), Maine Yankee experience is that the maximum gas release rate (fuel pellet to rod plenum in intact fuel rods) is less than 3% [30]. Therefore, the 12% release fraction established for standard PWR fuel burned up to 45,000 MWD/MTU is conservatively applied to the higher burnup PWR fuel.

Fuel rod free volume is calculated based on the fuel characteristics in Tables 3.4-5 and 3.4-6 for PWR and BWR fuel, respectively. Not all assemblies requested for loading are included in the tables, since assemblies with significantly higher free volume or lower fuel mass are bound by the cladding stress evaluations presented. Section 3.4.4 contains a sample free volume calculation of a fuel rod. While the maximum canister pressure calculation conservatively neglected the plenum spring volume, the spring volume is subtracted out of the plenum volume in the cladding maximum stress calculation to increase internal rod pressure.

Substituting the internal gas pressure resulting from the releasable gas inventories produced by 40,000 MWD/MTU burned fuel into the initial cladding stress (σ_{mhoop}) equation at a temperature of 380°C results in the assembly-specific maximum cladding stresses shown in Table 3.4-5 and Table 3.4-6. The Westinghouse 14x14 and GE 8x8 (150-inch fuel region) are the limiting PWR and BWR assembly types at 104 and 65 MPa stress levels, respectively.

The stress levels in the limiting assemblies are then evaluated at burnups ranging from 35,000 MWD/MTU to 50,000 MWD/MTU for PWR fuel and 35,000 MWD/MTU to 45,000 MWD/MTU for BWR fuel at temperatures of 300°C and 400°C for PWR fuels and 300°C and 450°C for BWR fuel. The evaluation results are presented in Table 3.4-7. This data is overlaid

on generic stress versus limiting temperature curves to arrive at cool time and burnup-specific maximum cladding allowable temperatures. The data, shown in Table 3.4-8, from which the generic curves are constructed, is taken from Table 3.1 of PNL-6189 [27].

The cladding temperature limit curves for the limiting PWR and BWR fuel assemblies are provided in Figures 3.4-13 and Figure 3.4-14. The intercept of each of the curves represents the maximum allowable cladding temperature at a given cool time and maximum assembly burnup. Fuel rod peak cladding stress level and the allowable cladding temperature are calculated using the assembly average burnup, even though some rods experience a higher burnup than the average. The average burnup is used, since the quantity of fission gas formation and the fuel rod gas temperature are conservatively determined. As shown in Table 3.4-9, allowable cladding temperature varies only slightly over a wide range of burnup for a given required cooling time. Consequently, the variation in cladding stress with burnup is also small.

3.4.6.2 Maximum Allowable Canister Heat Load

Thermal analysis was performed at three heat loads for PWR fuel and one heat load for BWR fuel to determine the corresponding maximum fuel cladding temperature. Only one heat load is analyzed for BWR fuel because the maximum computed clad temperature is 548°F (286.7°C) at the maximum heat load of 16 kW, which is already lower than the minimum allowable temperature limit for any BWR fuel (Table 3.4-9). Therefore, BWR fuel was not further analyzed and a fixed maximum decay heat of 16 kW is allowable for transport of BWR fuel.

The thermal models and methods, described in Section 3.4.1, used to determine the temperature of fuel cladding and system components for the design basis heat load are applied to determine the cladding temperature at reduced heat loads. The ANSYS calculated temperatures that provide input for correlating allowable cladding temperature to allowable heat load are:

	Fuel Clad Temperature		Heat Load
Fuel Type	(°F)	(°C)	(kW)
PWR	537	280.6	14
PWR	610	321.1	17
PWR	673	356.1	20
BWR	548	286.7	16

The PWR temperature versus heat load curve is plotted in Figure 3.4-15. To provide adequate design margin, the maximum allowable cladding temperatures are reduced by 5% prior to their use in the calculation of maximum allowable canister heat load. Maximum allowable canister heat loads are calculated for initial cool times ranging from 5 to 15 years and burnups ranging from 35,000 MWD/MTU to 50,000 MWD/MTU for PWR fuel and 35,000 MWD/MTU to 45,000 MWD/MTU for BWR fuel. The results of the PWR and BWR analysis are presented in Table 3.4-10. Since these temperatures are based on the PWR and BWR assemblies having the highest cladding stress levels, the maximum heat loads can be applied to all UMS design basis contents.

3.4.7 Evaluation of Package Performance for Normal Conditions of Transport

Results of thermal analysis of the Universal Transport Cask containing PWR and BWR fuel under normal conditions of transport are summarized in Tables 3.4-1 through 3.4-3. The maximum fuel rod cladding temperature is maintained below 810°F (432°C); temperatures of safety-related cask components are maintained within their safe operating ranges; and thermally induced stresses in combination with pressure and mechanical load stresses are shown in the structural analysis of Chapter 2.0 to be less than the allowable stresses. As shown in Section 3.4.2, the personnel barrier temperature of 153°F is below the allowable temperature of 185°F for exclusive use shipment. Therefore, the Universal Transport Cask can safely transport the design basis fuel under the normal conditions of transport specified in 10 CFR 71.71.

Figure 3.4-1 Three-Dimensional PWR Cask Finite Element Model

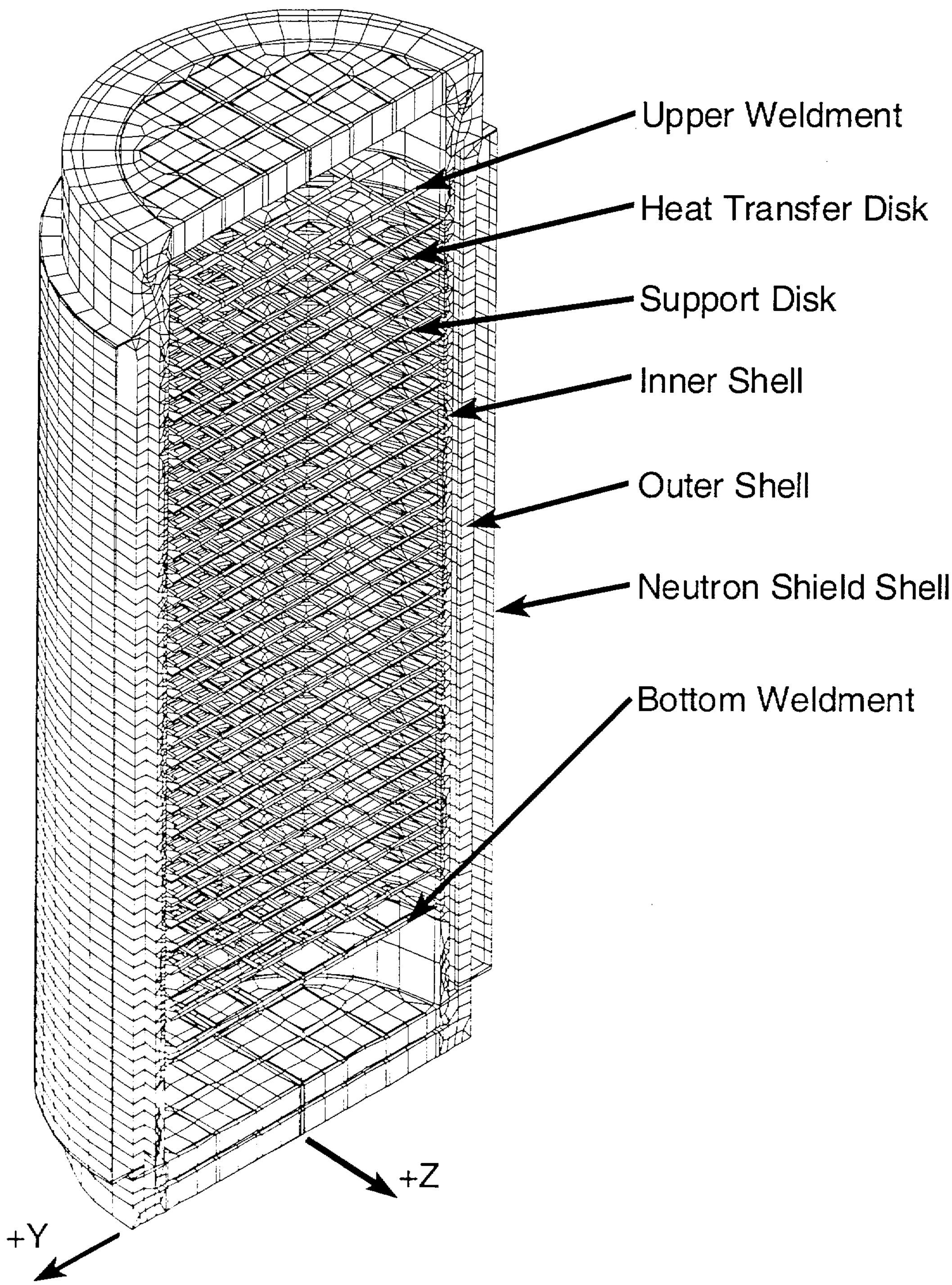


Figure 3.4-2 Design Basis PWR Fuel Assembly Axial Power Distribution

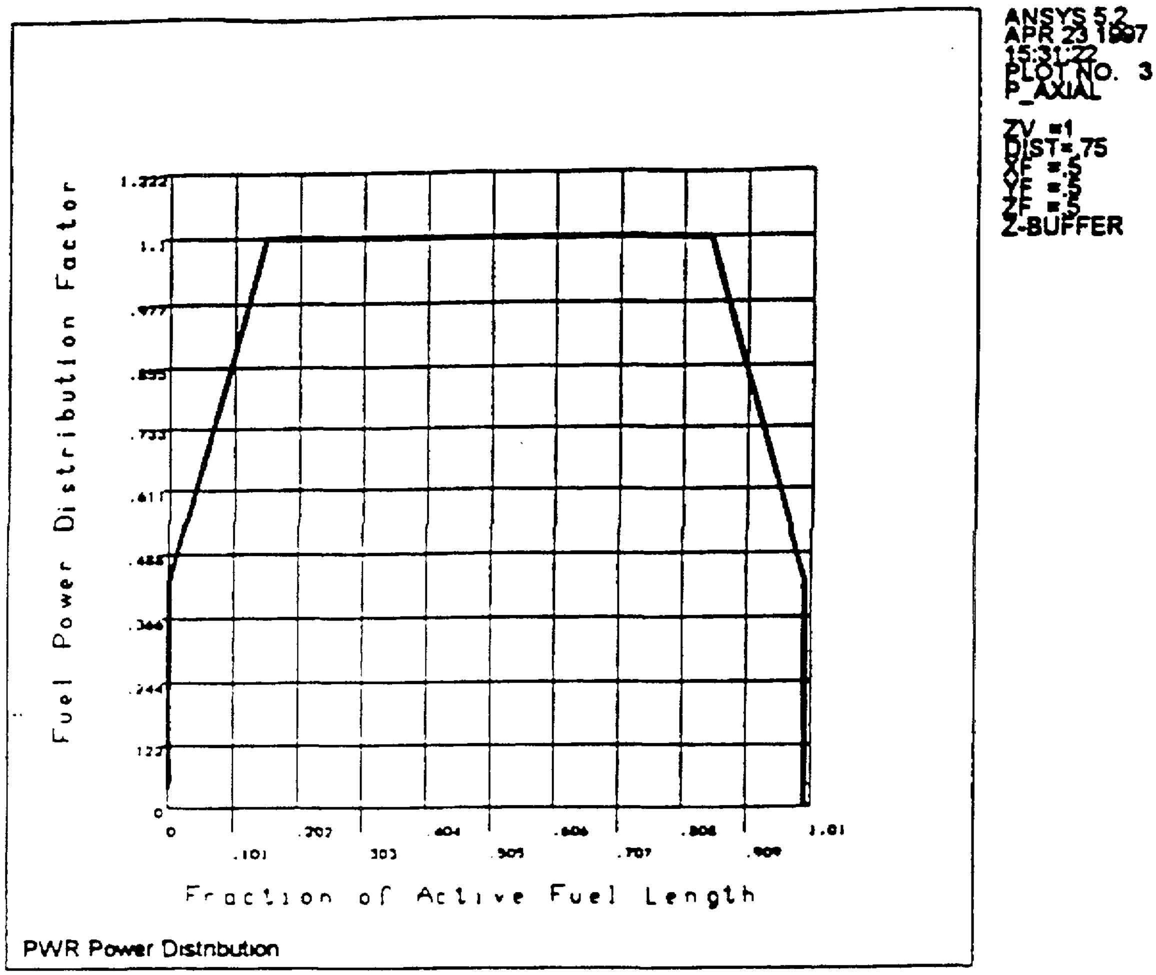


Figure 3.4-3 PWR 14x14 Fuel Assembly Two-Dimensional Finite Element Model

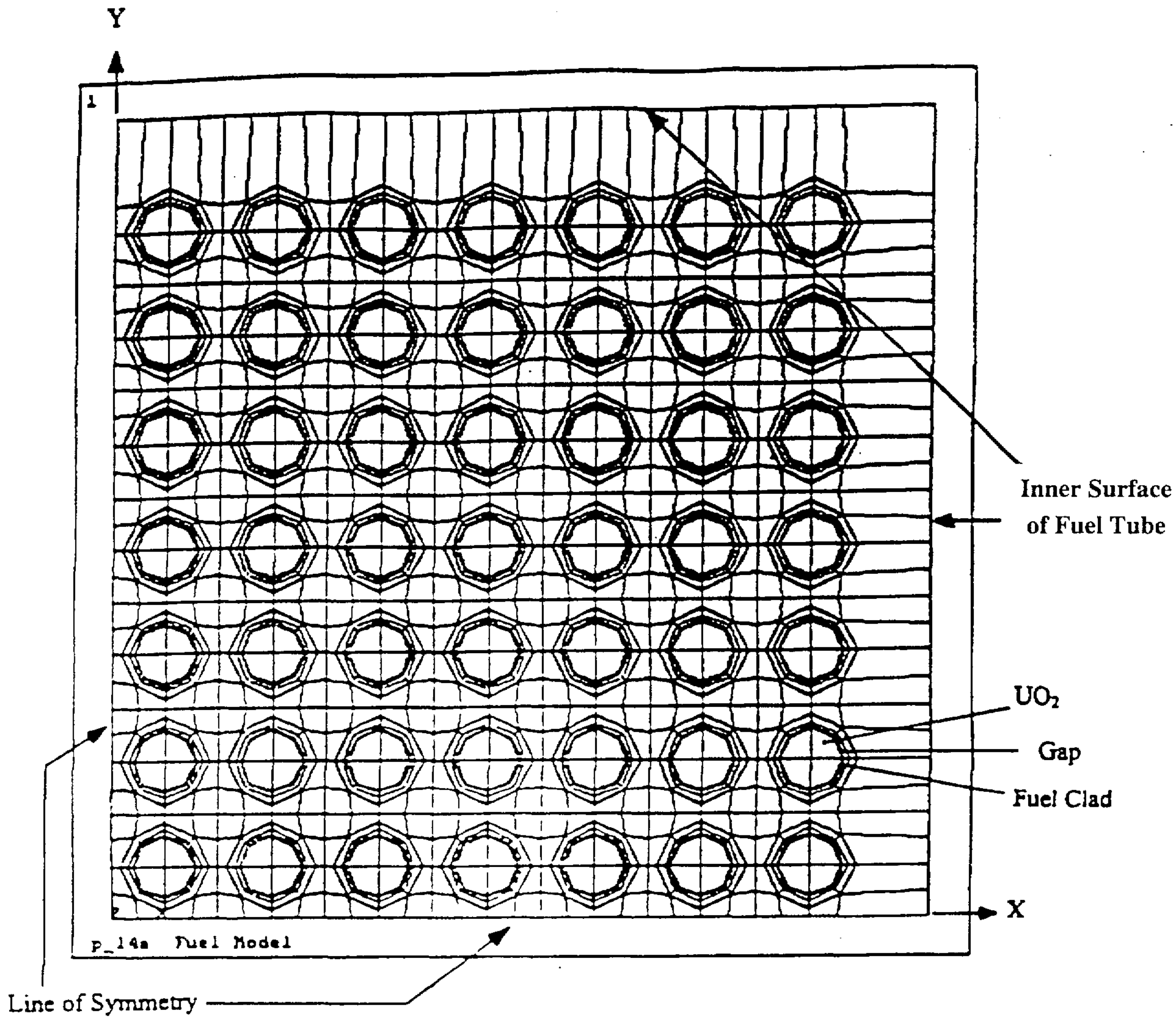


Figure 3.4-4 Two-Dimensional PWR Fuel Tube Model

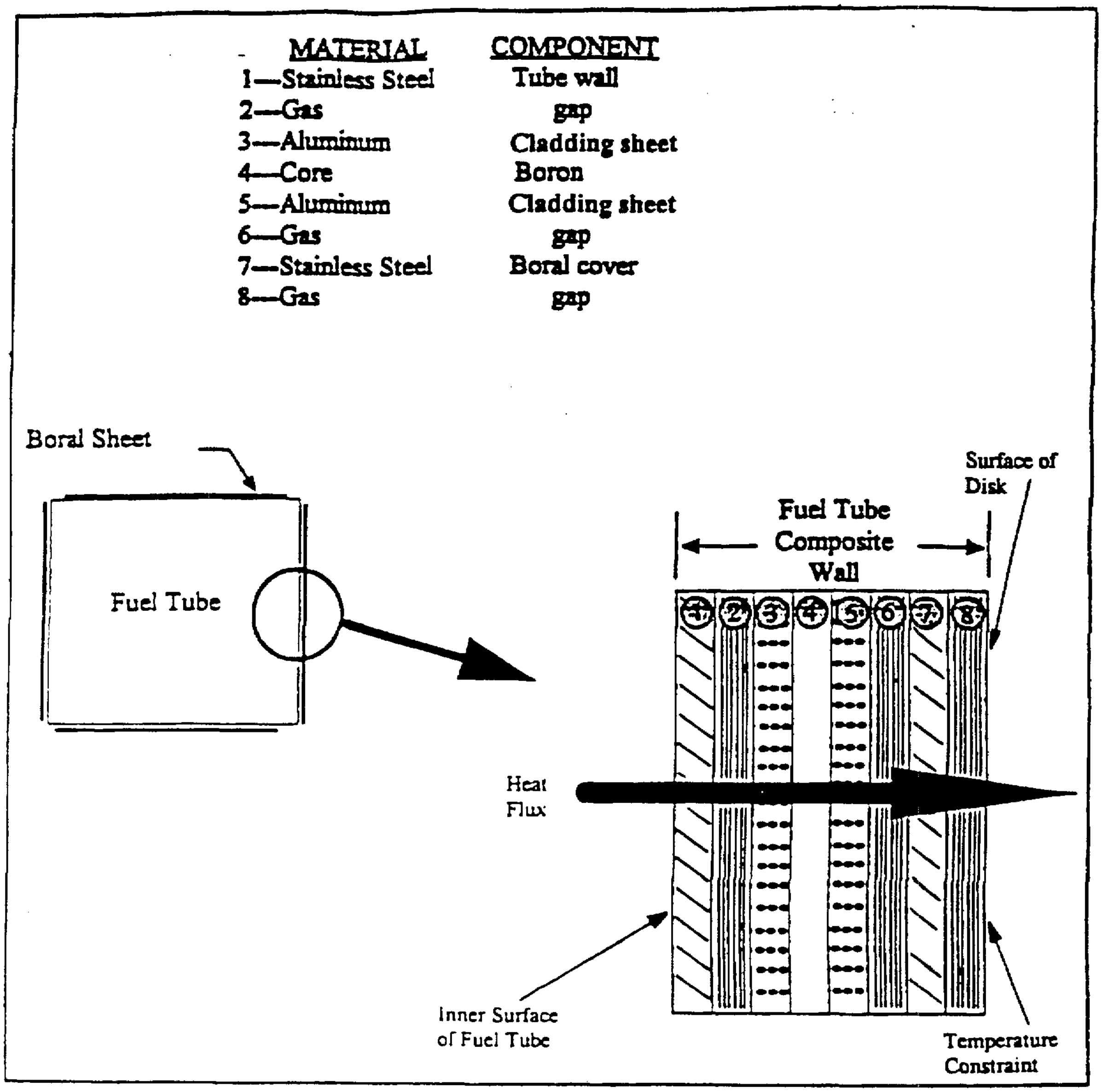
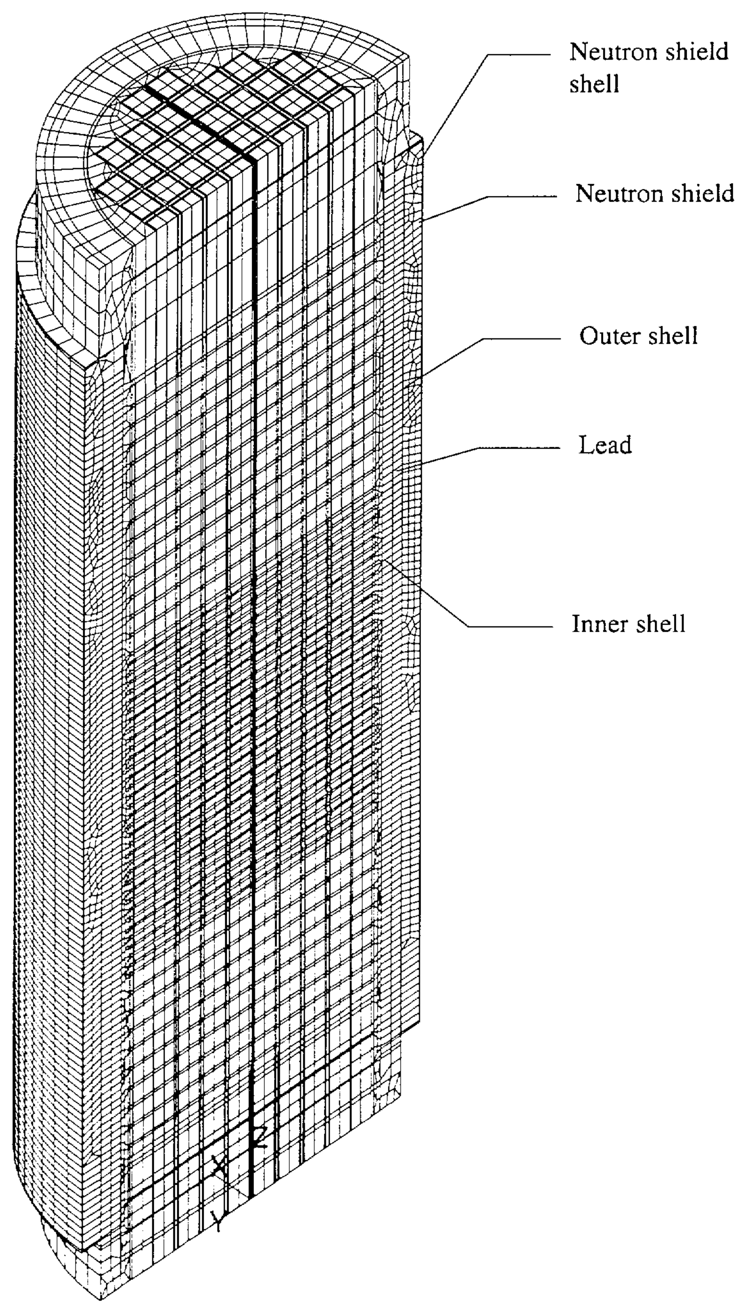
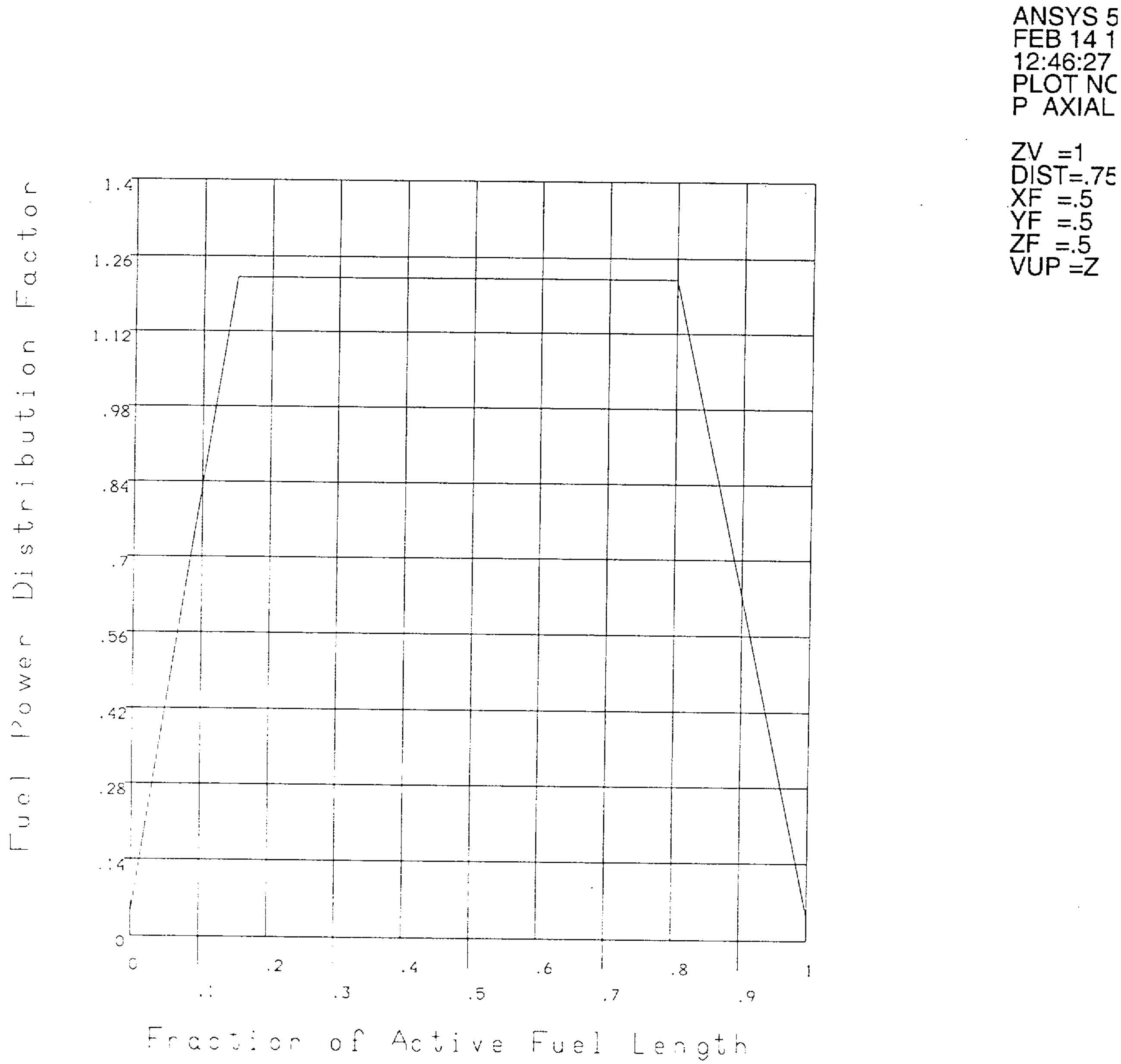


Figure 3.4-5 Three-Dimensional BWR Cask Finite Element Model



UMS 56 BWR Thermal Model

Figure 3.4-6 Design Basis BWR Fuel Assembly Axial Power Distribution



UMS56B Power Distribution

Figure 3.4-7 BWR 9x9 Fuel Assembly Two-Dimensional Finite Element Model

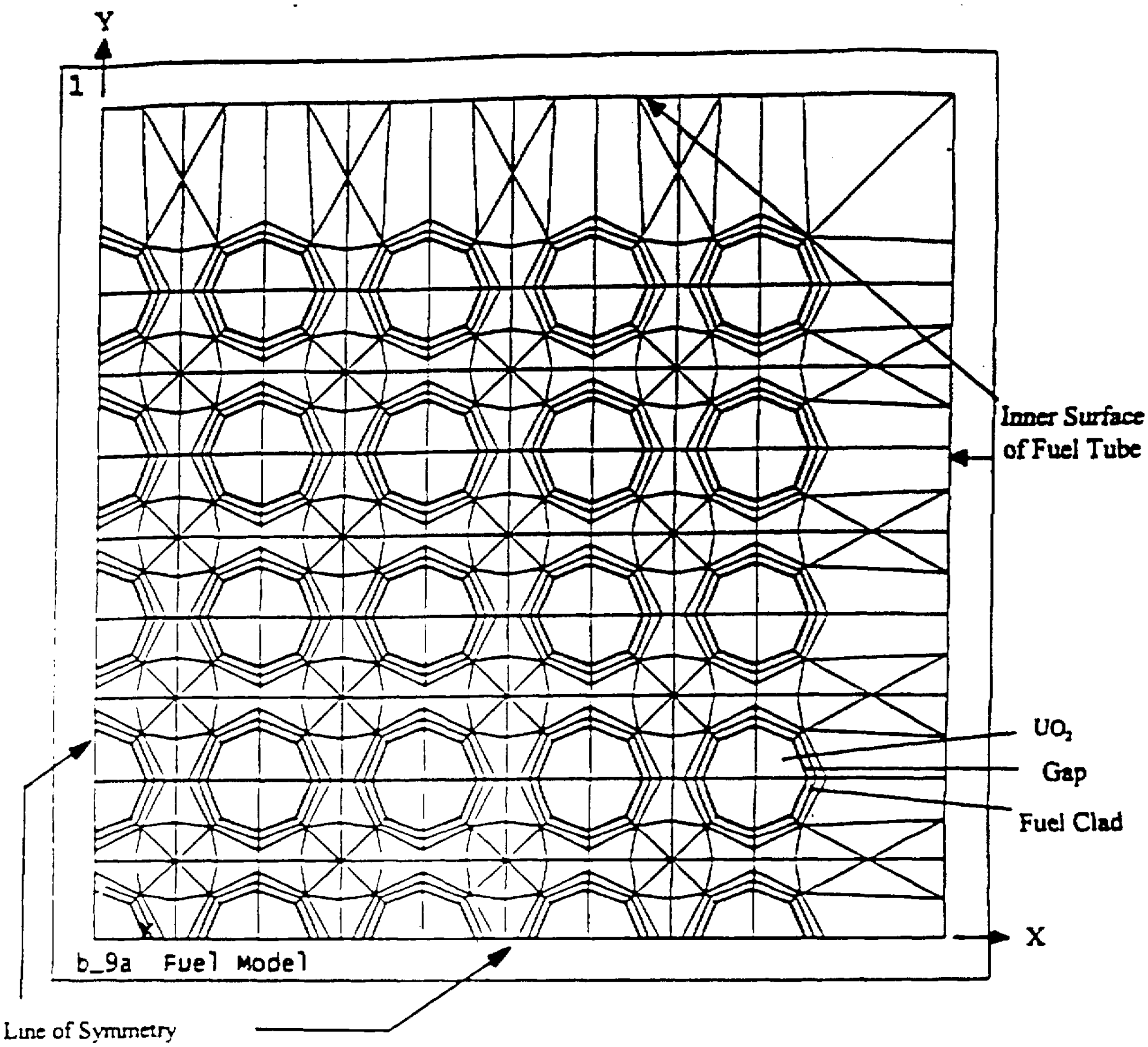


Figure 3.4-8 Two-Dimensional BWR Fuel Tube (with BORAL) Model

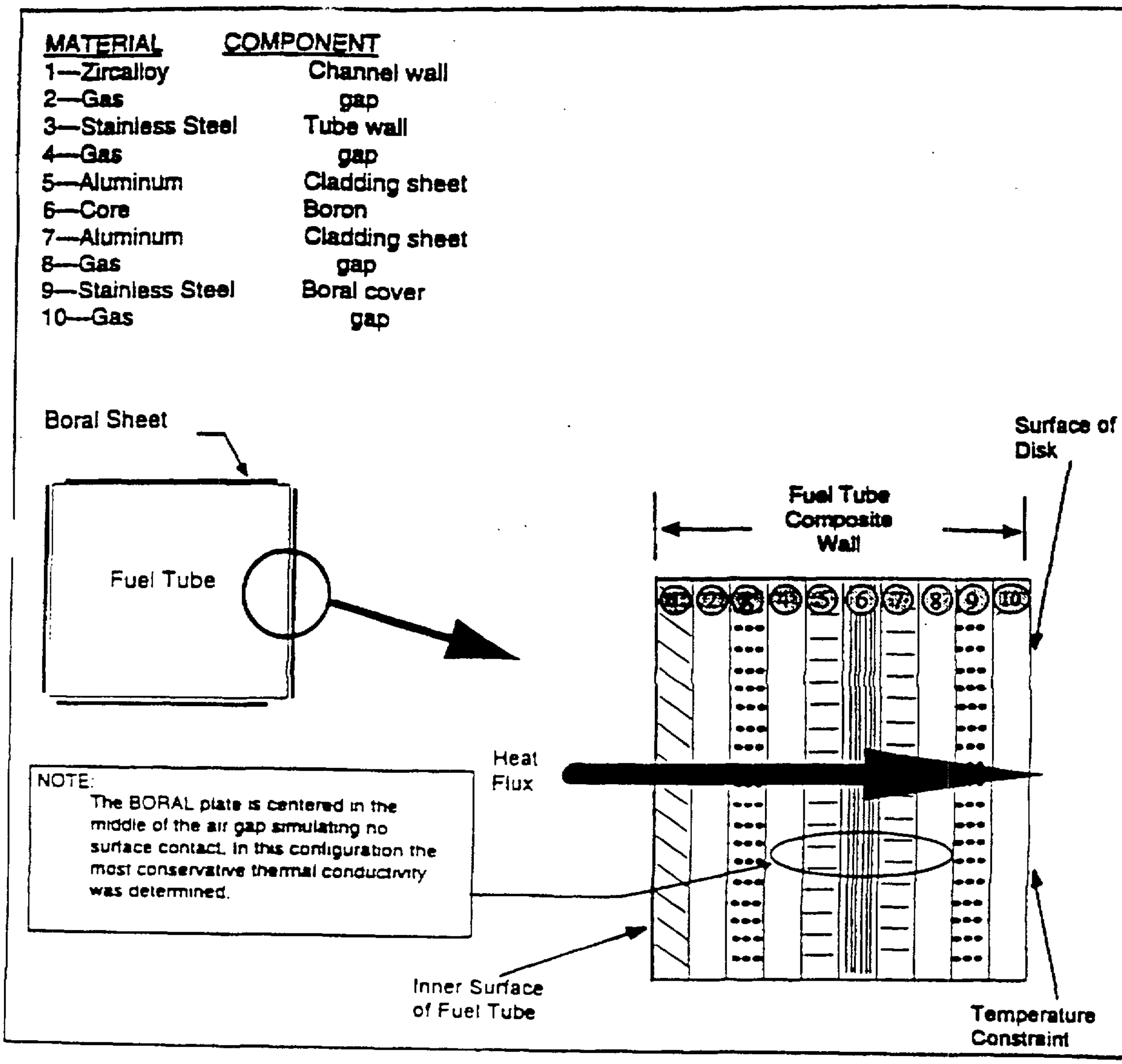


Figure 3.4-9 Two-Dimensional BWR Fuel Tube (without BORAL) Model

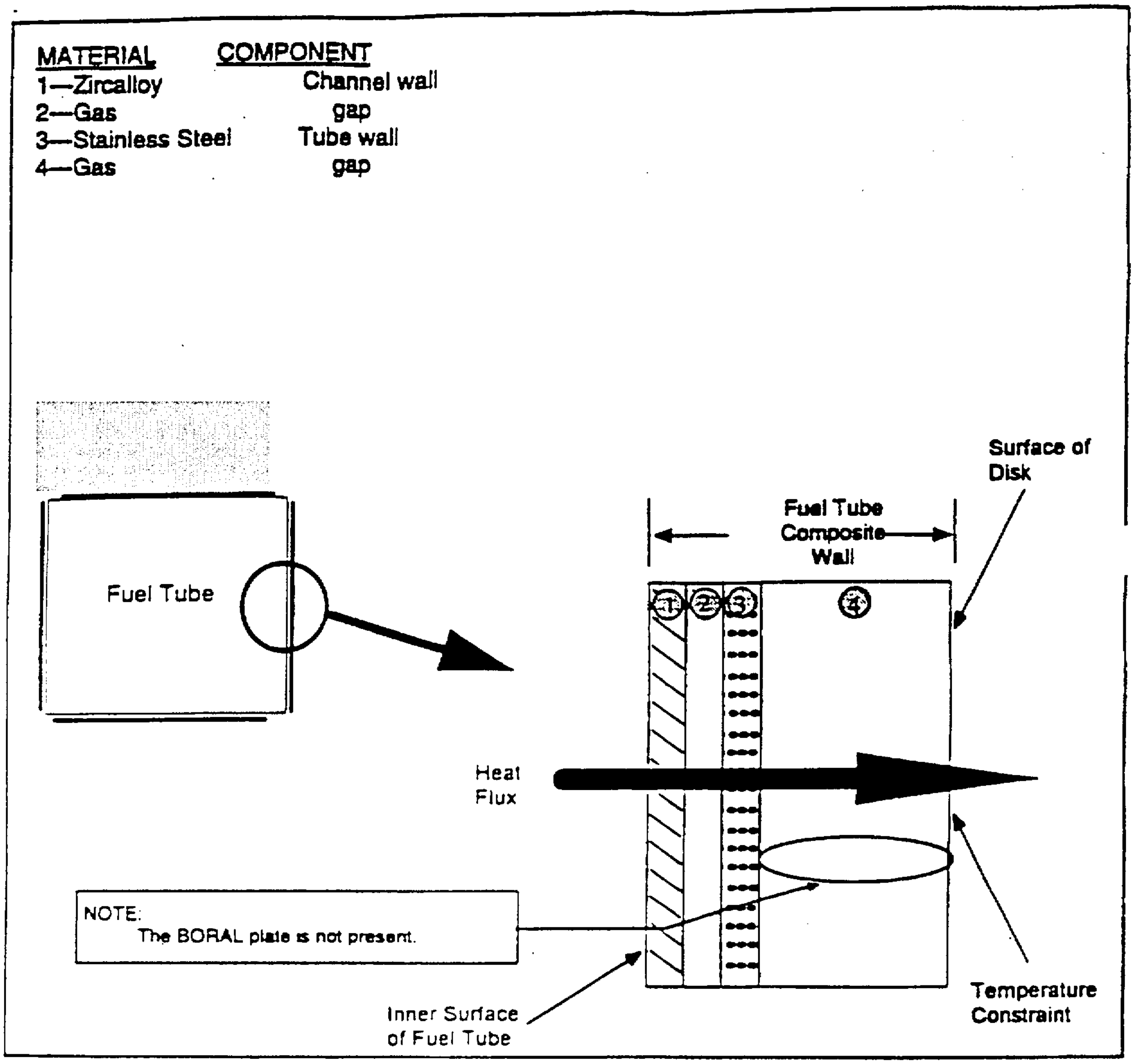


Figure 3.4-10 Cask Impact Limiter Thermal Model

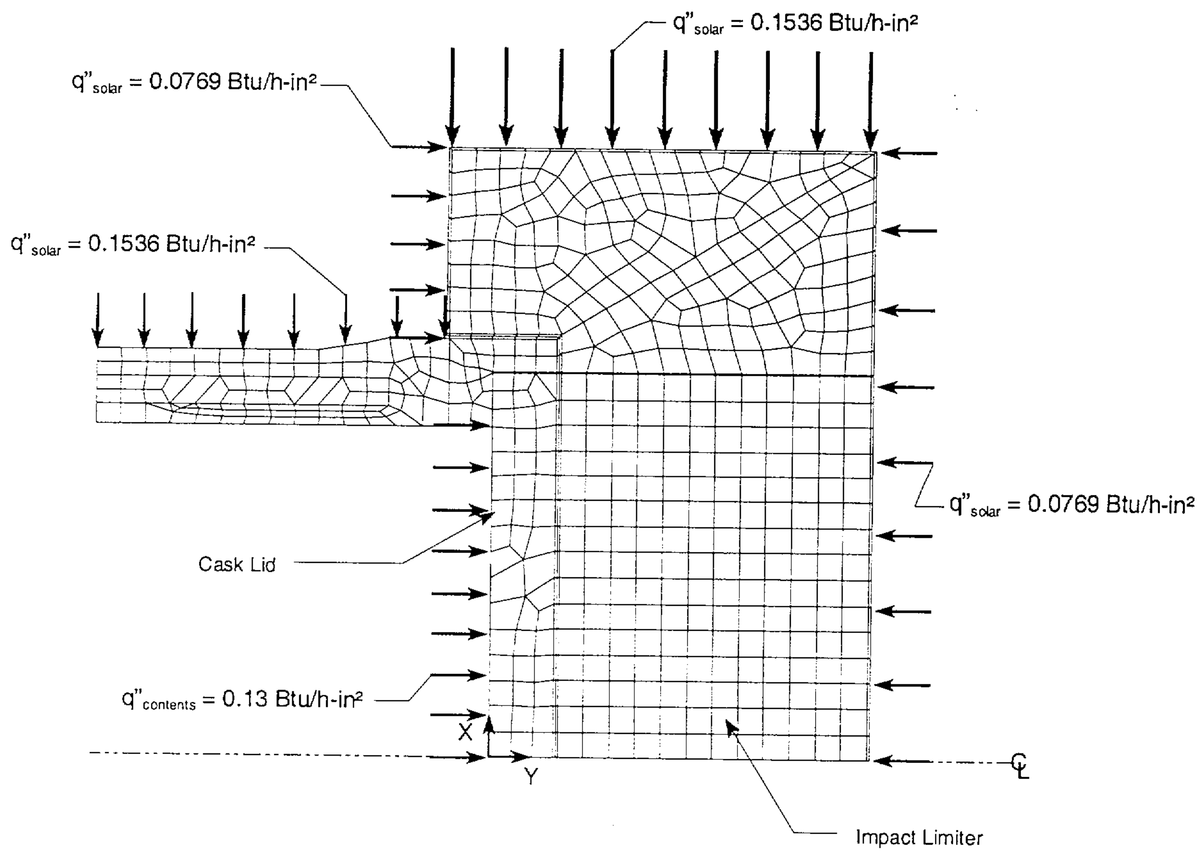


Figure 3.4-11 Personnel Barrier Thermal Model

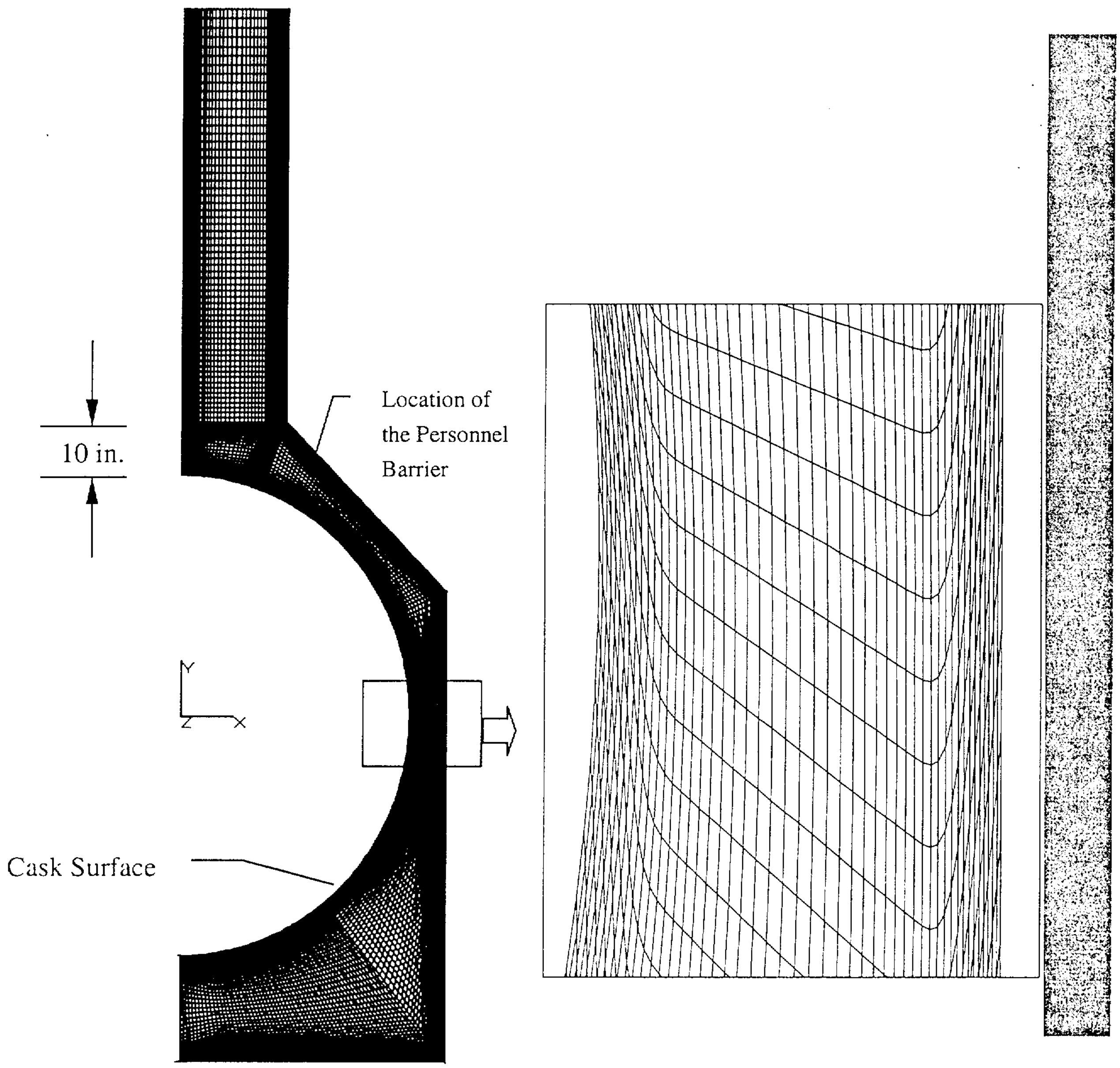
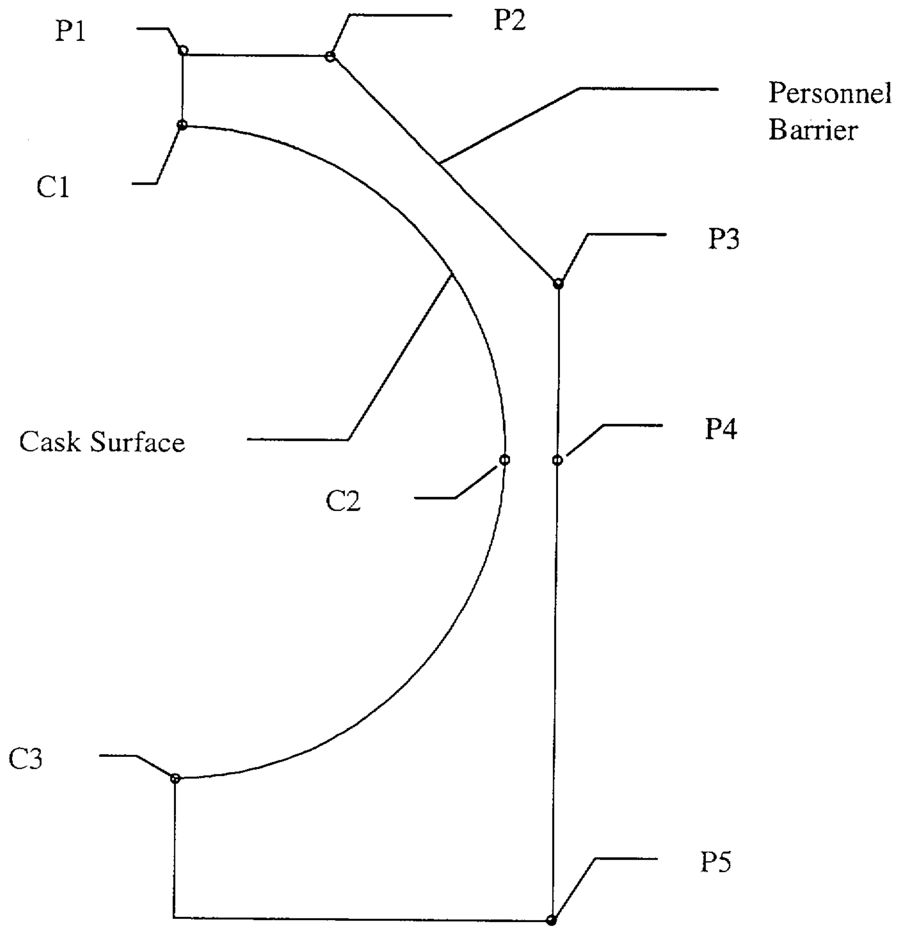


Figure 3.4-12 Temperature Results at Key Points of the Personnel Barrier



	Boundary Conditions			Calculated Temperature (°F)				
Location	C1	C2	C3	P1	P2	P3	P4	P5
Temperature	244	256	267	153	108	133	131	100

Figure 3.4-13 PWR Fuel Dry Storage Temperature versus Cladding Stress

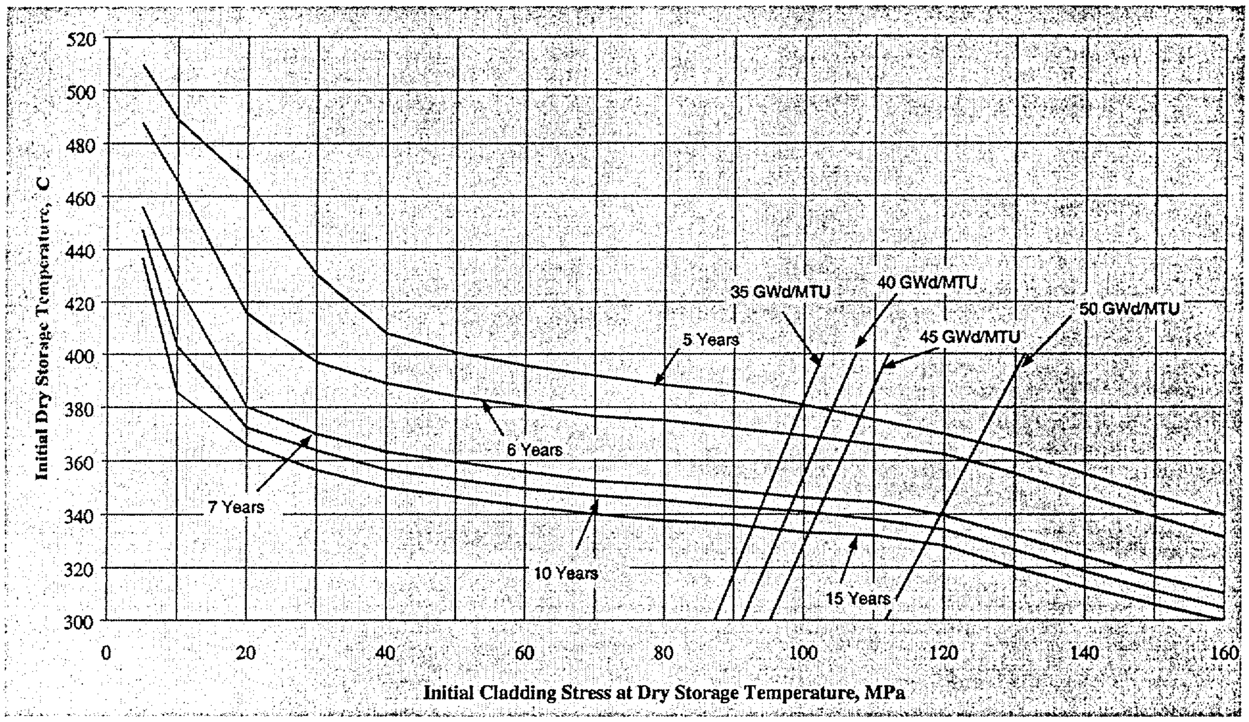


Figure 3.4-14 BWR Fuel Dry Storage Temperature versus Cladding Stress

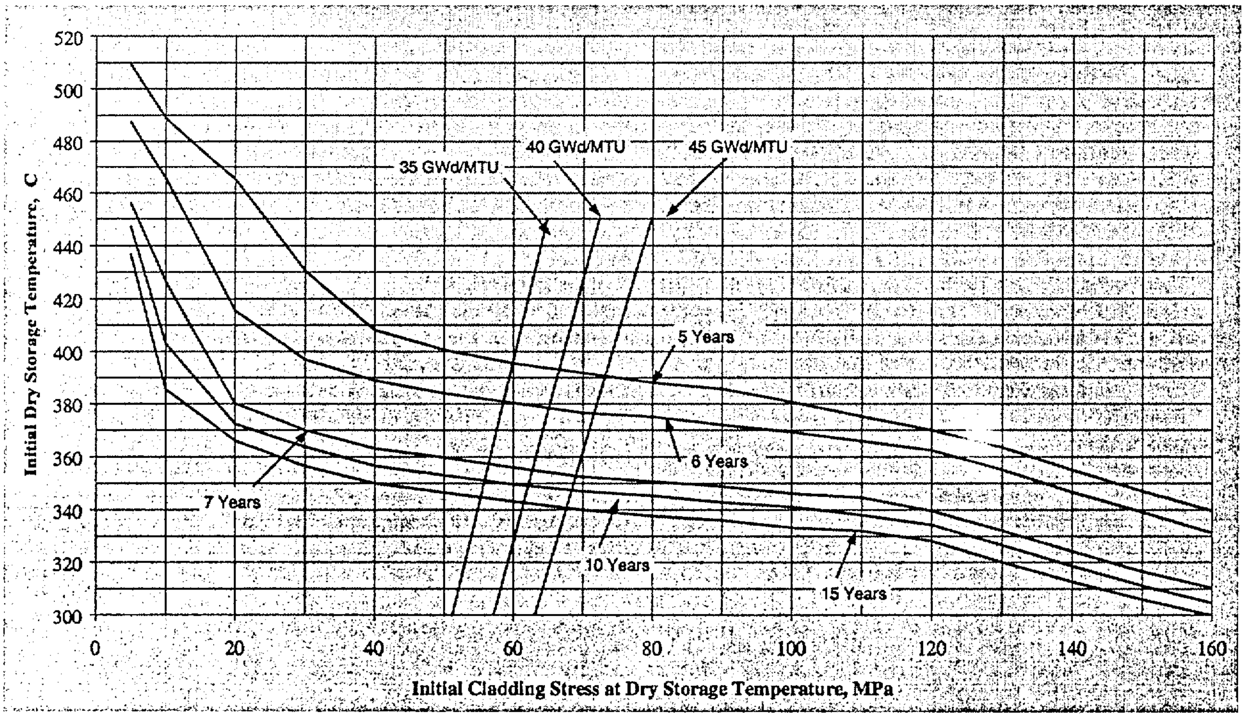


Figure 3.4-15 PWR Fuel Cladding Dry Storage Temperature versus Basket Heat Load

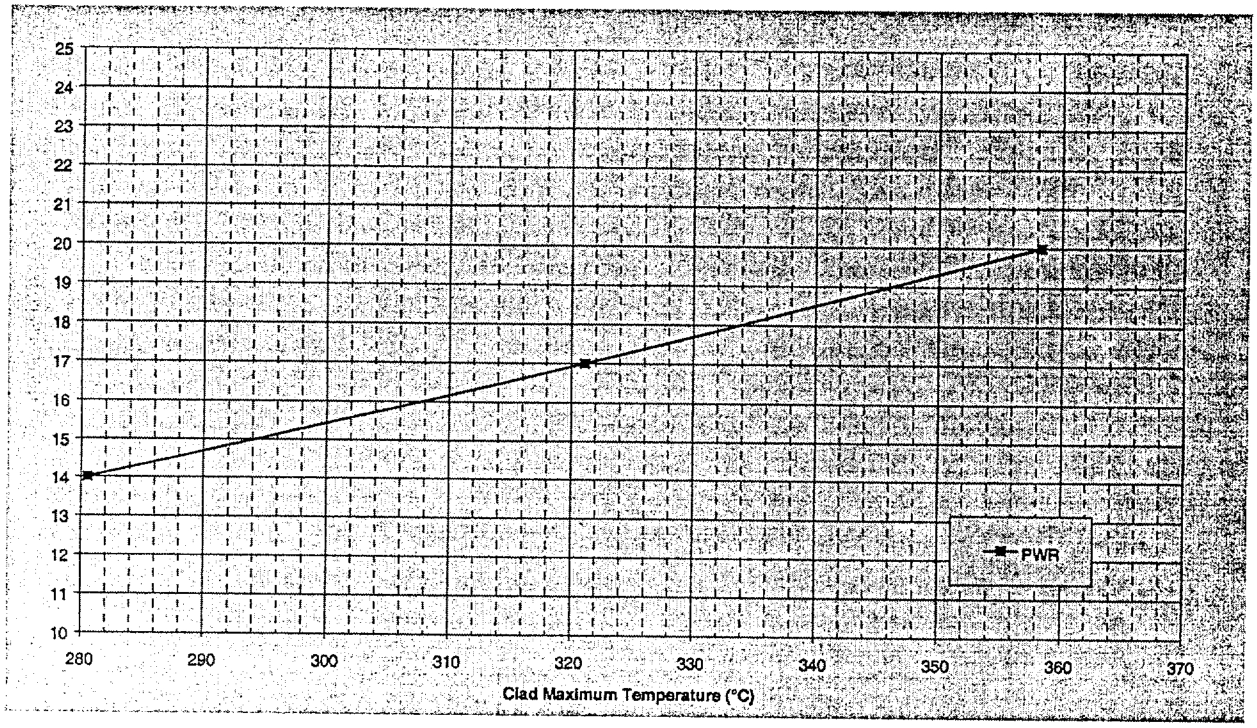


Table 3.4-1 Maximum Component Temperatures - Normal Conditions of Transport, Maximum Decay Heat, Maximum Ambient Temperature

Component	Temperature (°F) Cask with PWR Fuel Canister		Temperature (°F) Cask with BWR Fuel Canister	
	Canister Gas: Air [6]	Canister Gas: Helium	Canister Gas: Air [6]	Canister Gas: Helium
Cask Lid O-Rings/Vent Port O-ring[1]	268	266	207	204
Lower Drain Port O-ring[4]	225	224	232	230
Cask Radial Outer Surface	279	266	267	256
Radial Neutron Shield	314	293	301	286
Lead Gamma Shield	328	306	315	298
Aluminum Disk Exterior	292	268	321	298
Aluminum Disk Interior	683	605	608	515
Support Disk Exterior	271	255	213	208
Support Disk Interior	686	608	610	512
Canister Shell	399	408	369	363
Canister Shield Lid	272	270	212	208
Canister Bottom Plate	325	324	264	262
Maximum Fuel Rod Cladding	808	673	670	548
Cask Bottom	218	217	229	228
Bottom Forging	225	224	232	230
Inner Shell	368	344	336	312
Outer Shell	323	301	309	293
Top Forging[2]	256	250	196	194
Cask Lid	268	266	207	204
Cask Lid Bolt[3]	268	266	207	204
Average Gas Temperature in the Canisters[5]	492	453	432	366

Conditions:
100°F ambient temperature
20 kW decay heat load, 1.1 peaking factor - PWR
16 kW decay heat load, 1.22 peaking factor - BWR
Solar insolation
Cask cavity gas: helium
Canister cavity gas: air or helium

1. Cask lid O-rings and vent port O-rings not explicitly modeled—taken to be the maximum cask lid temperature.
2. Average temperature.
3. Cask lid bolts not explicitly modeled—taken to be the maximum temperature of the cask lid.
4. Lower drain port O-ring not explicitly modeled - taken to be the maximum temperature of the bottom forging.
5. Calculated as a volumetric average.
6. The design basis cover gas in the canister is helium. Temperature results for air as the cover gas are provided as worst-case temperatures for structural evaluation.

Table 3.4-2 Maximum Component Temperatures - Normal Conditions of Transport, Maximum Decay Heat, Minimum Ambient Temperature

Component	Temperature (°F) Cask with PWR Fuel Canister		Temperature (°F) Cask with BWR Fuel Canister	
	Canister Gas: Air [3]	Canister Gas: Helium	Canister Gas: Air [3]	Canister Gas: Helium
Cask Lid O-Rings/Vent Port O-ring[1]	154	140	67	62
Cask Radial Outer Surface	160	151	147	132
Radial Neutron Shield	197	178	181	162
Lead Gamma Shield	211	191	196	174
Maximum Basket[2]	599	505	517	404
Canister Shell	281	289	239	238
Canister Shield Lid	148	145	72	66
Canister Bottom Plate	208	205	131	127
Maximum Fuel Rod Cladding	744	578	589	440

Conditions: -40°F ambient temperature
20 kW decay heat load, 1.1 peaking factor - PWR
16 kW decay heat load, 1.22 peaking factor - BWR
No insolation
Cask cavity gas: helium
Canister cavity gas: air or helium

1. Cask lid O-ring and vent port O-rings not explicitly modeled—taken to be the maximum cask lid temperature.
2. Taken to be the greater of the maximum support disk and the maximum aluminum heat transfer disk temperatures.
3. The design basis cover gas in the canister is helium. Temperature results for air as the cover gas are provided as worst-case temperatures for structural evaluation.

Table 3.4-3 Universal Transport Cask Thermal Performance Summary for Component Operating Temperature

Temperature	Cask with PWR Fuel Canister (helium in cask cavity/helium in canister)	Cask with BWR Fuel Canister (helium in cask cavity/helium in canister)	Allowable Temperature Range
Maximum cladding temperature(°F)	673	548	< 716[1]
Component safe operating temperature ranges			
Cask lid O-rings	-40 to 266°F	-40 to 208°F	-40 to 300°F
Vent port coverplate O-ring	-40 to 266°F	-40 to 208°F	-40 to 300°F
Drain port coverplate-O-rings	-40 to 224°F	-40 to 230°F	-40 to 300°F
Radial NS-4-FR neutron shield	-40 to 293°F	-40 to 286°F	-40 to 300°F
Lead gamma shield	-40 to 306°F	-40 to 298°F	-40 to 600°F
Aluminum heat transfer disk	-40 to 605°F	-40 to 515°F	-40 to 700°F
PWR support disk	-40 to 608°F		-40 to 650°F
BWR support disk		-40 to 517°F	-40 to 700°F

1. In accordance with PNL-6189, the temperature limit of 380°C (716°F) is used for the evaluation of fuel considered in the design basis heat load (20 kW). For temperature limits corresponding to different burnup and cooling times, refer to Table 3.4-8.

Table 3.4-4 Maximum Internal Pressures for Normal Conditions of Transport

Fuel	Cavity	Condition	Pressure
PWR	Canister	3% fuel rod failure	1.44 atm ≈ 21.20 psia ≈ 6.50 psig
		100% fuel rod failure	4.20 atm ≈ 62.20 psia ≈ 47.50 psig
	Cask	3% fuel rod failure	1.50 atm ≈ 21.98 psia ≈ 7.28 psig
		100% fuel rod failure	3.86 atm ≈ 56.74 psia ≈ 41.96 psig
BWR	Canister	3% fuel rod failure	1.33 atm ≈ 19.56 psia ≈ 4.87 psig
		100% fuel rod failure	2.86 atm ≈ 42.04 psia ≈ 27.34 psig
	Cask	3% fuel rod failure	1.34 atm ≈ 19.74 psia ≈ 5.04 psig
		100% fuel rod failure	2.82 atm ≈ 41.44 psia ≈ 26.74 psig

Table 3.4-5 PWR Cladding Stress Level Comparison Chart

Fuel Type	B&W 15x15	B&W 17x17	CE 14x14	CE 16x16	WE 14x14	WE 15x15	WE 17x17
Rod OD (inch)	0.43	0.379	0.44	0.382	0.422	0.422	0.374
Cladding Thickness (inch)	0.0265	0.024	0.028	0.025	0.0225	0.0242	0.0225
Pellet OD (inch)	0.3686	0.3232	0.3765	0.325	0.3674	0.3659	0.3225
Active Fuel Length (inch)	144	143	137	150	145.2	144	144
Plenum Length (inch)	11.72	9.52	8.375	9.527	6.99	8.2	6.3
Spring Weight (lb)	0.042	0.026	0.1	0.1	0.07	0.044	0.037
Backfill Pressure (psig)	415	435	450	450	460	475	500
Fuel Mass (MTU)	0.4807	0.4658	0.4037	0.4417	0.4144	0.4646	0.4671
# of Fuel Rods	208	264	176	236	179	204	264
Free Volume ($inch^3$)	1.870	1.301	1.234	1.017	1.351	1.389	0.885
Pressure (psia) (380°C)	1357	1440	1636	1630	1612	1621	1793
Stress Level (Mpa)	73.8	76.9	85.8	83.8	104.2	96.2	101.9

Table 3.4-6 BWR Cladding Stress Level Comparison Chart

Fuel Type	EX 7x7	EX 8x8	EX 9x9	GE 7x7	GE 8x8a	GE 8x8b	GE 9x9
Rod OD (inch)	0.57	0.484	0.424	0.563	0.493	0.483	0.441
Cladding Thickness (inch)	0.036	0.036	0.03	0.032	0.034	0.032	0.028
Pellet OD (inch)	0.49	0.4045	0.3565	0.487	0.416	0.41	0.376
Active Fuel Length (inch)	144	150	150	144	144	150	150
Plenum Length (inch)	11.25	10.024	9.578	11.25	10.024	10.024	9.578
Spring Weight (lb)	0.13	0.1	0.047	0.13	0.1	0.1	0.047
Backfill Pressure (psig)	30	88.2	88.2	30	88.2	88.2	88.2
Fuel Mass (MTU)	0.196	0.1793	0.1666	0.1977	0.1855	0.1847	0.1979
# of Fuel Rods	48	62	74	49	63	62	79
Free Volume (inch^3)	2.631	1.708	1.469	3.084	1.929	1.912	1.758
Pressure (psia) (380°C)	1145	1369	1261	981	1257	1279	1189
Stress Level (MPa)	60.4	60.6	60.6	58.8	60.8	65.3	65.1

Table 3.4-7 Cladding Stress as a Function of Fuel Assembly Average Burnup and Temperature

	PWR		BWR	
Burnup	**300°C**	**400°C**	**300°C**	**450°C**
35,000 MWD/MTU	87.3 MPa	102.7 Mpa	51.2 MPa	64.8 MPa
40,000 MWD/MTU	91.3 MPa	107.4 Mpa	57.2 MPa	72.3 MPa
45,000 MWD/MTU	95.2 MPa	112.0 Mpa	63.1 MPa	79.8 MPa
50,000 MWD/MTU	111.7 MPa	131.4 Mpa	--	--

Table 3.4-8 Maximum Allowable Initial Storage Temperature (°C) as a Function of Initial Cladding Stress and Initial Cool Time

MPa	5 years	6 years	7 years	10 years	15 years
5	509.2	487.3	455.9	447	436.5
10	488.8	465.5	426.4	403	385.6
20	465.2	415.5	380.1	372.4	366
30	430.4	397	370.1	363.8	356.5
40	408.1	389	363.2	356.6	350
50	400.6	384	359.7	353.1	346.5
60	395.6	380.4	355.9	349.6	343.1
70	391.9	376.5	352.5	347	340
80	388.2	375	350.8	345.2	337.6
90	385.7	372	348.8	342.8	336.1
100	380.7	369.3	346.2	341	333.2
110	375.2	365.9	344.6	338	332.1
120	370	362.4	339.5	334.3	328.2
130	363.5	355.2	332.2	326.6	320
140	355	346.6	324.2	318.6	312.6
150	346.9	339.1	316.5	311.2	306
160	339.6	331.4	310.3	304.7	299.9

Table 3.4-9 Maximum Allowable Cladding Temperature for PWR and BWR Fuel

Cool Time [years]	PWR Clad Temperature Limit [°C] Burnup (MWD/MTU)				BWR Clad Temperature Limit [°C] Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000	35,000	40,000	45,000	50,000
5	380	378	376	368	396	394	391	[redacted]
6	370	368	366	360	380	378	376	[redacted]
7	348	347	346	340	357	355	353	[redacted]
10	342	341	340	335	351	349	348	[redacted]
15	335	334	333	329	345	342	340	[redacted]

Table 3.4-10 Maximum Allowable Decay Heat for PWR and BWR Systems

Cool Time [years]	PWR Decay Heat Limit[1] [kW] Burnup (MWD/MTU)				BWR Decay Heat Limit[1] [kW] Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000	35,000	40,000	45,000	50,000
5	20.00	20.00	19.87	19.28	16.00	16.00	16.00	[redacted]
6	19.43	19.28	19.14	18.70	16.00	16.00	16.00	[redacted]
7	17.82	17.74	17.67	17.23	16.00	16.00	16.00	[redacted]
10	17.38	17.30	17.23	16.87	16.00	16.00	16.00	[redacted]
15	16.87	16.79	16.72	16.43	16.00	16.00	16.00	[redacted]

1 Based on 5% Temperature Margin to Allowable.

3.5 Thermal Evaluation for Hypothetical Accident Conditions

This section provides thermal evaluation of the Universal Transport Cask containing PWR or BWR fuel under hypothetical accident conditions. The objective of the thermal analysis of the cask under hypothetical accident conditions is to demonstrate that the cask containment boundary structural components are maintained within their safe operating temperature ranges.

Because the fire accident is considered to be of short duration, the limits for maximum cladding temperature may be higher than that for normal conditions of transport. A cladding temperature limit of 1,058°F, however, is conservatively applied [3]. To determine their cumulative effect on the package, the tests specified in 10 CFR 71.73 are to be performed or analyzed in sequence. Thus, the Universal Transport Cask is analyzed for the fire transient, specified in 10 CFR 71.73(c)(4), assuming that the package is in a form consistent with the damage sustained in the free-drop and puncture tests of 10 CFR 71.73.

3.5.1 Thermal Models

Finite element models are used in the thermal evaluation of the Universal Transport Cask under hypothetical accident conditions. The same model is used to evaluate the cask transporting the PWR and the BWR fuel. Heat flux is applied to the inner shell surface of the cask model to simulate the decay heat generation. The distribution of the heat flux corresponds to the power distribution shown in Figures 3.4-2 and 3.4-6 for PWR and BWR fuel, respectively.

The environmental conditions and decay heat loads for the analysis are provided as discussed in Section 3.5. Convection during the fire accident has been considered. Results are given in the form of maximum component temperatures in Tables 3.5-1 (PWR) and 3.5-2 (BWR).

3.5.1.1 Analytical Models

Taking advantage of the symmetry of geometry and thermal loads of the cask about its major axis, the two finite element models of the cask (with PWR and with BWR fuel) are two-dimensional axis-symmetric representations of the cask. The finite element models are

constructed by using ANSYS two-dimensional thermal elements (PLANE55) with the axis-symmetric option activated. The spent fuel and basket are not explicitly modeled. The maximum temperatures of the basket components and fuel cladding are calculated by adding the maximum temperature difference between the cask inner shell and the component of interest from the normal condition results (Table 3.4-1, air cases) to the peak temperature of the inner shell. The inner shell peak temperatures for the hypothetical accident case are provided in Tables 3.5-1 and 3.5-2 for the PWR and BWR fuel, respectively. The cask model for PWR and BWR used in the accident condition evaluation are shown in Figures 3.5-1 through 3.5-3.

In each model, the cask body is modeled as three concentric shells: the inner stainless steel shell, the lead shielding, and the outer stainless steel shell. The portions of the lead region which extend above and below the neutron shield are protected by a layer of low conductivity material that effectively insulates the lead from the heat of the fire. Because the canister, fuel basket, and fuel assemblies are not explicitly modeled, no gas gaps occur in the models—all heat transfer through the models is by means of conduction only. The gap between the cask and lead is conservatively ignored, resulting in a greater heat input to the cask.

The analyses of the finite element models are composed of three distinct phases:

1. Initial conditions (steady-state): Maximum decay heat of the fuel; ambient temperature = 100°F; solar insolance.

2. 30-min fire (transient): Maximum decay heat of the fuel; fire temperature = 1,475°F (including convection and radiation); no solar insolance.

3. Postfire cool-down (transient): Maximum decay heat of the fuel; ambient temperature = 100°F; solar insolance.

For the first two phases of the analyses, the effective conductivity of the radial neutron shield is calculated in the same manner presented in Section 3.4.1.1.1. At the end of the 30-min fire transient, the neutron shield is considered to be voided of NS-4-FR, so that only the Cu/SS fins and stainless steel shell remain. The effective conductivity for this arrangement is then recalculated as discussed in Section 3.4.1.1.1. Air is substituted for the NS-4-FR material. The thickness of the fireblock material in an uncompressed state is .12 inches, but in the model .03 in. is used.

The effect of impact limiters is included in the model for the fire analysis. Previous scale model tests of the NAC-STC cask have demonstrated that the impact limiters remain on the cask after the 30-ft drop imposed by the hypothetical accident condition. The UMS® impact limiters are nearly identical to those of the NAC-STC. The fire transient models include natural convection and thermal radiation boundary conditions during all phases of the analyses and account for solar insolation effects in the pre- and postfire transient phase. The natural convection during the fire is modeled with a convection coefficient of $0.01222\Delta T^{(0.33)}$ (Btu/hr in^{2}°F [12] [5]. After the fire, the convection coefficient as described in Section 3.2.3 is used. The natural convection and thermal radiation boundary conditions are applied to all external cask surfaces not covered by the impact limiters. The solar insolance boundary condition is applied to the external surface of the neutron shield shell. During all phases of the analyses, the areas of the cask covered by the impact limiters are modeled as adiabatic surfaces.

3.5.1.2 Test Model

The thermal analyses presented in Section 3.5.3 demonstrate that the Universal Transport Cask is capable of meeting the design basis temperature requirements under hypothetical accident conditions. The methodology used in this analysis is conservative, consistent with those used in prior transport cask licensing, and sufficient to show that the cask meets the criteria set forth in Section 3.5. Therefore, no thermal test model is created.

3.5.2 Package Conditions and Environment

As demonstrated in Chapter 2.0, the Universal Transport Cask body sustains no major damage as a result of the free-drop and puncture events as demonstrated in Chapter 2.0. Therefore, the cask body is modeled in an undamaged configuration.

The emissivity of stainless steel is 0.36. However, during the 30-min fire portion of the transient analysis, the emissivity is assumed to be 0.9. Also, the emissivity of the fire is assumed to be 1.0.

At the end of the fire, the NS-4-FR in the neutron shield is assumed to be destroyed. The result is a lower conductivity and thus a greater resistance to heat leaving the cask. The emissivity of stainless steel is again assumed to be 0.36, also providing a greater resistance to heat leaving the cask. The cooldown is analyzed for a period of 18 hr after the end of the fire. At the end of the cooldown period, all cask components have already reached their maximum temperatures and have begun to cool down to their postfire, steady state temperatures.

3.5.3 Package Temperatures

The ANSYS computer code is used to evaluate the Universal Transport Cask for the hypothetical accident fire. A steady-state initial temperature profile is calculated on the basis of a 100°F ambient temperature and solar insolance and used as input for the 30-min fire transient, which considers exposure of the cask to a 1,475°F radiant environment. This exposure is followed by an 18-hr cooldown period, which considers exposure of the cask to a 100°F ambient temperature and solar insolance.

The safe operating temperature ranges of the components specified in Section 3.3.2 are also evaluated for the fire accident. These components include the seals and lead gamma shielding. The radial neutron shield temperature is not considered to be significant; therefore its loss is assumed in this accident. The shielding consequences of loss of the radial neutron shield are provided in Section 5.4.2.3.

The maximum component temperatures during the hypothetical fire accident and cooldown period are provided in Tables 3.5-1 (PWR) and 3.5-2 (BWR). The tables also show the maximum component temperatures for the fuel cladding, and the lead in the cask body. None of the safety-related components, with the exception of the radial neutron shield as noted previously, exceeds its safe operating temperature as a result of the fire accident. The temperature histories of the major cask components are shown in Figures 3.5-4 through 3.5-11 for PWR and Figures 3.5-12 through 3.5-19 for BWR.

3.5.4 Maximum Internal Pressures

In the following sections, the maximum internal pressures for the hypothetical accident conditions are calculated for the PWR and BWR Transportable Storage Canisters and for the Universal Transport Cask cavity. The maximum internal pressure for hypothetical accident conditions are summarized in Table 3.5-3.

3.5.4.1 Maximum Internal Pressure for PWR Fuel Canister and Cask

The maximum internal pressures in the PWR fuel canister and the cask are calculated for hypothetical accident conditions in the same manner described for normal conditions in Section 3.4.4. However, for accident conditions only the case in which 100% of the fuel rods fail releasing 30% of their total fission gas and all of the rod backfill gas is analyzed.

The calculated maximum post-fire accident average temperature of the gases within the PWR fuel canister with air as the backfill gas in the canister is 794°F. The average gas temperature with air as the cover gas within the canister is greater than the average gas temperature with helium as the cover gas. This average temperature of the PWR fuel canister cover gas is determined using the axis-symmetric cask model described in Section 3.5.1.1. As previously discussed in Section 3.4.4.1, the design basis PWR fuel assembly for the internal pressure calculation is the Westinghouse 17x17 fuel assembly.

3.5.4.1.1 Maximum Internal Pressure for PWR Fuel Canister (100% Fuel Rod Failure)

As previously determined in Section 3.4.4.1.1.2, the total quantity of gas in the PWR fuel canister with 100% failed fuel rods is:

$$N = 582 \frac{\text{Moles}}{\text{Canister}}$$

The maximum cavity temperature is 423°C. Using the ideal gas law, the hypothetical accident condition internal pressure in the cask with 100% failed fuel rods and the maximum thermal transient temperature, is:

$$P = \frac{\left(582 \frac{\text{Moles}}{\text{Canister}}\right) \times \left(0.0821 \frac{\text{atm}\,\ell}{\text{mole K}}\right) \times 696.34\ \text{K}}{\left(5{,}968 \frac{\ell}{\text{Canister}}\right)} = 5.60\text{atm} \approx 82.0\ \text{psia} \approx 67.30\ \text{psig}$$

3.5.4.1.2 Maximum Internal Pressure for Cask with PWR Fuel Canister (100% Fuel Rod Failure)

As previously determined in Section 3.4.4.1.2.2, the total quantity of gas in the cask with PWR fuel canister with 100% failed fuel rods is:

$$N = 627.24 \frac{\text{Moles}}{\text{Cask}}$$

The maximum cavity temperature is 423°C. Using the ideal gas law, the hypothetical accident condition internal pressure in the cask with 100% failed fuel rods, is:

$$P = \frac{\left(627.24\ \frac{\text{Moles}}{\text{Cask}}\right) \times \left(0.0821\ \frac{\text{atm}\,\ell}{\text{mole K}}\right) \times 696.34\ \text{K}}{\left(7{,}064.25\ \frac{\ell}{\text{Cask}}\right)} = 5.07\ \text{atm} \approx 74.56\ \text{psia} \approx 59.86\ \text{psig}$$

3.5.4.2 Maximum Internal Pressure for BWR Fuel Canister and Cask

The maximum internal pressures in the BWR fuel canister and the cask are calculated for hypothetical accident conditions in the same manner described for normal conditions in Section 3.4.4. However, for accident conditions only the case in which 100% of the fuel rods fail releasing 30% of their total fission gas and all of the rod backfill gas is analyzed.

The calculated maximum post-fire accident average temperature of the gases within the BWR fuel canister with air as the backfill gas in the canister is 700°F. The average gas temperature with air as the cover gas within the canister is greater than the average gas temperature with helium as the cover gas. This average temperature of the BWR fuel canister cover gas is determined using the axis-symmetric cask model described in Section 3.5.1.1. As previously discussed in Section 3.4.4.2, the design basis BWR fuel assembly for the internal pressure calculation is the Exxon-ANF 9x9 fuel assembly.

3.5.4.2.1 Maximum Internal Pressure for BWR Fuel Canister (100% Fuel Rod Failure)

As previously determined in Section 3.4.4.2.1.2, the total quantity of gas in the BWR fuel canister with 100% failed fuel rods is:

$$N = 467.58\ \frac{\text{Moles}}{\text{Canister}}$$

The maximum canister temperature is 371°C. Using the ideal gas law, the hypothetical accident condition internal pressure in the canister with 100% failed fuel rods, is:

$$P = \frac{\left(467.58\,\frac{\text{Moles}}{\text{Canister}}\right)\times\left(0.0821\,\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 644.17\,\text{K}}{\left(6{,}649.81\,\frac{\ell}{\text{Canister}}\right)} = 3.72\,\text{atm} \approx 54.66\,\text{psia} \approx 39.96\,\text{psig}$$

3.5.4.2.2 Maximum Internal Pressure for Cask with BWR Fuel Canister (100% Fuel Rod Failure)

As previously determined in Section 3.4.4.2.2.2, the total quantity of gas in the cask with BWR fuel canister with 100% failed fuel rods is:

$$N = 477.80\,\frac{\text{Moles}}{\text{Cask}}$$

The maximum cavity temperature is 371°C. Using the ideal gas law, the hypothetical accident condition internal pressure in the cask with 100% failed fuel rods, is:

$$P = \frac{\left(477.8\,\frac{\text{Moles}}{\text{Cask}}\right)\times\left(0.0821\,\frac{\text{atm}\,\ell}{\text{mole K}}\right)\times 644.11\,\text{K}}{\left(6{,}895.73\,\frac{\ell}{\text{Cask}}\right)} = 3.66\,\text{atm} \approx 53.86\,\text{psia} \approx 39.2\text{psig}$$

3.5.5 Maximum Thermal Stresses

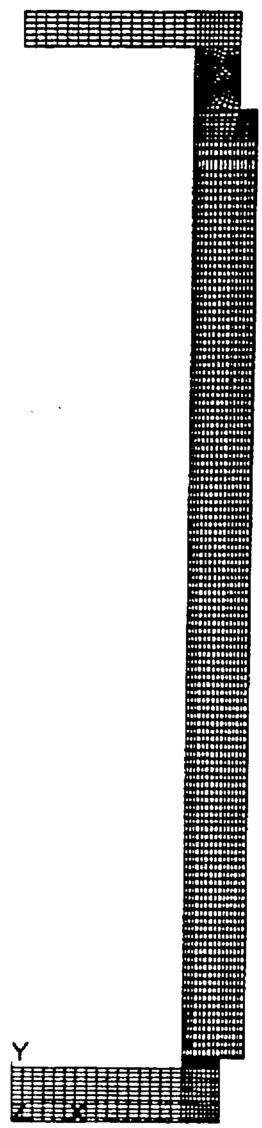
The maximum thermal stresses in the cask and the cask contents resulting from the hypothetical accident fire are not calculated. Thermal stresses are secondary stresses. Evaluation of secondary stresses is not required by the ASME code for accident conditions.

3.5.6 Evaluation of Package Performance for Hypothetical Accident Conditions

The Universal Transport Cask thermal performance has been assessed for the hypothetical accident fire transient, as specified in 10 CFR 71. Except for the radial neutron shield, which is

assumed to be lost, all cask components important to safety remain within their safe operating ranges. The ability of the cask to safely contain its radioactive contents is not compromised.

Figure 3.5-1 Two-Dimensional Axis-Symmetric Finite Element Cask Model (PWR and BWR)



Fire Transient Thermal Model

Figure 3.5-2 Upper Region of Two-Dimensional Axis-Symmetric Cask Finite Element Model (PWR and BWR)

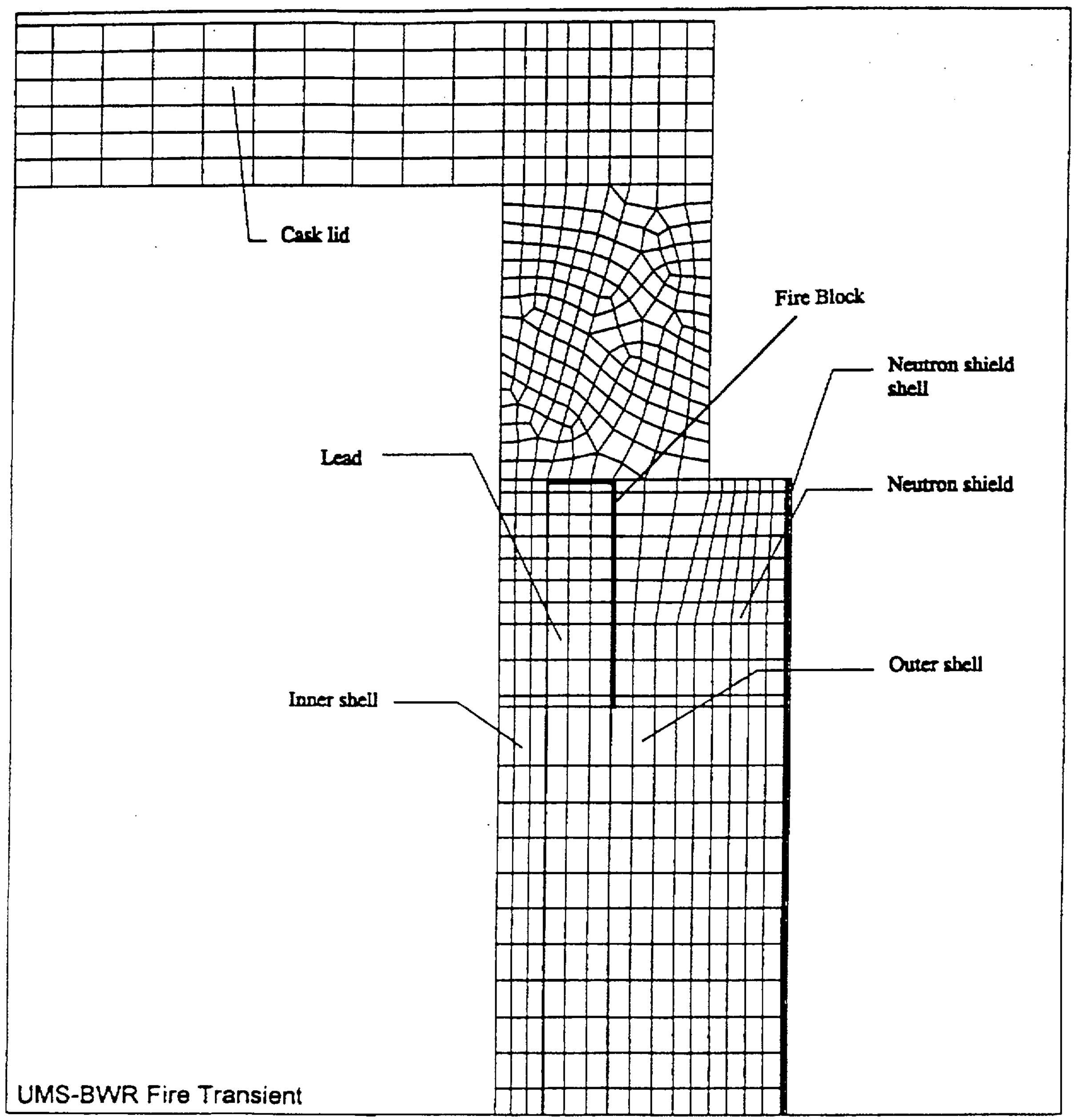


Figure 3.5-3 Lower Region of Two-Dimensional Axis-Symmetric Cask Finite Element Model (PWR and BWR)

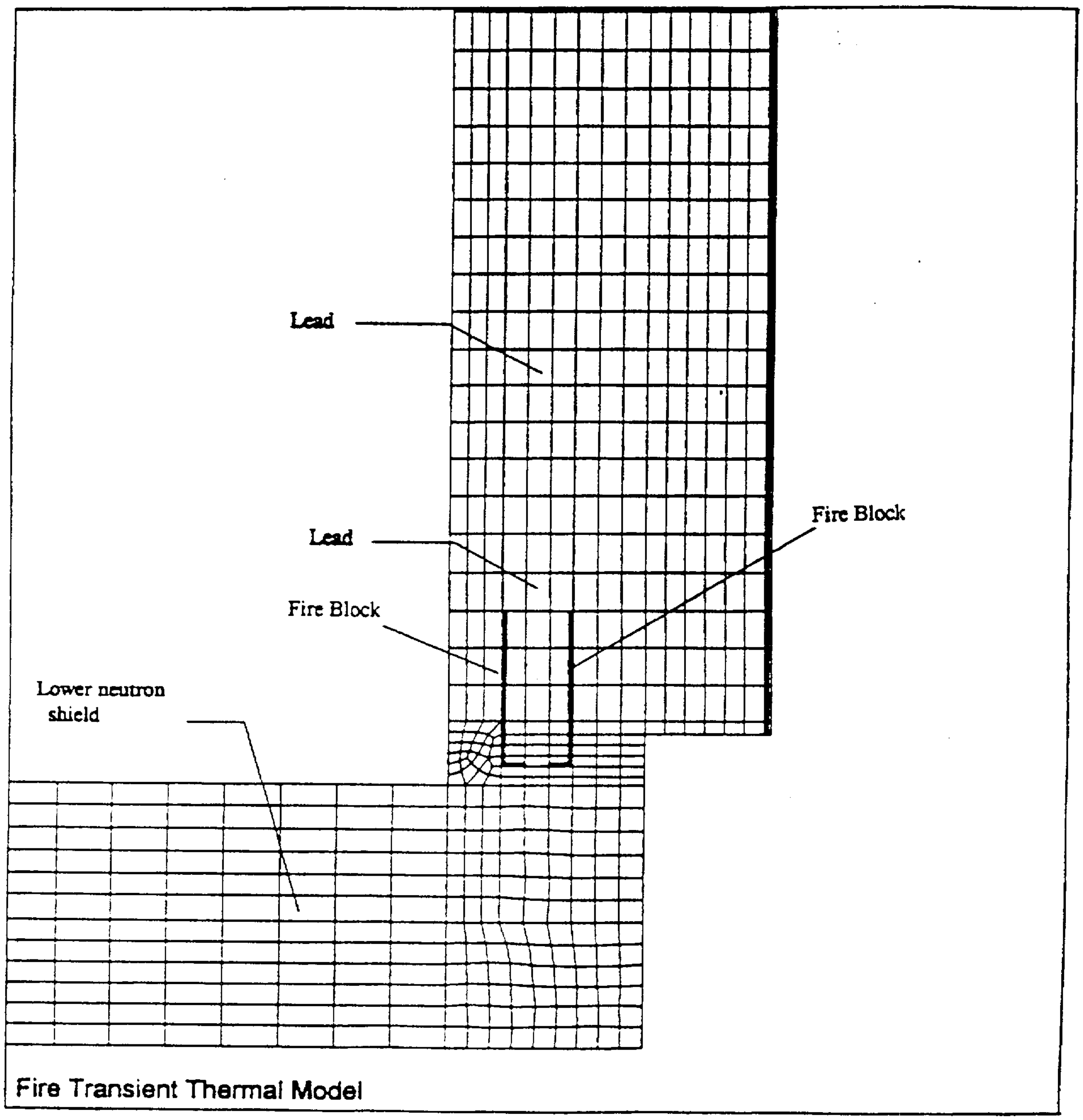


Figure 3.5-4 Hypothetical Accident Conditions Maximum Lead Temperature History (PWR)

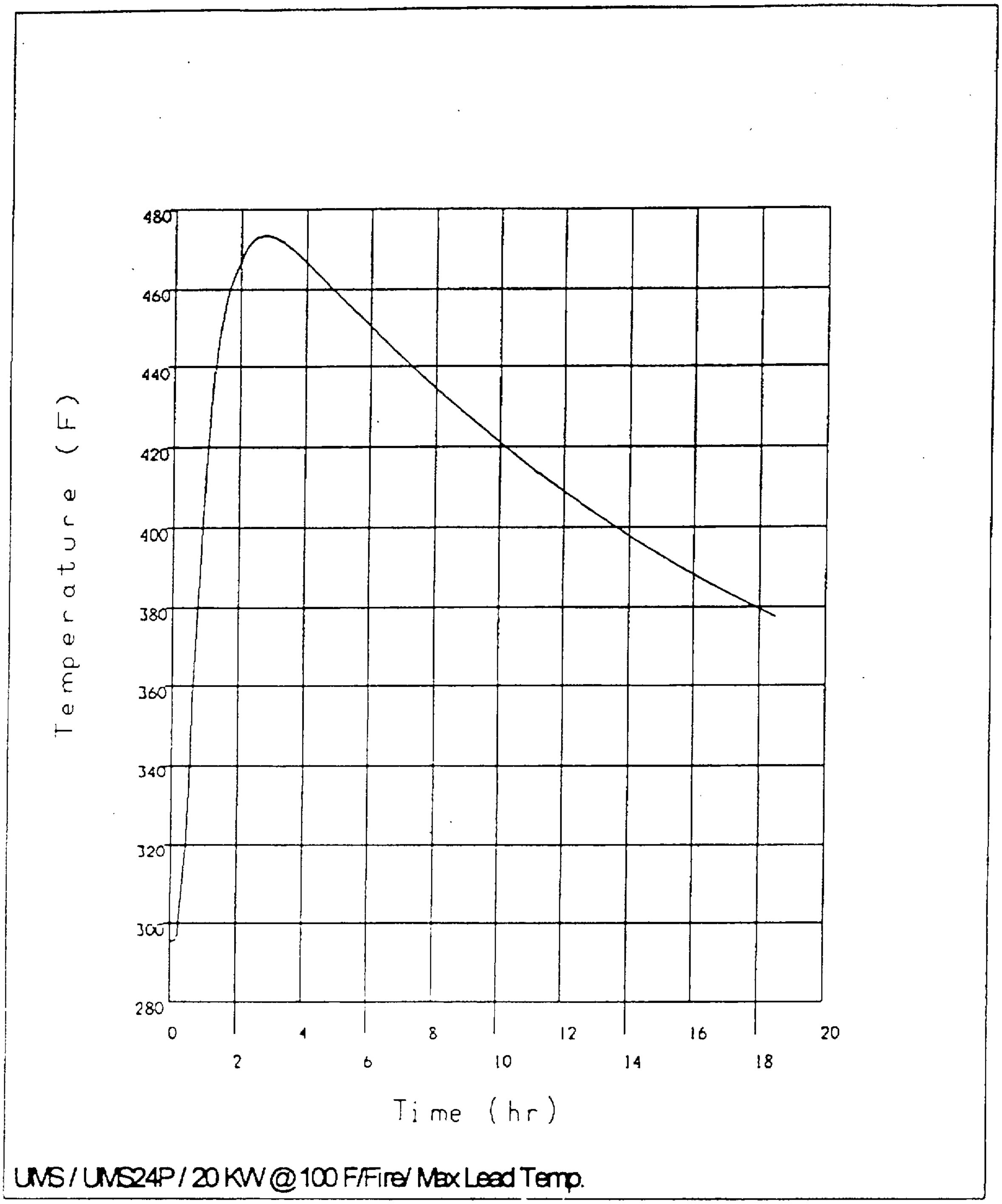


Figure 3.5-5 Hypothetical Accident Conditions Maximum Neutron Shield Exterior Temperature History (PWR)

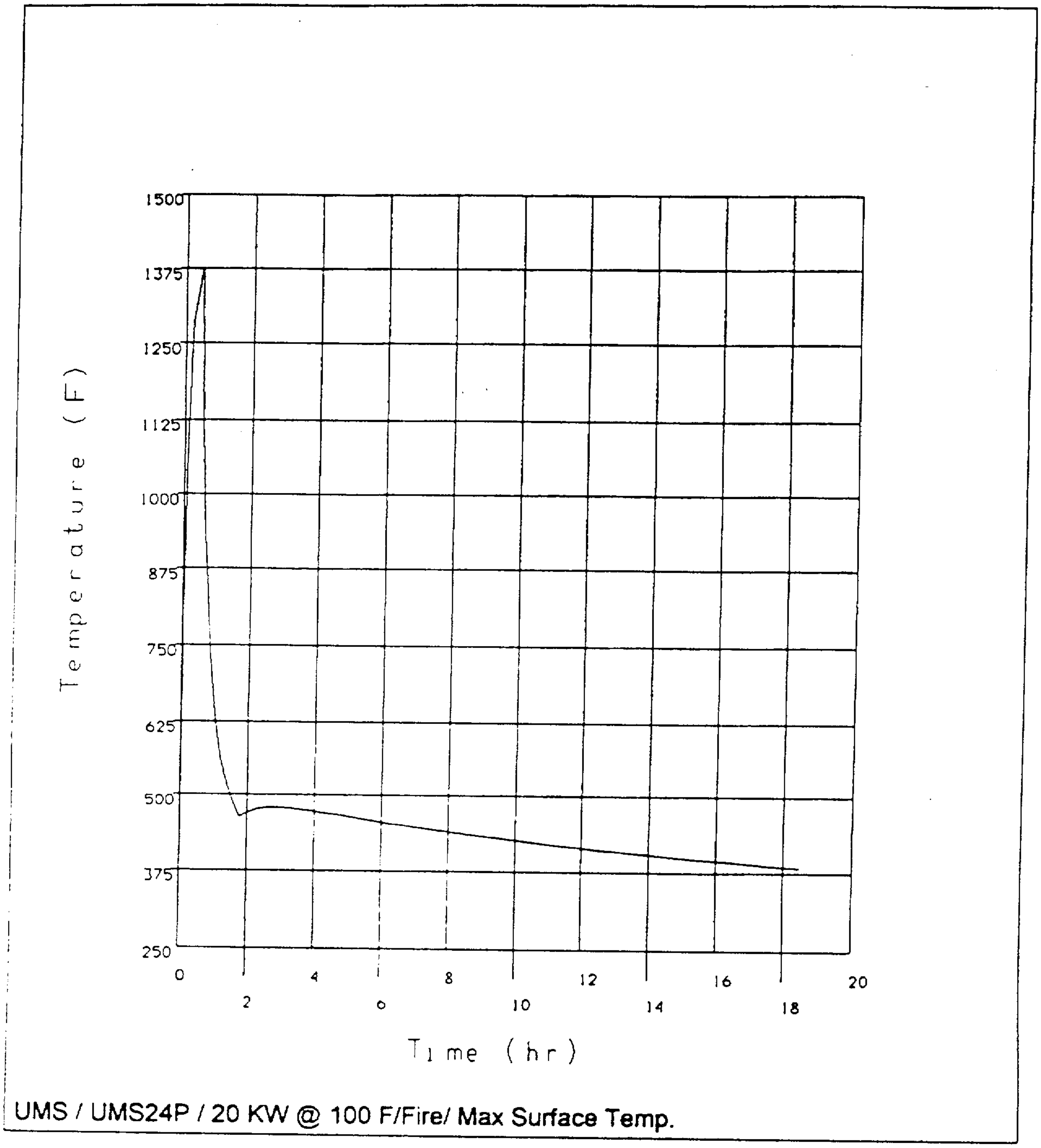


Figure 3.5-6 Hypothetical Accident Conditions Maximum Cask Inner Shell Temperature History (PWR)

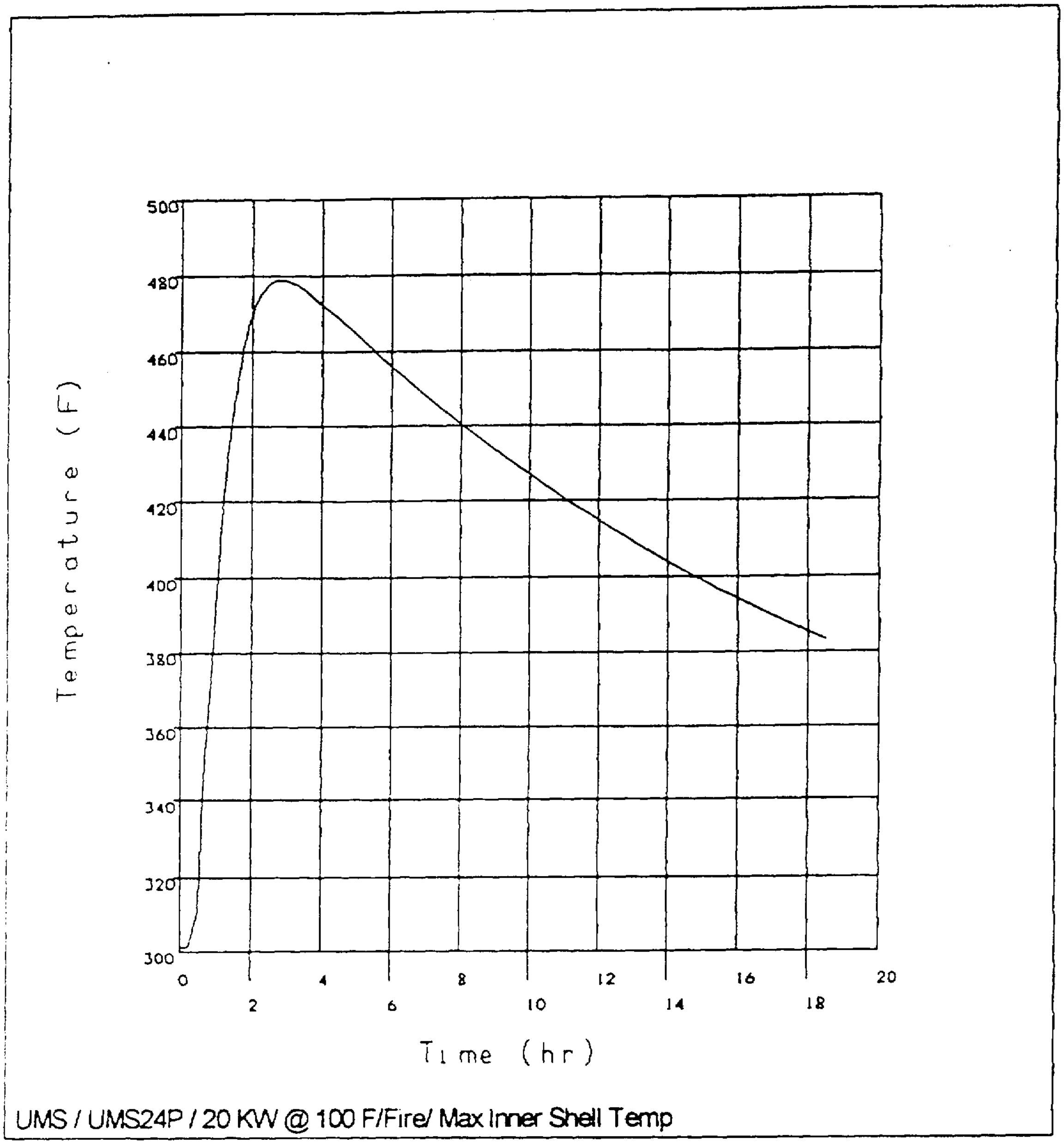


Figure 3.5-7 Hypothetical Accident Conditions Maximum Cask Outer Shell Temperature History (PWR)

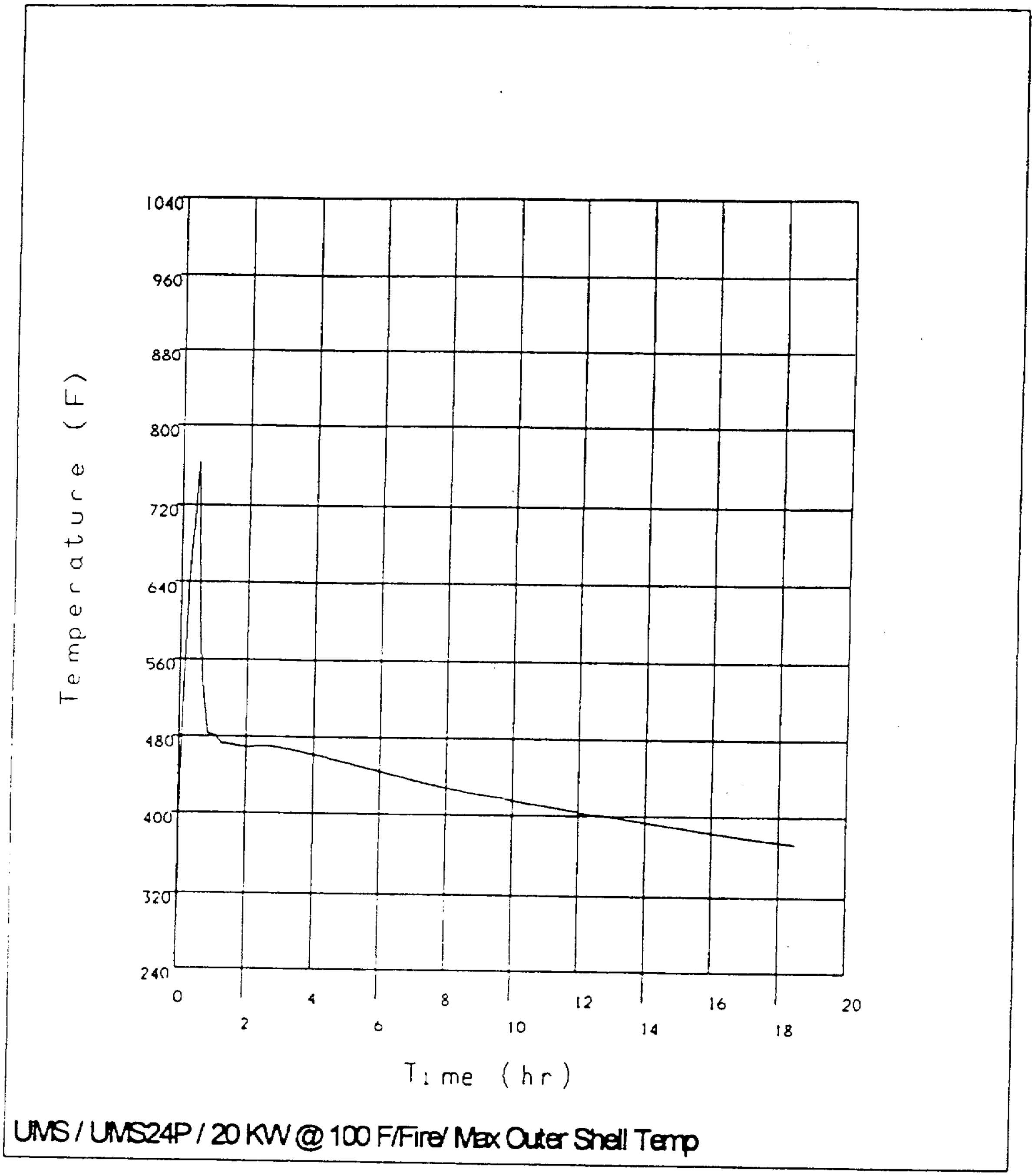


Figure 3.5-8 Hypothetical Accident Conditions Maximum Lower Neutron Shield Temperature History (PWR)

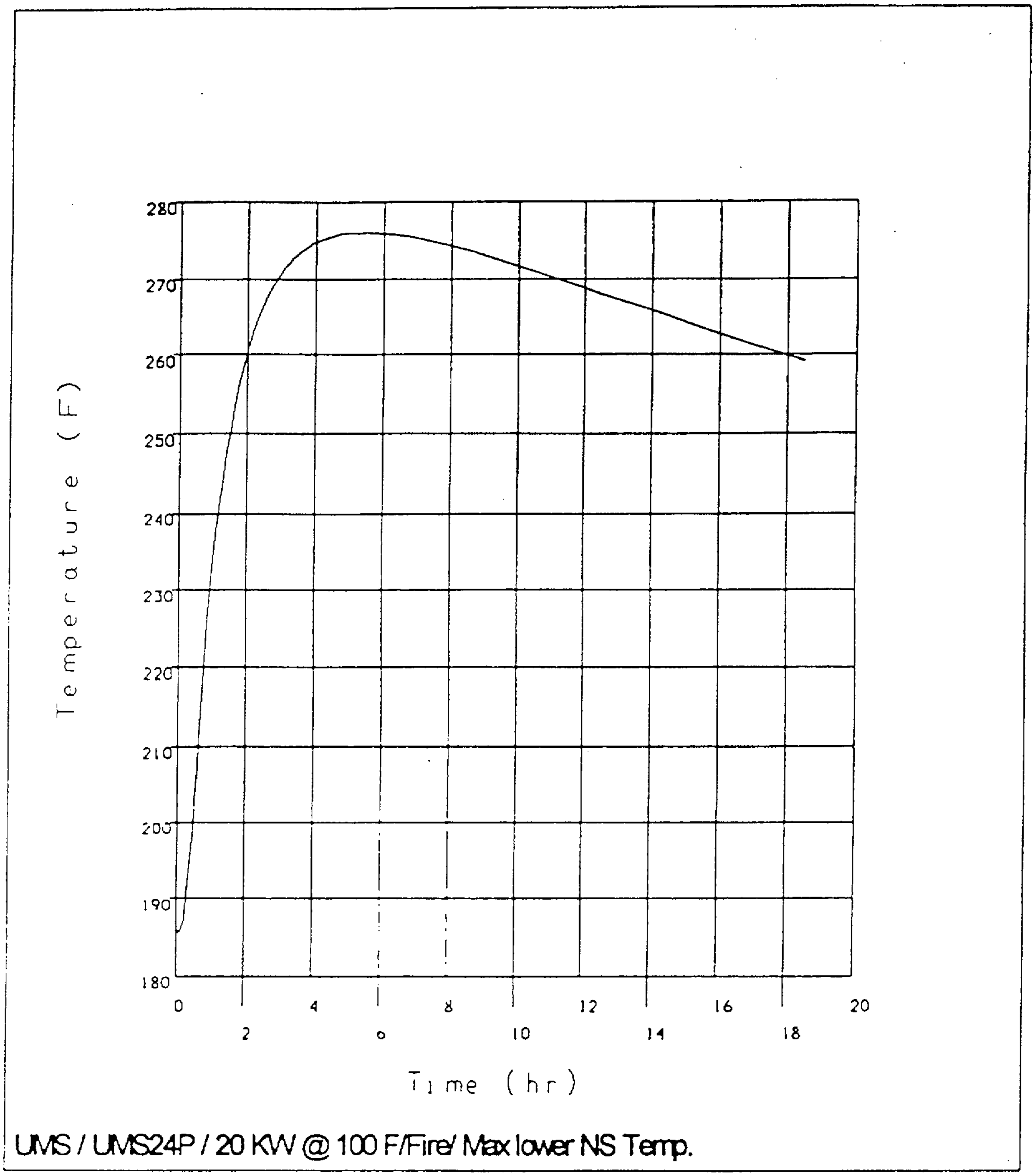
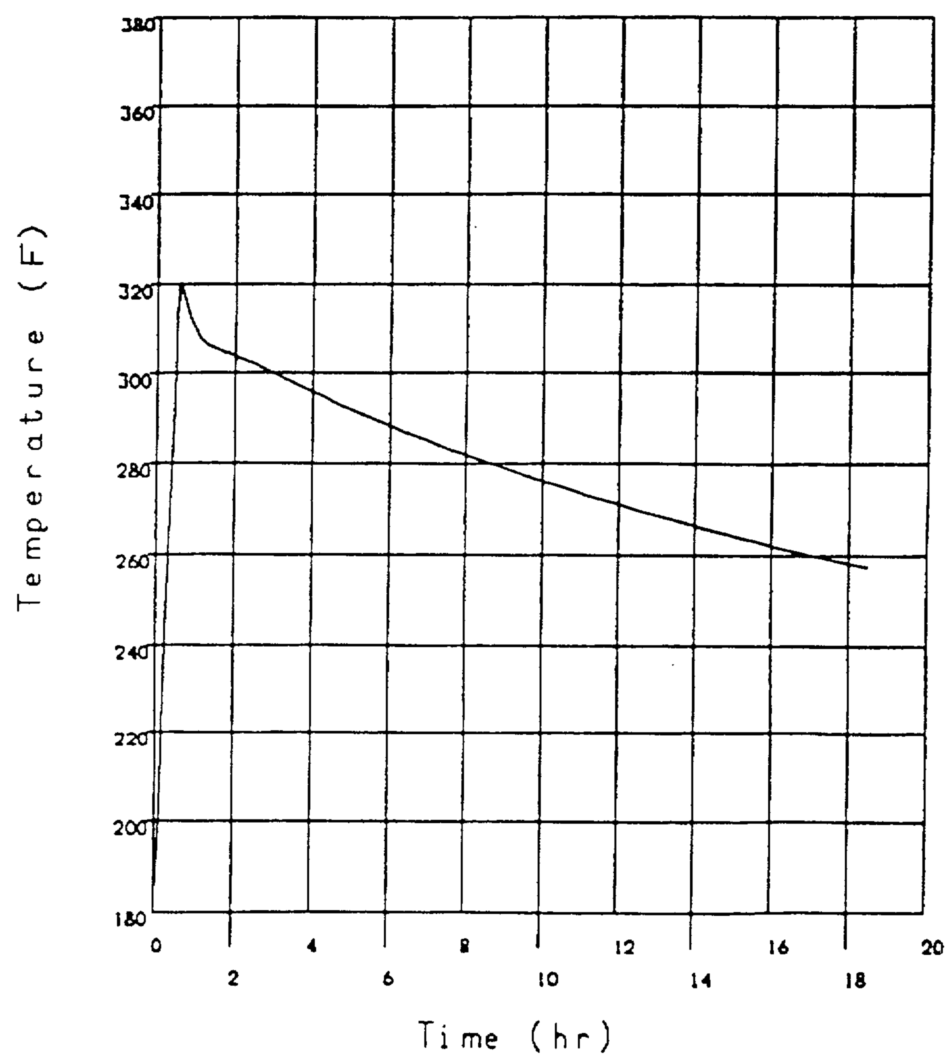


Figure 3.5-9 Hypothetical Accident Conditions Maximum Lower Drain Port O-Ring Temperature History (PWR)



UMS / UMS24P/ 20 KW @ 100 F/Fire/ Lower Drain Port O ring Temperature

Figure 3.5-10 Hypothetical Accident Conditions Maximum Cask Lid Vent Port O-Ring Temperature History (PWR)

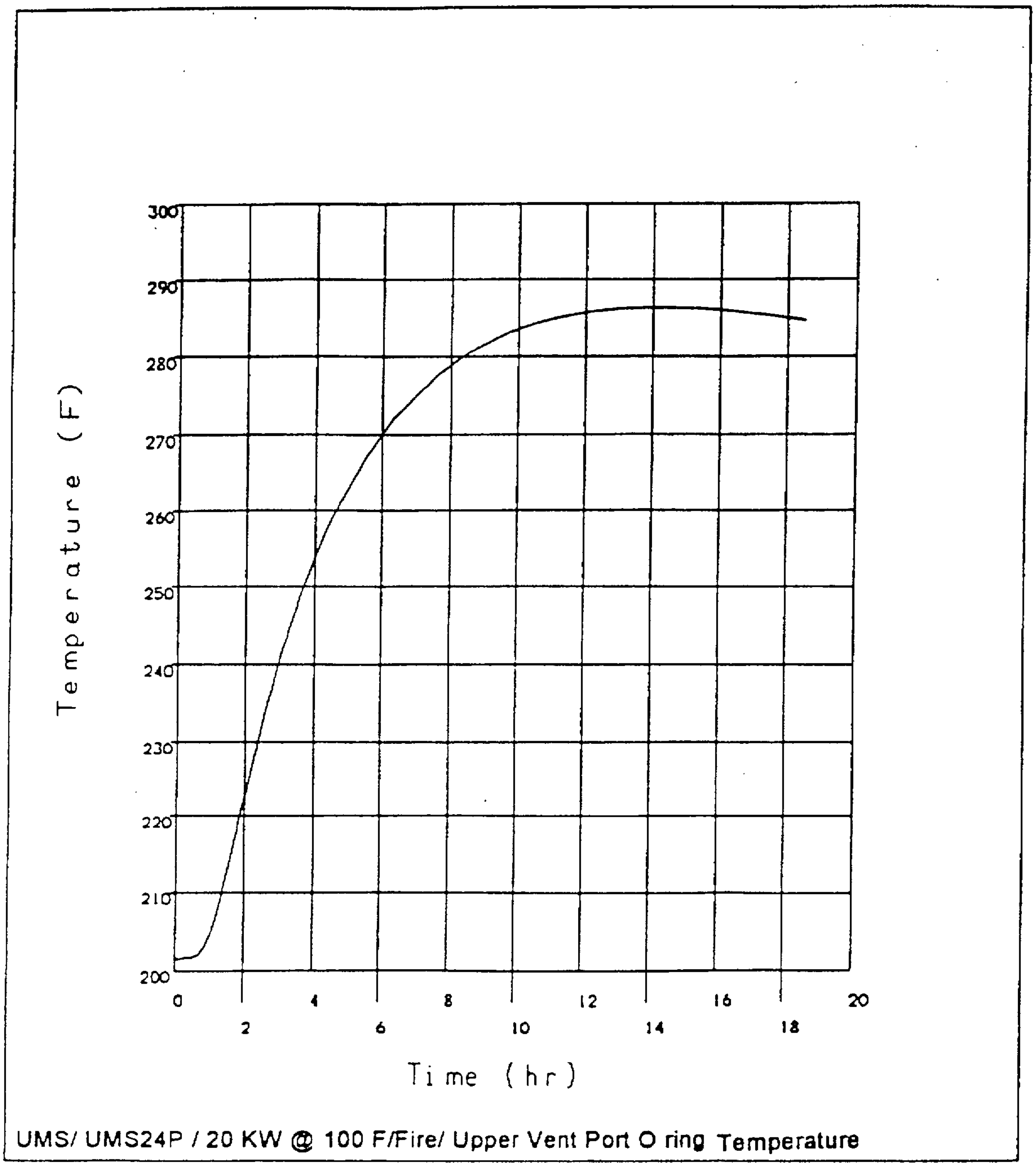
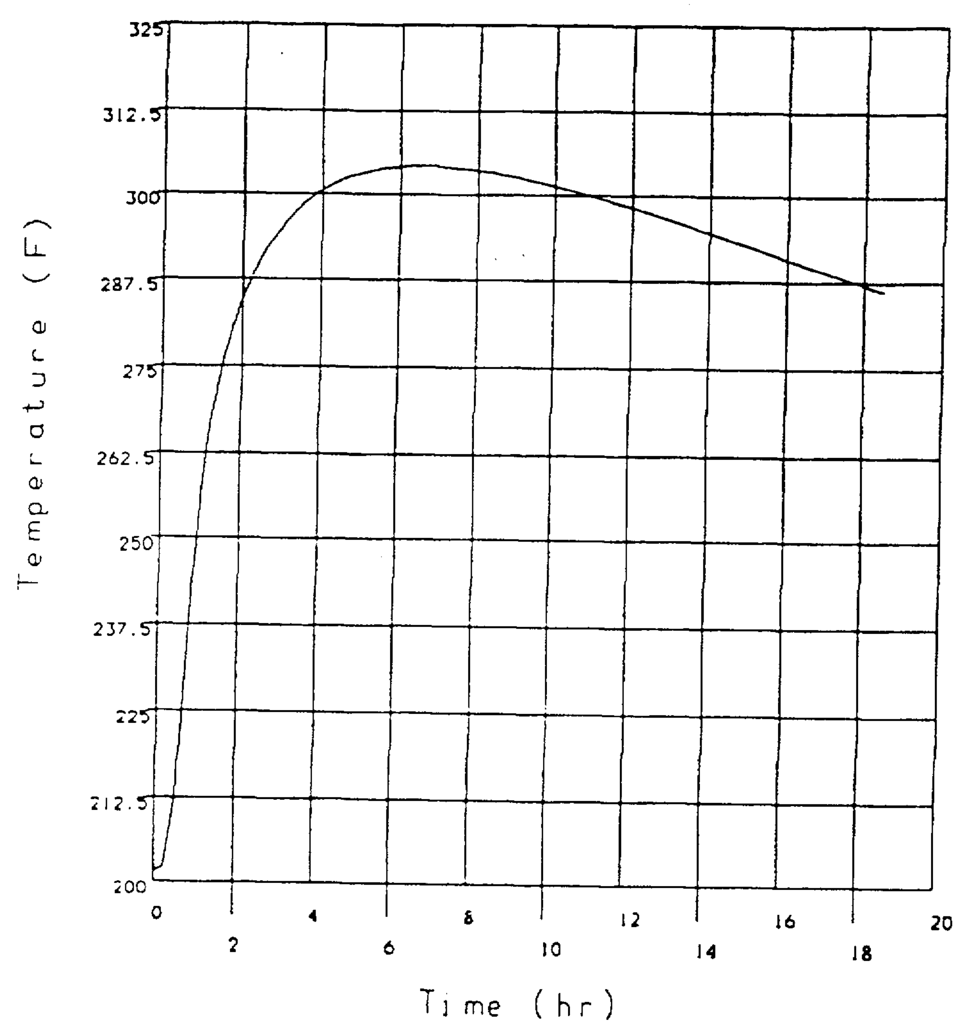
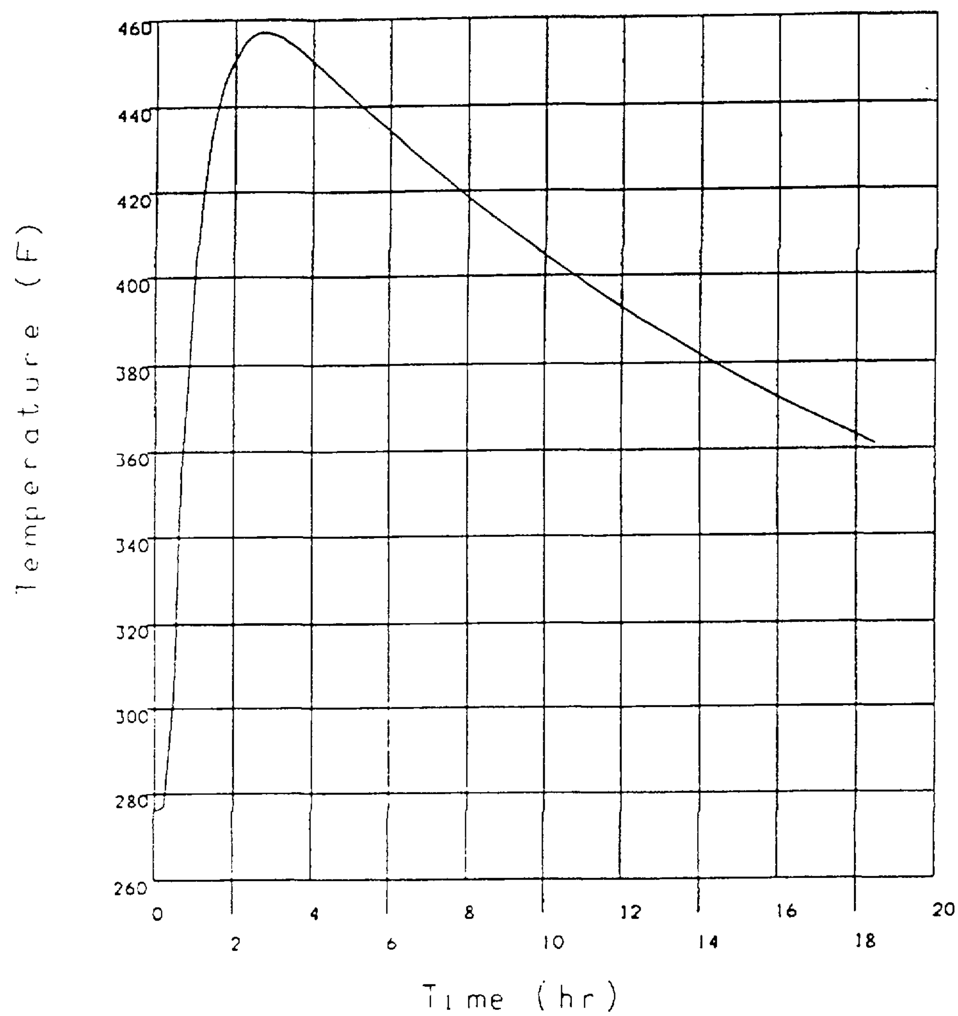


Figure 3.5-11 Hypothetical Accident Conditions Maximum Cask Lid O-Rings Temperature History (PWR)



UMS / UMS24P / 20 KW @ 100 F/Fire/ Cask Lid O ring Temp

Figure 3.5-12 Hypothetical Accident Conditions Maximum Lead Temperature History (BWR)



UMS / BWR / 16 KW @ 100 F/Fire/ Max Lead Temp.

Figure 3.5-13 Hypothetical Accident Conditions Maximum Neutron Shield Exterior Temperature History (BWR)

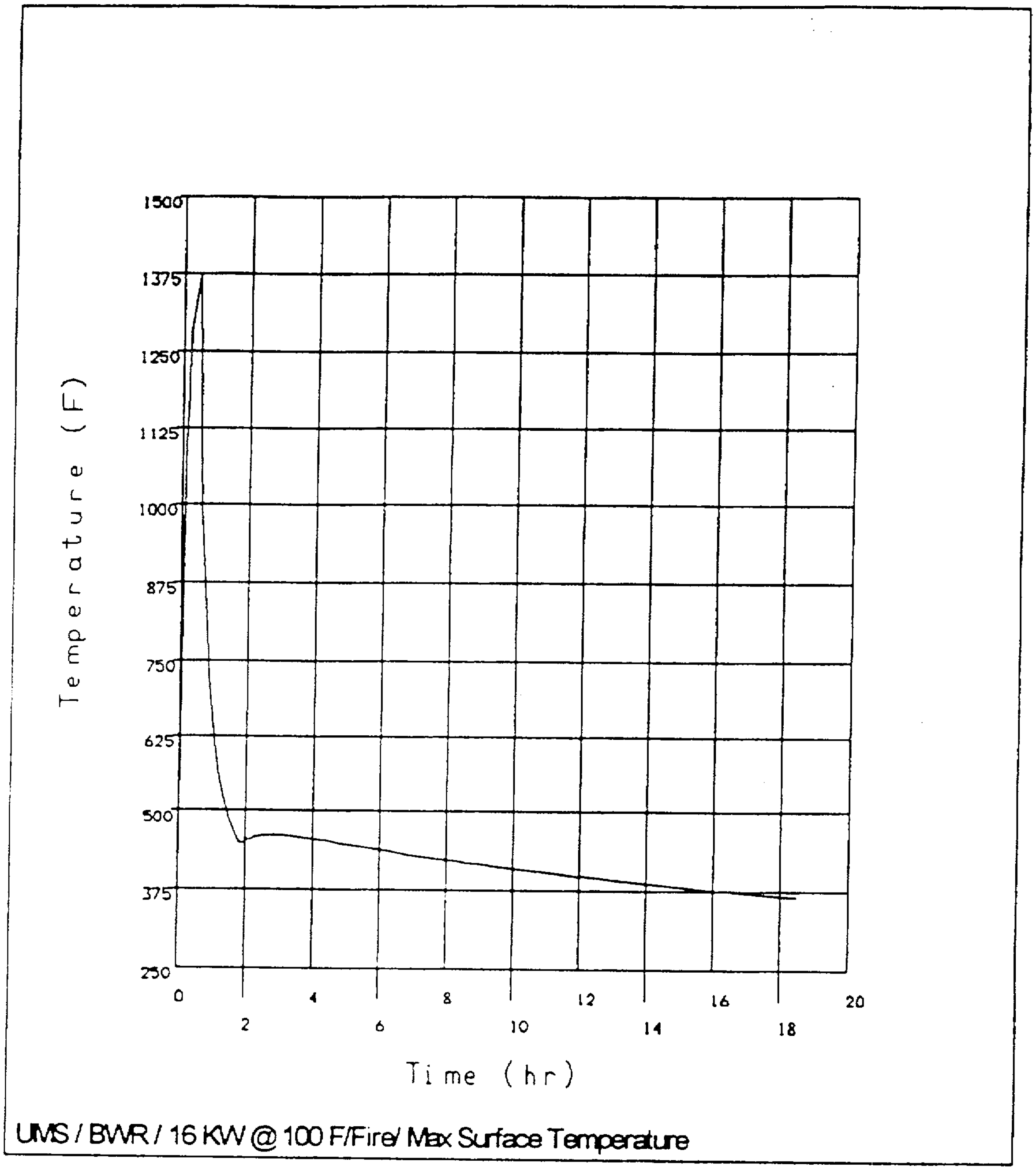


Figure 3.5-14 Hypothetical Accident Conditions Maximum Cask Inner Shell Temperature History (BWR)

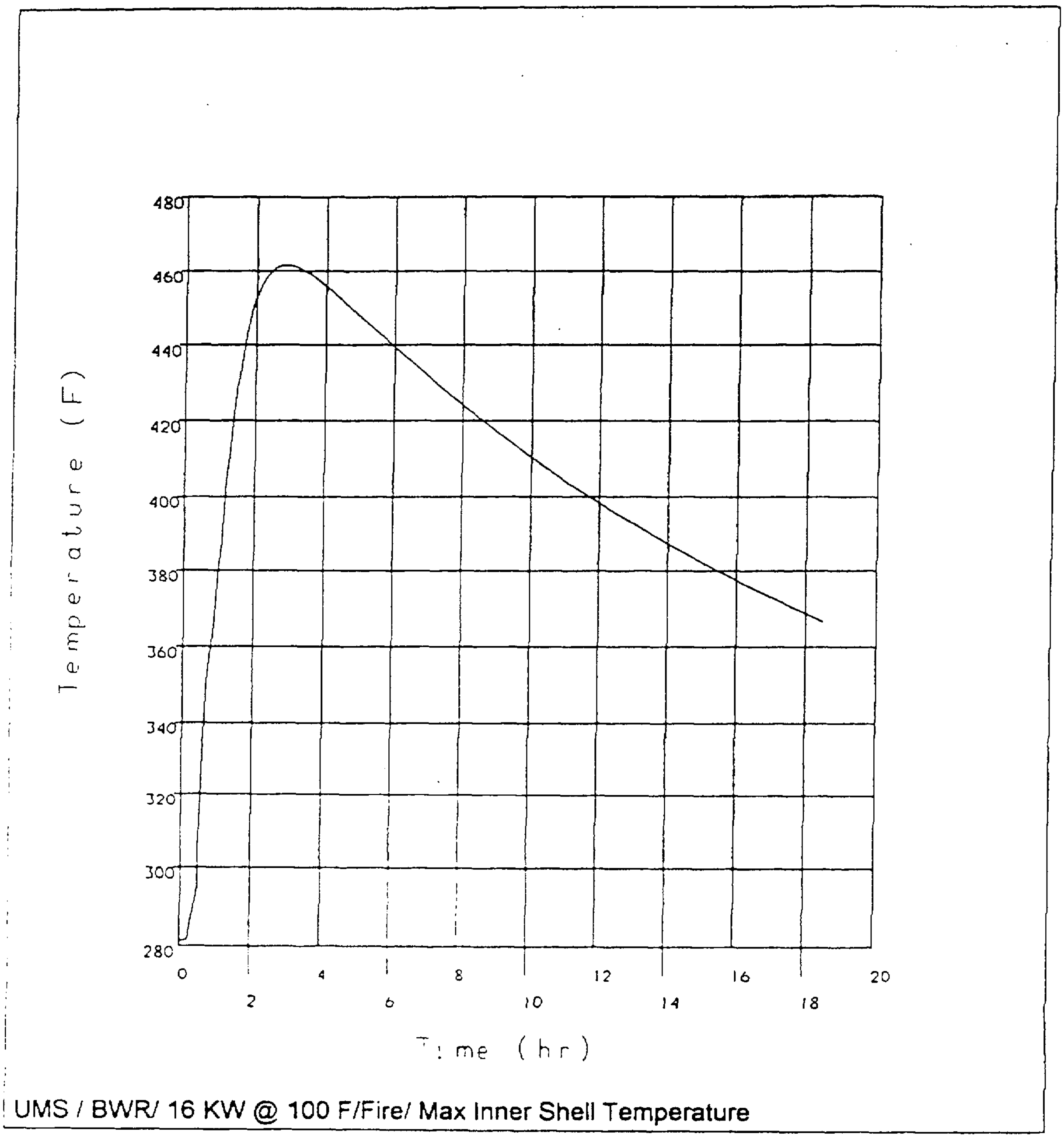
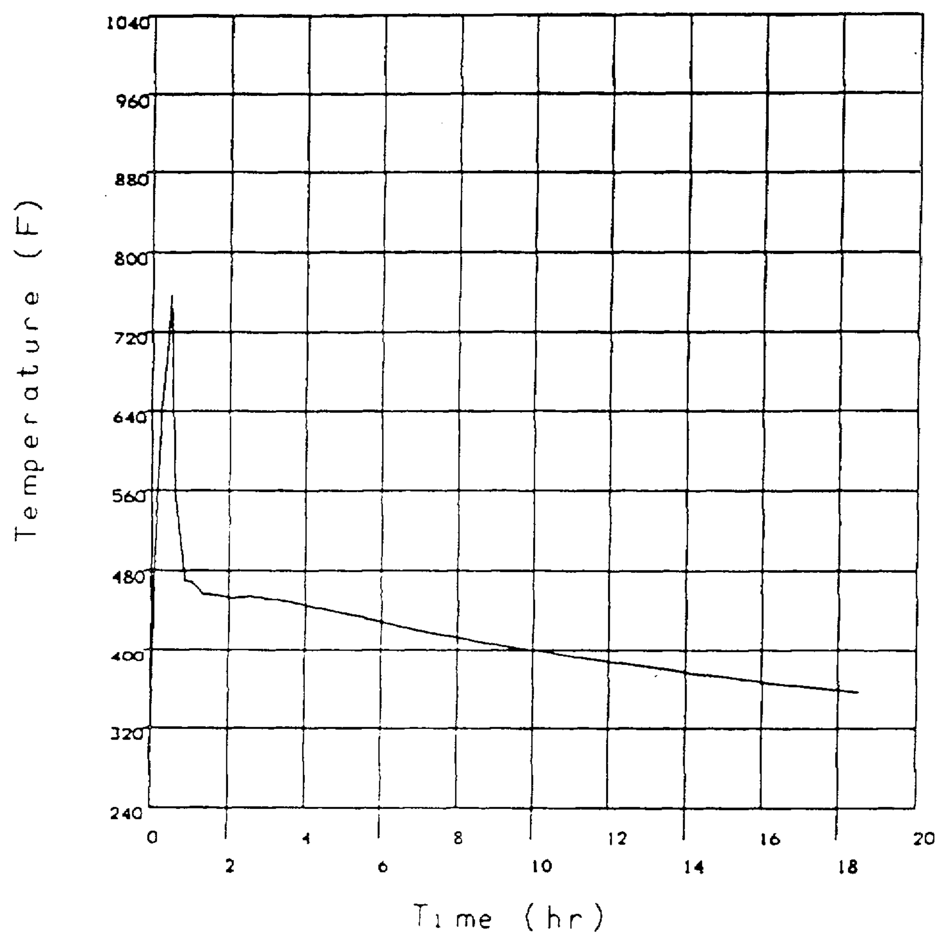


Figure 3.5-15 Hypothetical Accident Conditions Maximum Cask Outer Shell Temperature History (BWR)



UMS / BWR/ 16 KW @ 100 F/Fire/ Max Outer Shell Temperature

Figure 3.5-16 Hypothetical Accident Conditions Maximum Lower Neutron Shield Temperature History (BWR)

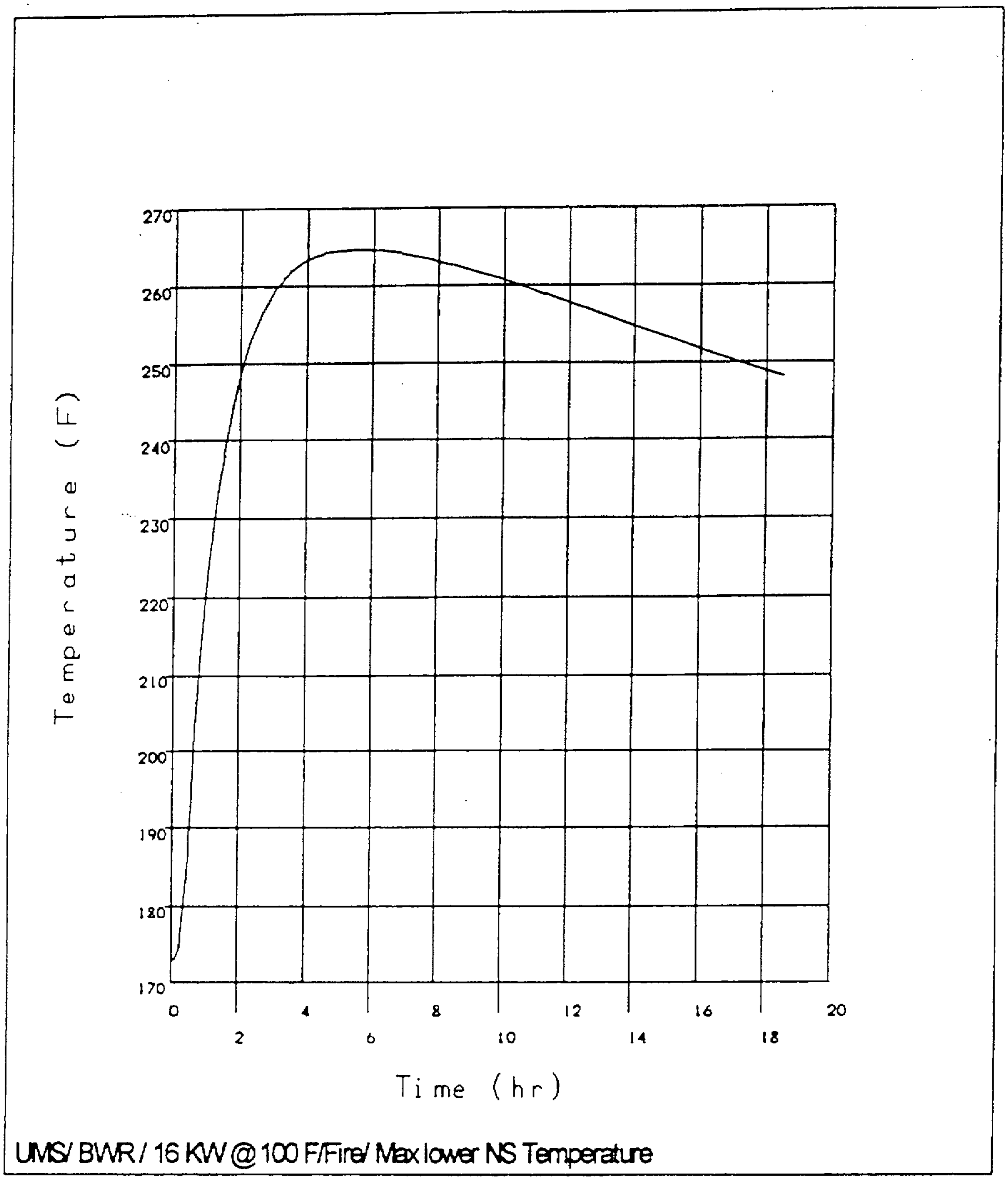


Figure 3.5-17 Hypothetical Accident Conditions Maximum Lower Drain Port O-Ring Temperature History (BWR)

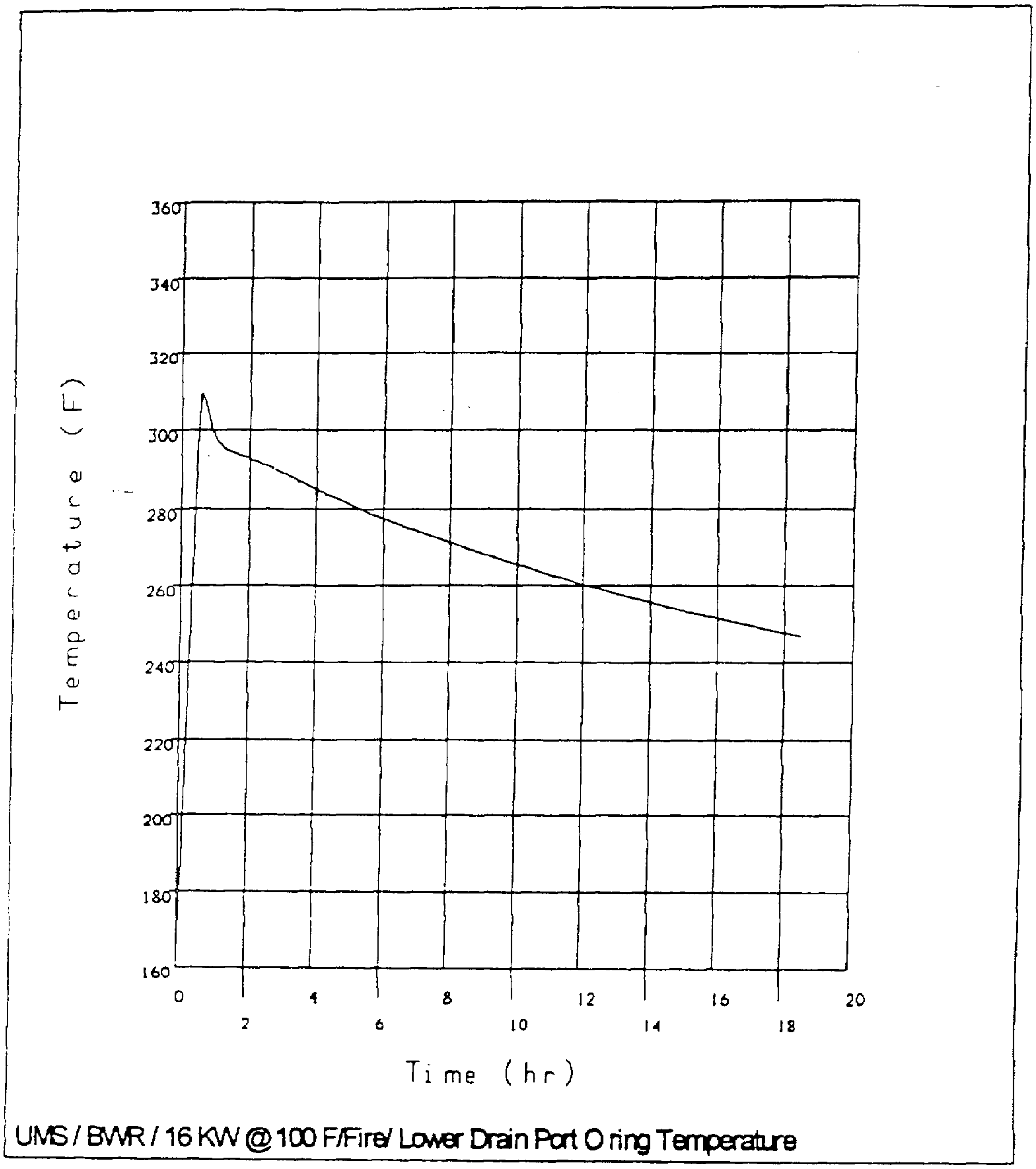


Figure 3.5-18 Hypothetical Accident Conditions Maximum Cask Lid Vent Port O-Ring Temperature History (BWR)

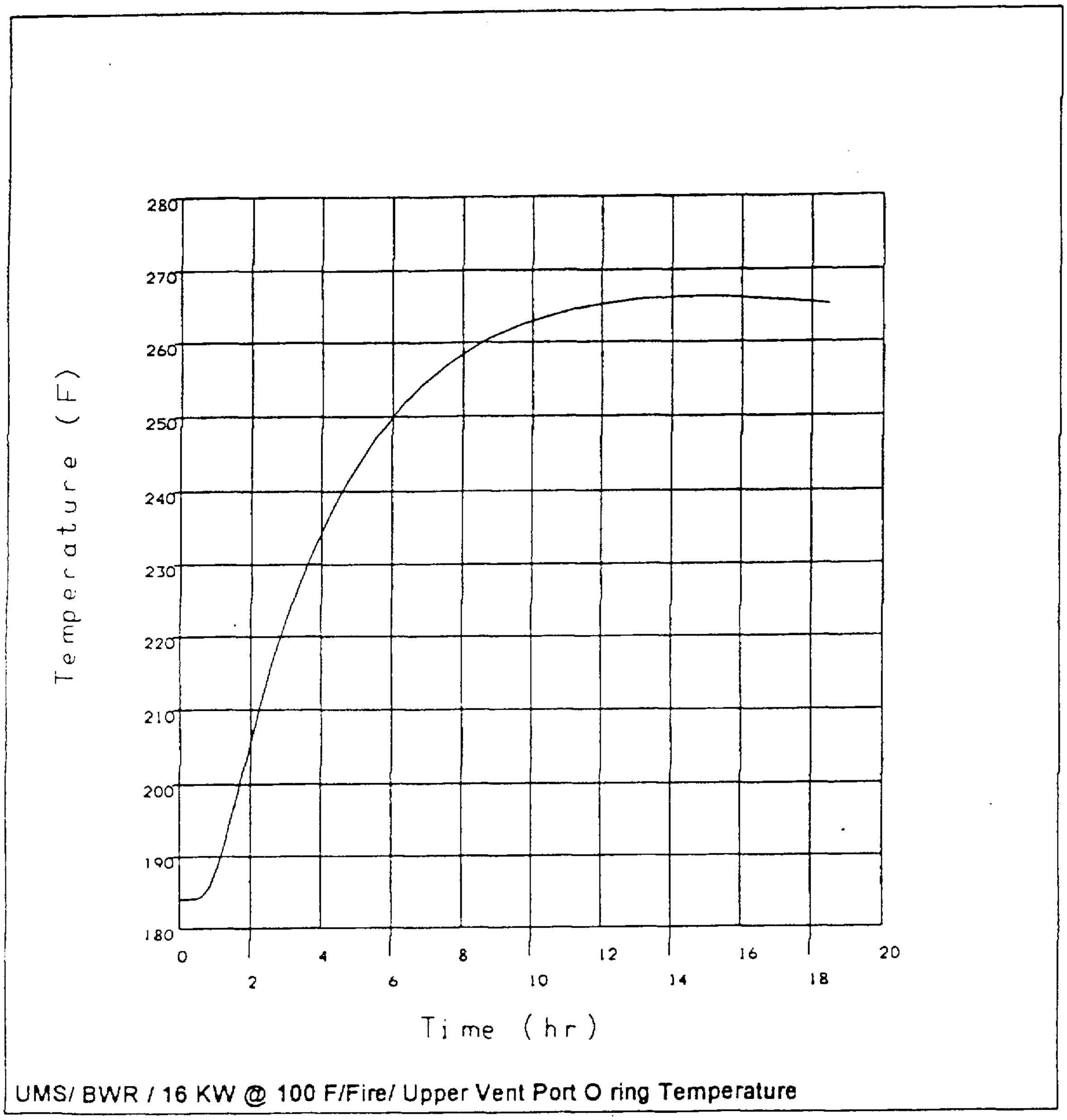
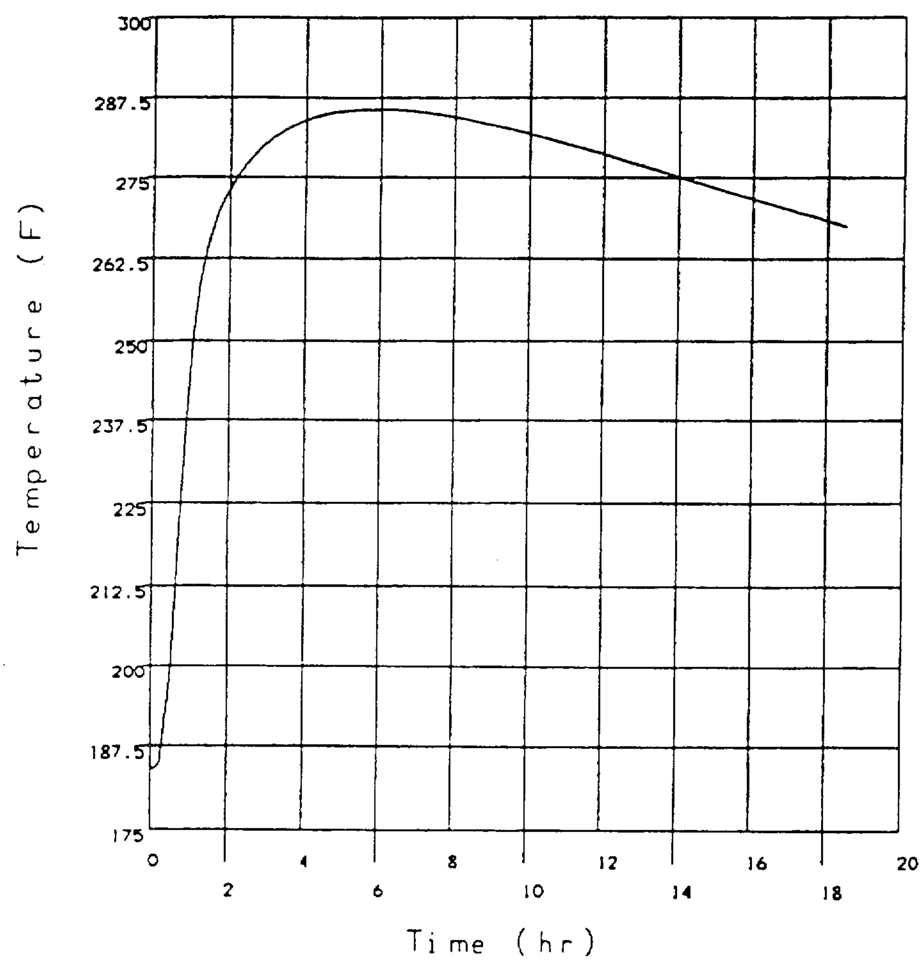


Figure 3.5-19 Hypothetical Accident Conditions Maximum Cask Lid O-Rings Temperature History (BWR)



UMS/ BWR / 16 KW @ 100 F/Fire/ Cask Lid O ring Temperature

Table 3.5-1 Maximum Component Temperatures - Hypothetical Accident Condition Fire Transient (PWR Cask)

Component	Temperature (°F)	Time (Hours)	Temperature Limit (°F)
Cask Lid Bolt[1]	306	5.8	650
Cask Lid O-rings[2]	304	5.5	375[4]
Lower Drain Port O-ring[2]	320	0.8	375[4]
Cask Lid Vent Port O-ring[2]	286	13.0	375[4]
Cask Radial Outer Surface	1,376	0.5	—[5]
Lead Gamma Shield	473	2.9	600
Canister Gas[3]	603	—	—
Maximum Fuel Rod Cladding[3]	919	—	1,058
Cask Inner Shell	479	3.0	—[5]
Aluminum Heat Transfer Disks[3]	794	—	—[5]
Support Disks[3]	797	—	—[5]

Conditions: 30-min, 1475°F Fire
20 kW decay heat

1 Cask lid bolt not explicitly modeled—maximum temperature taken to be the maximum temperature of the cask lid.

2 O-rings not explicitly modeled - maximum temperature taken at the O-ring region in the component of interest.

3 Estimated by adding the maximum temperature gradient between the cask inner shell and component of interest from normal conditions results to the peak temperature of the cask inner shell during the hypothetical accident analysis.

4 Accident temperature limit is 375°F for 10-hr or less durations.

5 These components remain well below their respective material melting temperatures during the hypothetical accident condition fire; therefore, the intended performance of these components is not adversely affected by the hypothetical accident condition fire.

Table 3.5-2 Maximum Component Temperatures - Hypothetical Accident Condition Fire Transient (BWR Cask)

Component	Temperature (°F)	Time (Hours)	Temperature Limit (°F)
Cask Lid Bolt[1]	287	5.8	—
Cask Lid O-rings[4]	286	5.0	375[3]
Lower Drain Port O-ring[4]	309	0.8	375[3]
Cask Lid Vent Port O-ring[4]	266	13.0	375[3]
Cask Radial Outer Surface	1,376	0.5	—[5]
Lead Gamma Shield	457	2.9	600
Canister Gas[2]	558	—	—
Maximum Fuel Rod Cladding[2]	796	—	1,058
Cask Inner Shell	462	3.0	—[5]
Aluminum Heat Transfer Disks[2]	733	—	—[5]
Support Disks[2]	736	—	—[5]

Conditions: 30-min, 1475°F Fire
16 kW decay heat

1 Cask lid bolt not explicitly modeled—maximum temperature taken to be the maximum temperature of the cask lid.

2 Estimated by adding the maximum temperature gradient between the cask inner shell and component of interest from normal conditions results to the peak temperature of the cask inner shell during the hypothetical accident analysis.

3 Accident temperature limit is 375°F for 10-hr or less durations.

4 O-rings are not explicitly modeled - maximum temperature taken at the O-ring region in the component of interest.

5 These components remain well below their respective material melting temperatures during the hypothetical accident condition fire; therefore, the intended performance of these components is not adversely affected by the hypothetical accident condition fire.

Table 3.5-3 Maximum Internal Pressures for Hypothetical Accident Conditions

Fuel	Cavity	Condition	Pressure[1]
PWR	Canister	100% fuel rod failure	5.60 atm ≈ 82.0 psia ≈ 67.3 psig
	Cask[2]	100% fuel rod failure	5.07 atm ≈ 74.56 psia ≈ 59.86 psig
BWR	Canister	100% fuel rod failure	3.72 atm ≈ 54.66 psia ≈ 39.96 psig
	Cask[2]	100% fuel rod failure	3.67 atm ≈ 53.90 psia ≈ 39.20 psig

1. The pressure calculation considers the fire accident condition maximum cavity temperature.
2. The cask cavity pressure assumes failure of the canister confinement boundary.

THIS PAGE INTENTIONALLY LEFT BLANK

3.6 Thermal Evaluation for Site Specific Contents

3.6.1 Maine Yankee Site Specific Contents

The standard spent fuel assembly for the Maine Yankee site is the Combustion Engineering (CE) 14x14 fuel assembly. Fuel of the same design has also been supplied by Westinghouse and by Exxon. The standard 14x14 fuel assembly is included in the population of the design basis PWR fuel assemblies for the UMS Transport System (See Table 1.2.5). The maximum decay heat for the Maine Yankee fuel is limited to the design basis heat load for the PWR fuels (20 kW total, or 0.83 kW per assembly). This heat load is bounded by the thermal evaluations in Sections 3.4 and 3.5 for the normal conditions of transport and hypothetical accident conditions, respectively.

The Maine Yankee site specific fuels and GTCC waste are described in Sections 1.3.1.1.1 and 1.3.1.1.2, respectively.

The thermal evaluations of the Maine Yankee site specific fuels and the GTCC waste are provided in Sections 3.6.1.1 and 3.6.1.2, respectively.

3.6.1.1 Spent Fuel

The Maine Yankee site specific fuels included in this evaluation are:

1. Consolidated fuel rod lattices consisting of a 17x17 lattice fabricated with 17x17 grids, 4 stainless steel support rods and stainless steel end fittings. One of these lattices contains 283 fuel rods and 2 vacancies. The other contains 172 fuel rods, with the remaining locations either empty or containing stainless steel dummy rods.
2. Standard fuel assemblies with a Control Element Assembly (CEA) inserted in each one.
3. Standard fuel assemblies that have been repaired by removing damaged fuel rods and replacing them with stainless steel dummy rods, solid zirconium rods, or 1.95 wt % enriched fuel rods.
4. Standard fuel assemblies that have had the burnable poison rods removed and replaced with hollow Zircaloy tubes.

5. Standard fuel assemblies with in-core instrument thimble assemblies stored in the center guide tube.
6. Standard fuel assemblies that are designed with variable enrichment (radial) and axial blankets.
7. Standard fuel assemblies that have fuel rods removed.

The thermal evaluations of these site specific fuels are provided below. The maximum heat load per assembly is limited to the design basis heat load (0.83 kW) for all Maine Yankee site specific fuels.

1. Consolidated Fuel

There are two (2) consolidated fuel lattices (pseudo assemblies). The maximum decay heat of each consolidated fuel assembly is 0.279 kW. The heat load of the consolidated fuel lattice with 283 fuel pins is bounded by the design basis PWR fuel assembly, since its heat load is only one-third (0.279/0.83) of the design basis heat load.

The second consolidated fuel lattice has 172 fuel rods with 76 stainless steel dummy rods at the outer periphery of the lattice. Due to the presence of the stainless steel rods, the effective thermal conductivities of this assembly may be slightly lower than those of the standard CE 14x14 fuel assembly. While the stainless steel rods provide better conductance in the axial direction, the radiation heat transfer is less effective at the surface of stainless steel rods, as compared to the standard fuel rods. The radiation is a function of surface emissivity and the emissivity for stainless steel (0.36) is less than one-half of that for Zircaloy (0.75). A parametric study is performed to demonstrate that the thermal performance of the UMS PWR basket loading configuration consisting of 23 standard CE 14x14 fuel assemblies and the consolidated fuel lattice with stainless rods is bounded by that of the configuration consisting of 24 standard CE 14x14 fuel assemblies. Two finite element models are used in the study: a two-dimensional fuel assembly model and a three-dimensional periodic canister internal model.

The two-dimensional model is used to determine the effective thermal conductivities of the consolidated fuel lattice with stainless steel rods. Considering the symmetry of the consolidated

fuel, the finite element model represents a one-quarter section as shown in Figure 3.6.1.1-1. The same methodology used in Section 3.4.1.1.2 for the two-dimensional fuel model for PWR fuel is employed in this model. The model includes the fuel pellets, cladding, helium between the fuel rods, and helium occupying the gap between the fuel pellets and cladding. In addition, the rods at the two outer layers are modeled as solid stainless steel rods to represent the configuration of this consolidated fuel lattice. Modes of heat transfer modeled include conduction and radiation between individual rods for steady-state condition. ANSYS PLANE55 conduction elements and LINK31 radiation elements are used in the model. Radiation elements are defined between rods and from rods to the boundary of the model (inside surface of the fuel tube). The effective conductivity for the fuel is determined by using the same procedure as documented in Section 3.4.1.1.2.

The three-dimensional periodic canister internal model consists of a periodic section of the canister internals. The model contains one support disk with two heat transfer disks (half thickness) on its top and bottom, the fuel assemblies, the fuel tubes and the helium in the canister, as shown in Figure 3.6.1.1-2. The purpose of this model is to compare the maximum fuel cladding temperatures of the following cases:

1) Base Case: All 24 positions loaded with standard CE 14x14 fuel assemblies.
2) Case 2: 23 positions with standard fuel, with one consolidated fuel lattice in position 2.
3) Case 3: 23 positions with standard fuel, with one consolidated fuel lattice in position 3.
4) Case 4: 23 positions with standard fuel, with one consolidated fuel lattice in position 4.
5) Case 5: 23 positions with standard fuel, with one consolidated fuel lattice in position 5.

Positions 2, 3, 4, and 5 are shown in Figure 3.6.1.1-3. Based on symmetry, these locations represent all of the possible locations for consolidated fuel in the basket.

The fuel assemblies and fuel tubes are represented by homogeneous regions with effective thermal conductivities. The effective conductivities for the consolidated fuel are determined by the two-dimensional fuel assembly model discussed above. The effective conductivities for the CE 14x14 fuel assemblies are established based on the model described in Section 3.4.1.1.2. Properties for the fuel tubes are determined by the two-dimensional fuel tube model in Section 3.4.1.1.3. Volumetric heat generation corresponding to the design basis heat load of 0.83 kW per

assembly is applied to the CE14x14 fuel regions in the model. Similarly, a heat generation rate corresponding to 0.279 kW is applied to the consolidated fuel assembly region. The heat conduction in the axial direction is conservatively ignored by assuming that the top and bottom surfaces of the model are adiabatic. A constant temperature of 400°F is applied to the outer surface of the model based on the PWR fuel maximum canister temperature of 399°F for the Transport Cask under normal conditions (Table 3.4-1). Steady state thermal analysis is performed for all five cases and the calculated maximum fuel cladding temperatures in the model are:

	Base Case	Case 2	Case 3	Case 4	Case 5
Maximum Fuel Cladding Temperature (°F)	714	695	698	701	700

The maximum temperatures for Cases 2 through 5 are less than that for the Base Case. Additionally, a sensitivity study analysis was performed to evaluate the effect of the number of stainless steel rods considered in the consolidated fuel assembly thermal model (see Figure 3.6.1.1-1). The thermal model was modified such that only the outer layer of rods are modeled as stainless steel rods with the remainder modeled as fuel rods. This differs from the model presented in Figure 3.6.1.1-1, which considers two layers of rods modeled as stainless steel. All five cases (i.e., "Base Case" and cases 2 through 5) were re-analyzed using the effective thermal conductivities generated by the modified thermal model. The results of this analysis show that the maximum fuel cladding temperature for each of the five cases is within 0.1°F of the results using the original consolidated fuel assembly thermal model. It is concluded that the thermal performance of the configuration consisting of 23 standard CE 14x14 fuel assemblies and one consolidated fuel lattice is bounded by that of the configuration consisting of 24 standard CE 14x14 fuel assemblies. These evaluations show that a consolidated fuel lattice can be located in any basket position, based on heat load. Conservatively, however, the consolidated fuel lattice is limited to loading in one of the four corner locations of the basket.

2. Standard CE 14x14 fuel assemblies with Control Element Assemblies

A Control Element Assembly (CEA) consists of a solid B_4C rod encapsulated in a stainless steel tube. The B_4C material has a conductivity of 1.375 BTU/hr-in-°F. With the CEA inserted into the guide tubes of the CE 14x14 fuel assembly, the effective conductivity in the axial direction of

the fuel assembly is increased because solid material replaces helium in the guide tubes. The change in the effective conductivity in the transverse direction of the fuel assembly is negligible since the CEA is inside of the guide tubes. Note that the total heat load, including the small amount of extra heat generated by the CEA, remains below the design basis heat load. Therefore, the thermal performance of the fuel assemblies with CEAs inserted is bounded by that of the standard fuel assemblies.

3. Standard fuel assemblies that have been repaired by removing damaged fuel rods and replacing them with stainless steel dummy rods, solid zirconium rods, or 1.95 wt % enriched fuel rods.

The maximum number of fuel rods replaced by stainless steel rods is four (4) per assembly, which is about 2% of the total number of rods in each assembly (176). The change in the effective conductivity of the assembly is negligible. Note that the conductivity of the stainless steel is similar to that of Zircaloy and better than that of the UO_2. The resultant increase in effective conductivity of the repaired fuel assembly in the axial direction offsets the decrease in the effective conductivity in the transverse direction (due to slight reduction of radiation heat transfer at the surface of the stainless steel rods). The maximum number of fuel rods replaced by solid Zirconium rods is five (5) per assembly. Since the solid Zirconium rod has a higher conductivity than the fuel pin (UO_2 with Zircaloy clad), the effective conductivity of the repaired fuel assembly is increased. The thermal properties for the enriched fuel rod remain the same as for standard fuel rods, so there is no change in effective conductivity of the fuel assembly results from the use of fuel rods enriched to 1.95 wt % ^{235}U. These rods replace other fuel rods in the assembly after the first or second burnup cycles were completed. Therefore, these replacement fuel rods have been burned a minimum of one cycle less than the remainder of the assembly, producing a proportionally lower per rod heat load. The heat load (on a per rod basis) of the fuel rods in a standard assembly, bounds the heat load of the 1.95 wt % ^{235}U enriched fuel rods. Consequently, the loading of modified fuel assemblies is bounded by the thermal evaluation of the standard fuel assembly.

4. Standard fuel assemblies that have had the burnable poison rods removed and replaced with hollow Zircaloy tubes.

There are 16 locations where rods were removed and replaced with hollow Zircaloy tubes. Since the maximum heat load for these assemblies is 0.552 kW per assembly (two-thirds of the design basis heat load) and the number of hollow Zircaloy rods is only about one-tenth (16/176) of the total number of the fuel rods, the thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

5. Standard fuel assemblies with in-core instrument thimble assemblies stored within the center guide tube of each fuel assembly.

Storing an in-core instrument thimble assembly in the center guide tube of a fuel assembly will slightly increase the axial conductance of the fuel assembly (helium replaced by solid material). Therefore, there is no negative impact on the thermal performance of the fuel assembly with this configuration. The thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

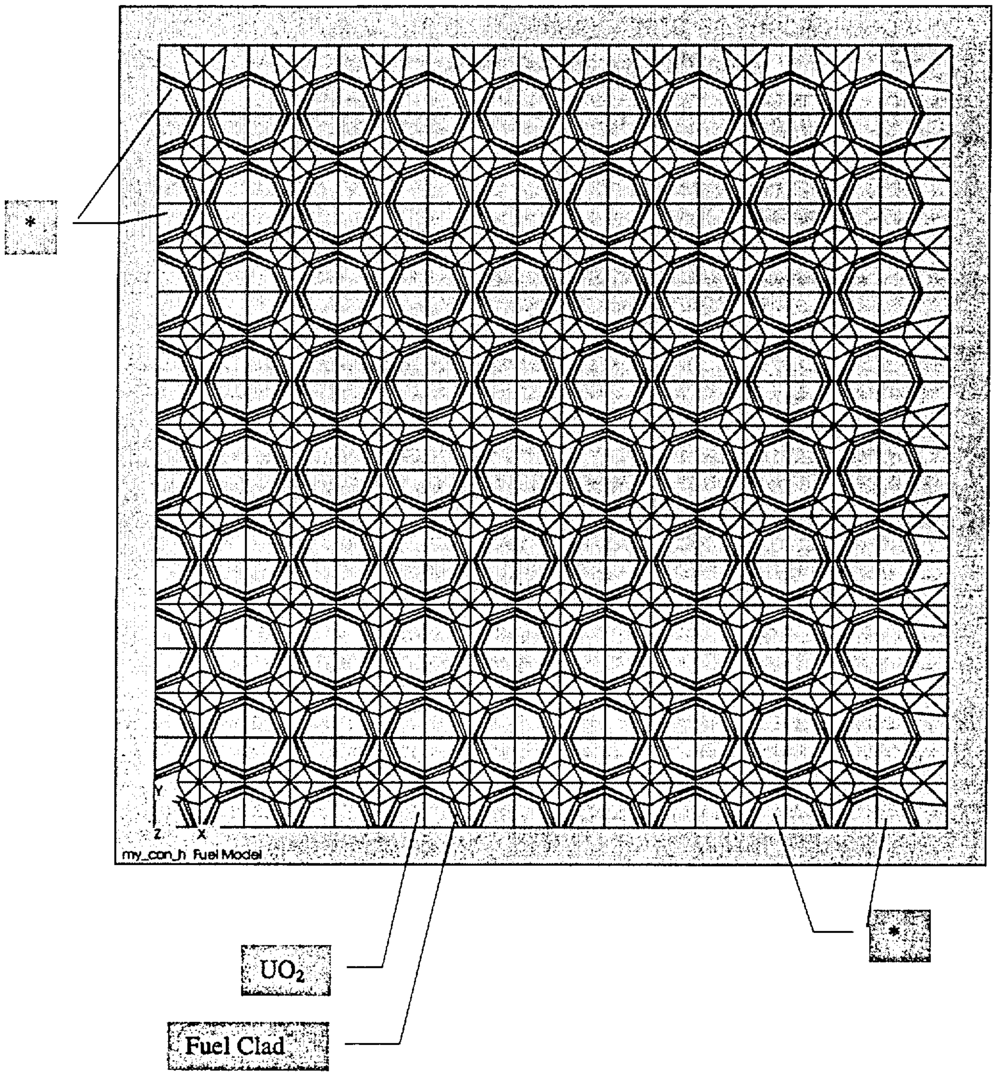
6. Standard fuel assemblies that are designed with variable enrichment (radial) and axial blankets.

The thermal conductivities of the fuel assemblies with variable enrichment and axial blankets are considered to be essentially the same as those of the standard fuel assemblies. Since the heat load per assembly is limited to the design basis heat load, there is no significant effect on the thermal performance of the system due to this loading configuration.

7. Standard fuel assemblies that have fuel rods removed from the lattice.

There is one fuel assembly that has 107 rods removed. This fuel assembly has a heat load of 70 watts (only 8% of the design basis heat load of 0.83 kW). For the rest of fuel assemblies that have fuel rods removed from the lattice, the maximum number of removed fuel rods is 14, which is 8% (14/176) of the total number of rods in one fuel assembly. The maximum heat load for any one of these fuel assemblies is conservatively determined to be 0.63 kW. This heat load is 24% less than the design basis heat load of 0.83 kW. Therefore, the thermal performance for the configuration that contains standard fuel assemblies bounds that of the fuel assemblies with removed rods.

Figure 3.6.1.1-1 Fuel Assembly Quarter Symmetry Model for Maine Yankee Consolidated Fuel



* Two outer layers (rows) of rods are modeled as stainless steel

Figure 3.6.1.1-2 Maine Yankee Three-Dimensional Periodic Canister Internal Model

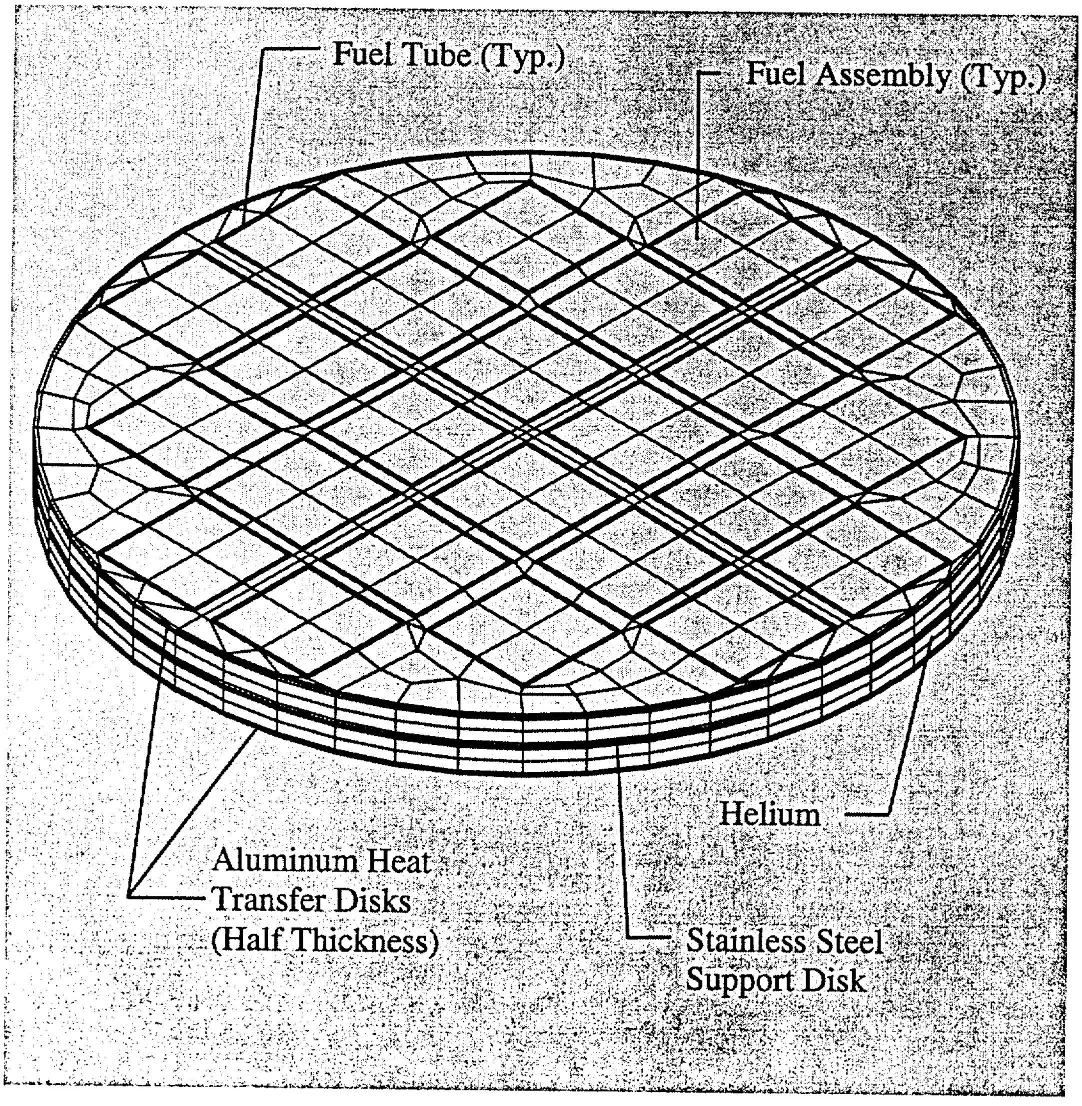
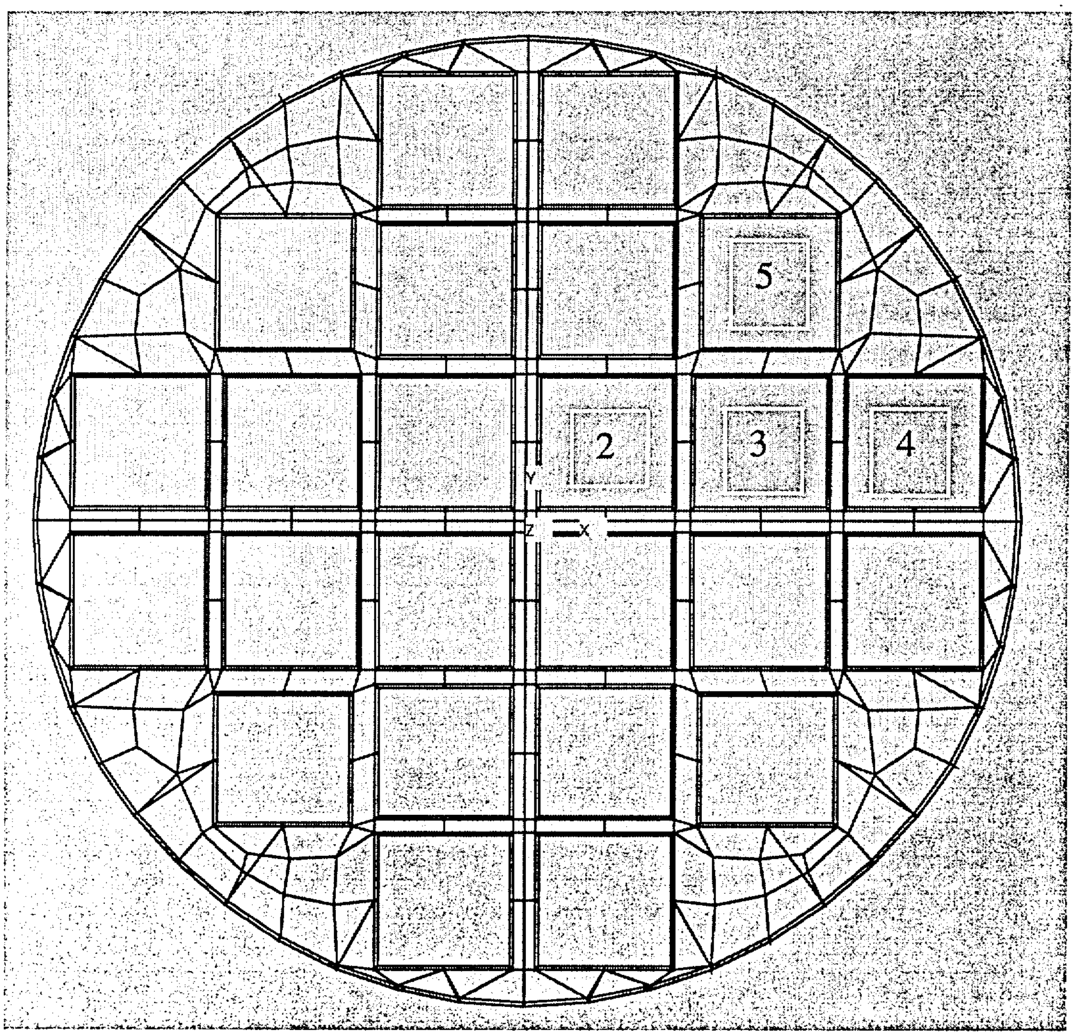


Figure 3.6.1.1-3 Evaluated Locations for a Maine Yankee Consolidated Fuel Lattice in the PWR Fuel Basket



3.6.1.2 Maine Yankee Greater Than Class C Waste

Normal Conditions of Transport

The UMS basket for Maine Yankee Greater Than Class C (GTCC) waste is analyzed for normal conditions of transport thermal conditions using a two-dimensional, axi-symmetric finite element model (see Figure 3.6.1.2-1). The model consists of a section—two support disk periods in height—of the GTCC basket, the canister shell, and the cask body. The model is constructed using PLANE55 two-dimensional thermal solid elements and MATRIX50 radiation superelements.

The normal transport thermal conditions analyzed are:

Case 1:	100°F ambient temperature, solar heat flux
Case 2:	100°F ambient temperature, no solar heat flux
Case 3:	-40°F ambient temperature, no solar heat flux

The GTCC waste contents generate 4.49 kW of decay heat. However, a heat generation rate of 6 kW (5 kW times a peaking factor of 1.2) is conservatively used in the analysis. A heat flux of 0.868 Btu/h-in^2 is applied to the innermost surface of the GTCC basket shield plate (i.e., the left side of the model) to simulate the 6 kW heat generation rate. For Case 1, a solar heat flux of 0.154 Btu/h-in^2 is applied to the outermost surface of the cask body (i.e., the right side of the model).

Heat is transferred through the model via thermal conduction and thermal radiation, and is rejected to the surrounding environment by natural convection (see Section 3.2.3) and thermal radiation. Thermal radiation between the internal surfaces is modeled using the MATRIX50 radiation superelement. The radiating surfaces have emissivities assigned to them based on their respective materials.

Steady-state heat transfer analyses are performed using ANSYS for the boundary conditions corresponding to normal transport conditions (Case 1, Case 2, and Case 3). The results are summarized in Table 3.6.1.2-1.

Accident Conditions

The thermal response of the GTCC waste basket to the hypothetical accident condition fire event is conservatively estimated by combining the temperature results for normal transport conditions with the transport cask inner shell temperature for the accident conditions (design basis PWR contents). Specifically, the temperature difference between the component of interest and the transport cask inner shell, calculated from the temperatures presented in Table 3.6.1.2-1 for Case 2 is added to the maximum temperature of the transport cask inner shell (with PWR contents) for the accident condition (479°F, Table 3.5-1).

The calculated temperatures of the GTCC waste basket/canister/transport cask for accident conditions are summarized in Table 3.6.1.2-2.

Figure 3.6.1.2-1 Two-Dimensional Axi-symmetric Thermal Model of the Maine Yankee GTCC Waste Basket/Canister/Transport Cask

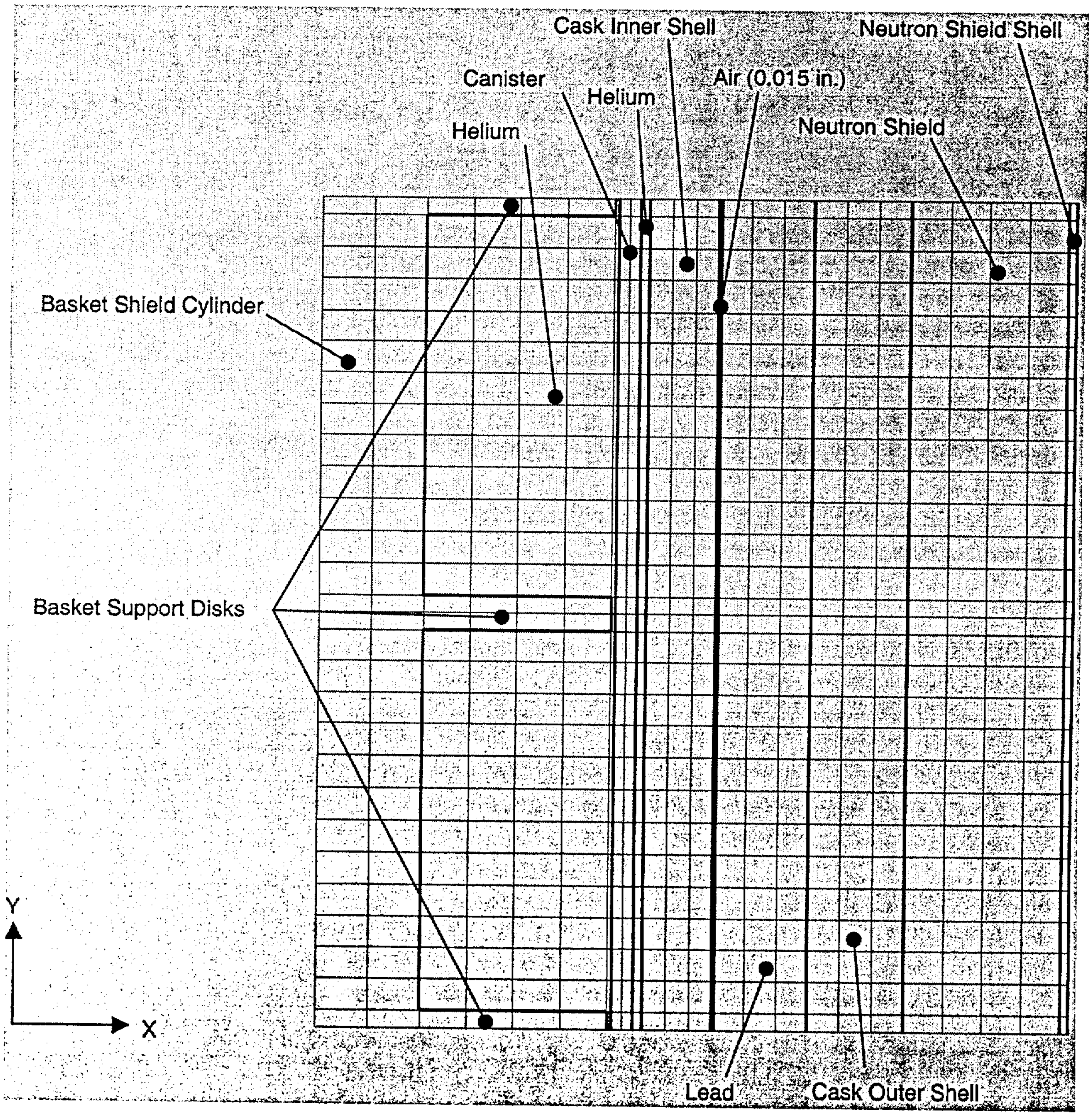


Table 3.6.1.2-1 Summary of the Maine Yankee GTCC Waste Basket/Canister/Transport Cask Maximum Temperatures

Case[1]	GTCC Waste Basket (°F)	Canister Shell (°F)	Transport Cask Inner Shell (°F)
Case 1	287	209	187
Case 2	275	196	173
Case 3	163	58	31

1. Normal Conditions of Transport

Table 3.6.1.2-2 Summary of Maximum Calculated Accident Condition Temperatures for the Maine Yankee GTCC Waste Basket

Component/Region	ΔT – Temperature difference between maximum component temperature and maximum temperature of cask inner shell for Case 2 (Table 3.6.1.2-1) (°F)	Calculated maximum component temperature for accident conditions (°F)
GTCC Waste Basket	102	102 + 479[1] = 581
Canister	23	23 + 479 = 502

1. Table 3.5-1.

THIS PAGE INTENTIONALLY LEFT BLANK

3.7 References

1. Title 10 of the Code of Federal Regulations, Part 71 (10 CFR 71), "Packaging and Transportation of Radioactive Materials," April 1996.

2. IAEA Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Materials," 1985 Edition, as amended 1990.

3. Johnson, A.B., and Gilbert, E.R., "Technical Basis for Storage of Zircaloy-Clad Fuel in Inert Gases," PNL-4835, 1985.

4. ARMCO Product Data Bulletin No. S-22, "17-4PH, Precipitation Hardening Stainless Steel," ARMCO, Inc., 1988

5. ANSYS Revision 5.5, Computer Program, ANSYS, Inc., Houston, PA.

6. NS-4-FR Fire Resistant Neutron and/or Gamma Shielding Material - Product Technical Data, Genden Engineering Services & Construction Company, Tokyo, Japan.

7. MIL-HDBK-5F, "Metallic Materials and Elements for Aerospace Vehicle Structures," U.S. Department of Defense, November 1990.

8.

9. Kreith, F., and Bohn, M.S., Principles of Heat Transfer, 5th Edition, West Publishing Company, 1993.

10. SAND90-2406, Sanders, T.L., et al., "A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements," TTC-1019, UC-820, November 1992.

11. Olander, D.R., "Fundamental Aspects of Nuclear Reactors Fuel Elements," Technical Information Center (U.S. Department of Energy), 1985.

12. Proceedings of the 11th International Conference on the Packaging and Transportation of Radioactive Materials (PATRAM '95), Las Vegas, Nevada, December 3-8, 1995.

13. Kreith, F., Principles of Heat Transfer, 3rd Edition, New York, Intext Educational Publishers.

14. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, SECTION II, PART D - PROPERTIES, 1995 Edition, New York, American Society of Mechanical Engineers, 1995.

15. Bucholz, J.A., Scoping Design Analyses for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, or 10-Year-Old Spent Fuel, Oak Ridge National Library, ORNL/CSD/TM - 149, 1983.

16. Baumeister T. and Mark, L.S., Standard Handbook for Mechanical Engineers, 7th Edition, New York, McGraw-Hill Book Co., 1967.

17. Chapman, A.J. Heat Transfer, 4th Edition, New York, MacMillan Publishing Company, 1987.

18. Edwards, TRUMP, A Computer Program for Transient and Steady State Temperature Distributions in Multimensional Systems, Lawrence Radiation Laboratory, Livermore, Rept, UCLR-14754, Rev. 1 (May 1, 1968).

19. AAR BORAL Sheet Manufacturers Data, Sheet Product Performance Report 624, Brooks & Perkins Advanced Structures Company©, 1983.

20. AAR Standard Specification Sheet for BORAL™ Composite Sheet, Brooks & Perkins Advances Structures Company, BRJREVO-940107.

21. UNIFRAX Product Specifications, Fiberfrax® Ceramic Fiber Paper, C-1423, Unifrax Corporation, Niagara Falls, NY, June 1996.

22. Avallone, E.A., Baumeister, T. III, Marks's Standard Handbook for Mechanical Engineers, (9th Edition), New York, McGraw-Hill Book Company, 1986.

23. Hanford Engineering Development Laboratory, Nuclear Systems Materials Handbook, Volume 1, Design Data, Westinghouse Hanford Company, TID26666.

24. Ross R.B., Metallic Specification Handbook, 4th Edition, London, Chapman and Hall, 1992.

25. Hagrman, D.L., Reymann, G.A., Matpro-Version 11 A Handbook of Material Properties for Use in the Analysis of Light Water Reactor Rod Behavior, Idaho Falls, ID, EG&G Idaho, Inc., 1979.

26. Rust, J.H., Nuclear Power Plant Engineering, Atlanta, GA, S.W., Holland Company, 1979.

27. PNL-6189, Levy, et al., Pacific Northwest Laboratory, "Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircalloy Clad Fuel Rods in Inert Gas," May 1987.

28. PNL-6364, Pacific Northwest Laboratory, "Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere," October 1987.

29. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," March 1972.

30. Pati, S.R., Garde, A.M., Clink, L.J., "Contribution of Pellet Rim Porosity to Low-Temperature Fission Gas Release at Extended Burnups," American Nuclear Society Topical Meeting on LWR Fuel Performance, April 17-20, 1988, Williamsburg, VA.

THIS PAGE INTENTIONALLY LEFT BLANK

Table of Contents

Table of Contents (Continued)

List of Figures

List of Tables

4.0 CONTAINMENT

This chapter presents the Universal Transport Cask containment evaluation. PWR and BWR fuel are evaluated separately.

The B&W 15x15 fuel assembly is the bounding PWR assembly because of its high fuel mass (maximum for PWR assemblies) and fission product inventory. It is therefore used for the PWR fuel allowable leak rate calculations. The analysis conservatively uses a fuel rod helium prepressurization of 500 psi. The maximum design burnup of 50,000 MWD/MTU, at a minimum 1.9 wt% ^{235}U enrichment with a 5-year cool time, is conservatively used to generate the containment analysis source term. The transport cask PWR containment analysis assumes the cask holds 24 design basis PWR fuel assemblies. The PWR fuel leak rate analysis assumes the presence of control components. These components reduce the amount of cask free volume, thereby resulting in a bounding condition containment evaluation. The consolidated fuel evaluation is based on Combustion Engineering 14x14 fuel assembly fuel rods in a 17x17 array configuration. The B&W fuel assembly remains bounding for containment when the number of consolidated assemblies is assumed to be no more than four in each canister. This assumption is conservative, since only one consolidated fuel assembly may be loaded in any canister.

The GE 9x9 (with 79 fuel rods) fuel assembly is the bounding BWR assembly and is, therefore, used in the leakage calculations for the cask transporting BWR fuel. As for the PWR fuel, a maximum design burnup of 50,000 MWD/MTU, at a minimum 1.9 wt% ^{235}U enrichment with a 5-year cool time, is also conservatively used to generate the containment analysis source term for the BWR fuel. The GE 9x9 assembly contains the highest fuel load and surface area of any BWR assembly analyzed. The crud concentration on the larger surface area plays a significant role in BWR fuel assembly calculation of allowable release rate.

Activities for the fuel assemblies are determined by detailed SAS2H isotopic depletion calculations. The depletion calculation shows that enrichment and activity are inversely correlated. The lower enrichment, 1.9 wt% ^{235}U for both the PWR and BWR analysis (with a burnup of 50,000 MWD/MTU), therefore produces bounding activities for the containment analysis.

The free volume employed in the pressure and release calculation is conservatively set to the smallest free cask volume found in the UMS canister classes (Class 1-3 PWR and 4-5 BWR). No credit is taken for the canister as a containment boundary but its mass is considered when

calculating cask free volume. The Universal Transport Cask containment boundary is defined in Section 4.1. Results of package containment analyses for normal conditions of transport and hypothetical accident conditions are provided in Sections 4.2 and 4.3, respectively. These results demonstrate that the package meets the containment requirements of 10 CFR 71.51 [1] and IAEA Safety Series No. 6 (Paragraph 548) [2] for normal conditions of transport and hypothetical accident conditions.

4.1 Containment Boundary

The Universal Transport Cask containment boundary is defined by the following components: (1) inner shell; (2) bottom forging; (3) top forging; (4) cask lid and lid inner EPDM O-ring; (5) vent port coverplate and vent port coverplate inner EPDM O-ring; and (6) drain port coverplate and drain port coverplate inner EPDM O-ring.

There are three possible paths for the escape of radioactive material from the Universal Transport Cask during transport operation. These paths are past the inner EPDM O-ring seals on the lid, on the vent port coverplate and on the drain port coverplate. EPDM O-ring manufacturers data is provided in Section 4.5.2.

The cask containment integrity is verified through leak testing prior to all transport operations. A mass spectrometer leak detector is used to verify that leakage does not exceed the limits established in Section 4.2.3. These limits are in accordance with the requirements of 10 CFR 71.51 and IAEA Safety Series No. 6 (paragraph 548).

4.1.1 Containment Vessel

The primary containment vessel for the Universal Transport Cask consists of a 67.61-in. ID, 2-in.-thick inner shell; a 4.25-in.-thick bottom forging; a 8.825-in.-thick top forging; and a closure lid. The containment vessel components are fabricated from Type 304 stainless steel in accordance with the applicable requirements of the ASME Boiler and Pressure Vessel Code, [3].

4.1.2 Containment Penetrations

The Universal Transport Cask primary containment boundary is described in Section 4.1. The penetrations in the cask primary containment vessel are the vent and drain ports, and the lid. The penetrations are designed to seal the boundary and to ensure that leakage from the cavity does not exceed the established limits. 10 CFR 71.51 establishes release limits under both normal conditions of transport and hypothetical accident conditions . The quick-disconnects installed in the vent and drain openings and in the lid test port are not considered part of the containment boundary. The vent and drain port coverplates are fabricated from SA-240, Type 304 stainless steel.

4.1.3 Seals and Welds

4.1.3.1 Seals

The EPDM O-rings of the lid, vent port coverplate, and drain port coverplate are the seals that provide primary containment, as described in Section 4.1. Section 4.5.2 contains the specifications for the EPDM O-rings. The cask is leak tested before acceptance from the manufacturer and after fuel loading. These tests are described in Table 4.1-1.

4.1.3.1.1 Containment System Fabrication Verification

When fabrication is complete, containment system fabrication verification leak testing is performed on the cask containment as described in Section 8.1.3. This leak test verifies that the leak rate of the assembled containment boundary does not exceed the allowable reference air leak rate of 2.4×10^{-5} ref·cm^3/sec for BWR fuel under normal conditions of transport. As shown in Table 4.2-4, the allowable leak rate for a cask containing BWR fuel bounds the allowable leak rate for the cask containing PWR fuel. The maximum allowable leak rates and the corresponding test sensitivities are evaluated in Section 4.2.3 for normal conditions of transport and in Section 4.3.2 for hypothetical accident conditions. Based on the analysis presented in these sections, the normal conditions allowable leak rate bounds the allowable leak rate for accident conditions.

4.1.3.1.2 Containment System Periodic Verification

The containment system periodic verification is performed on the Universal Transport Cask package containment boundary seals and components , in accordance with the leak test acceptance criteria established for the containment system fabrication verification (Section 8.1.3). This verification leak test conforms to test method A5.4 of Table A1 of ANSI N14.5-1997 [4].

Whenever a containment seal or component is replaced, the O-ring or containment component is leak tested following replacement according to the requirements of the containment system periodic verification (Section 8.1.3). This test verifies that the replacement seal or component has been properly installed and that the leak rate meets the allowable leak rate requirements established for the containment system fabrication verification specified in Section 4.2.3.

4.1.3.1.3 Containment System Verification Prior to Transport

As specified in the loading procedure (Section 7.1.3), the containment system is leak tested in accordance with Section 8.1.3 (helium leak tested), if components of the containment boundary are replaced during loading operations. If containment boundary components are not replaced, then the containment boundary is pressure tested in accordance with Paragraph 7.6.4 of ANSI N14.5-1997 to demonstrate containment boundary assembly in accordance with the operating procedures.

For unloaded (empty) transport, pressure testing is used to demonstrate containment boundary assembly.

The assembly pressure test configuration conforms to test method A5.1 of Table A1 of ANSI N14.5-1997.

4.1.3.2 Welds

Circumferential and longitudinal welds are used to fabricate the inner shell and to attach it to the top and bottom forgings. The longitudinal welds in the cylindrical sections are staggered circumferentially by 90° or 180°. Containment vessel welds are full penetration bevel or groove welds to ensure structural integrity. Upon completion of the inner shell welds, the welds are radiograph-inspected and accepted in accordance with ASME Code Section III NB-5320.

Upon completion of containment vessel fabrication, the cask containment boundary is hydrostatically tested in accordance with ASME Code requirements to ensure the integrity of the welds and containment components (Section 8.1.2.3). Following hydrostatic testing, all containment vessel welds are visually inspected by the dye penetrant examination method and evaluated in accordance with ASME Code requirements. Following fabrication, the containment boundary o-rings are leak tested in accordance with Section 8.1.3. The post-fabrication leak test is based on the bounding BWR fuel allowable leak rate of 2.4×10^{-5} ref cm^3/sec as shown in Table 4.2-4. Test equipment and methods are selected to assure a minimum test sensitivity of one half the reference leak rate, or 1.2×10^{-5} ref cm^3/sec. The equivalent allowable helium leak rate is 3.3×10^{-5} cm^3/sec at standard conditions. The required helium leak test sensitivity is 1.6×10^{-5} cm^3/sec. The helium leak rate at standard conditions is specified since the test is conducted

using helium as the detector gas. Specification of standard conditions is conservative, since the actual pressure may be higher than the postulated 0 psig (1 atmosphere) pressure that is assumed.

4.1.4 Closure

The primary closure assembly for the Universal Transport Cask consists of the lid, bolts, and o-rings. The lid is recessed and bolted into the top forging of the cask body. The 6.5-in. thick, 78.17-in. diameter lid is made of ASME SA-336, Type 304 stainless steel. The lid is retained by 48 bolts that are 2-8 UN socket head cap screws fabricated from SB-637, Grade N07718 nickel alloy steel bolting material. The initial torque for installation of the lid bolts is as specified in Table 7-1. The bottom surface of the lid is sealed to the top forging of the cask body by a set of EPDM o-rings, with the inner o-ring forming the containment boundary. The second (outer) o-ring provides an annulus to test the inner o-ring seal.

The vent port is recessed into the lid and the drain port is recessed into the bottom forging. The vent and drain port coverplates are secured by four 1/2-13 UNC bolts fabricated from SA-193, Grade B6, Type 410 stainless steel.

Similar to the inner lid configuration, each of the vent and the drain coverplates is sealed by a set of o-rings, with the inner o-ring forming the containment boundary. The second (outer) o-ring provides an annulus to test the inner o-ring seal.

Table 4.1-1 Containment Verification Leak Test Requirements and Schedule

	Post-Fabrication and Annual Maintenance[1]	**Loaded Transport (O-Ring Replacement)**[2]	**Loaded Transport (No O-Ring Replacement)**[2]	**Empty Transport**[2]
Allowable Reference Leak Rate[3]	2.4×10^{-5} cm^3/sec	2.4×10^{-5} cm^3/sec	1×10^{-3} cm^3/sec	1×10^{-3} cm^3/sec
Allowable Helium Leak Rate[3]	3.3×10^{-5} cm^3/sec	3.3×10^{-5} cm^3/sec	--	--

1 All o-rings are replaced during Annual Maintenance.

2 The need for o-ring replacement is determined by inspection or by leak test results. Only the appropriate set of o-ring is replaced as necessary prior to transport.

3 The allowable leak rate is based on the bounding GE 9x9 BWR fuel assembly and is the same for the transport of GTCC waste.

THIS PAGE INTENTIONALLY LEFT BLANK

4.2 Containment Requirements for Normal Conditions of Transport

The Universal Transport Cask must maintain a radioactivity release rate of not more than 10^{-6} A_2/hr under normal conditions of transport, as required by 10 CFR 71.51 and IAEA Safety Series No. 6 (paragraph 548). For the cask containing PWR fuel, this condition is satisfied by maintaining a maximum reference (air at standard conditions) leak rate of 5.0×10^{-5} ref·cm^3/sec, or 6.7×10^{-5} cm^3/sec helium at the test condition, which is conservatively considered to be the standard conditions. For the cask containing BWR fuel, the radioactivity release rate requirement is satisfied by maintaining a maximum reference leak rate from the cask of 2.4×10^{-5} ref·cm^3/sec, or 3.3×10^{-5} cm^3/sec helium at the test conditions. Consequently, the BWR allowable leak rate is conservatively applied as the containment boundary test condition for post fabrication testing, annual testing, and when containment components are replaced during cask use. Calculations of these limits are provided in this section.

The structural and thermal evaluations of the Universal Transport Cask are provided in Chapters 2.0 and 3.0, respectively. Results of these evaluations also demonstrate that cask containment is maintained during normal conditions of transport. Therefore, the package satisfies the containment requirements of 10 CFR 71.71.

4.2.1 Containment of Radioactive Material

The 10 CFR 71 limit for the release of radioactive material under normal conditions of transport is 10^{-6} A_2/hr. In this analysis, A_2 for a mixed gas is determined by using the method described in 10 CFR 71, Appendix A. The release fractions for the various radionuclides transported in the Universal Transport Cask are obtained from NUREG/CR-6487 [4] and summarized in Table 4.2-1 (located at the end of this section). The curie content per isotope for 5-year cooled PWR and BWR design basis fuel assemblies is provided in Section 4.5-3.

In addition to the radionuclides produced by the fuel material, fuel assemblies develop a coating of impurities deposited by cooling water during power generation. This coating is known as crud. Crud contains mostly nonradioactive elements but also contains a significant amount of ^{60}Co. NUREG/CR-6487 lists the maximum ^{60}Co concentrations on spent fuel assemblies to be 140 μCi/cm^2 for PWR assemblies and 1,254 μCi/cm^2 for BWR assemblies at initial discharge. The surface areas of the design basis PWR and BWR assemblies (B&W 15x15 and GE 9x9) are

calculated to be 3.25×10^5 cm^2 and 1.77×10^5 cm^2, respectively. The total PWR crud activity, calculated by the conservative surface area and maximum activity density, envelopes assemblies with control components inserted. Fuel assembly characteristics are listed in Tables 1.2-4 and 1.2-5 for PWR and BWR fuel assemblies respectively.

4.2.1.1 Calculation of Allowable Leak Rates

The maximum permissible leak rate from the cask under normal conditions of transport is determined from the 10 CFR 71 limit of 10^{-6} A2/hr.

$$R_N = L_N C_N \le A_2 \text{ x } \cdot 1 \text{x} 10^{-6} \text{hr}^{-1} \text{ or}$$
$$R_N = L_N C_N \le A_2 \text{ x } \cdot 2.78 \text{x} 10^{-10} \sec^{-1}$$

where:

L_N = is the volumetric gas leakage rate [cm^3/s]

C_N = is the curies per unit volume (termed "activity density") of the radioactive material that passes through the leak path [Ci/cm^3]

R_N = Release rate for normal transport conditions [Ci/sec]

Activity Density of Radioactive Material (C_N)

The total inventory of fission product gases, volatiles fines and crud are shown in Table 4.5.3-1 through Table 4.5.3-6 . These inventories are calculated by using the source terms produced by the SAS2H [6] sequence , the release fractions and the postulated crud (^{60}Co) . The ^{60}Co content is decayed 5 years from discharge to the design basis fuel cool time. The PWR analysis is based on 24 design basis fuel assemblies. The BWR analysis is based on 56 design basis fuel assemblies.

$$C_n = C_{Crud} + C_{Volaties} + C_{FissionGas} + C_{Fines}$$

$$C_{Crud} = \frac{f_C M_T}{V} = \frac{f_C S_C N_A (N_R S_{AR} + S_{Ch})}{V}$$

where:

C_{crud} = activity density inside containment vessel resulting from crud spallation [Ci/cm^3]

M_T	=	total crud activity inventory [Ci]
f_c	=	crud spallation factor
V	=	free volume inside containment vessel [cm_3]
S_C	=	crud surface activity [Ci/cm^2]
N_R	=	number of fuel rods per assembly
N_A	=	number of assemblies
S_{AR}	=	surface area per rod [cm^2]
S_{Ch}	=	channel surface area [cm^2] (BWR fuel only).

and,

$$C_{fines} = \frac{f_F W_R A_R N_R N_A f_B}{V}$$

where:

C_{fine}	=	activity concentration inside containment vessel resulting from fines released from cladding breaches [Ci/cm^3]
f_F	=	fraction of fuel rod's mass released as fines resulting from cladding breach
f_B	=	fraction of fuel rods that develop cladding breach
W_R	=	mass of the fuel in fuel rod [g]
N_R	=	number of fuel rods per assembly
N_A	=	number of assemblies
A_R	=	specific activity of fines emitted from cladding breach in fuel rod [Ci/g]
V	=	containment vessel void volume [cm^3].

and,

$$C_{vg} = C_{vol} + C_{gas} = \frac{N_R N_A f_B W_R (A_V f_V + A_G f_G)}{V}$$

where:

C_{vg}	=	releasable activity concentration inside the containment vessel resulting from gases and volatiles released from cladding breaches [Ci/cm^3]
C_{vol}	=	releasable activity concentration inside the containment vessel resulting from volatiles released from cladding breaches [Ci/cm^3]
C_{gas}	=	releasable activity concentration inside the containment vessel resulting from gases released from cladding breaches [Ci/cm^3]
W_R	=	mass of the fuel in a fuel rod [g]

N_R	=	number fuel rods per assembly
N_A	=	number of assemblies
f_B	=	fraction of rods that develop cladding breaches
A_V	=	specific activity of volatiles in fuel rod [Ci/g]
f_V	=	fraction of volatiles in fuel rod released if rod develops cladding breach
A_G	=	specific activity of gas in fuel rod [Ci/g]
f_G	=	fraction of gas that would escape from fuel rod that develops cladding breach
V	=	is the void volume inside containment vessel [cm^3].

Activity Values for Radionuclides

A_2 values used in this analysis (based on 10 CFR 71 Appendix A) are listed in Section 4.5.3 for all radionuclides produced by the SAS2H analysis (plus ^{60}Co). The mixture A_2 values are shown in Tables 4.2-2 and 4.2-3 . For those isotopes for which no specific A_2 values are given in 10 CFR 71 Appendix A, the generic values listed in Table A.2 of Appendix A, are applied. A_2 values for mixed isotopes are calculated from the following:

$$A_2 = \frac{1}{\sum \frac{F_i}{A_2^i}}$$

where:

$$F_i = \frac{S_i}{S_n}$$

and

F_i = The fraction of isotope i with respect to the entire mixture
S_i = The activity of isotope i [Ci]
S_n = Total group activity [Ci]

Mixture A_2 values are determined for gas, volatile, fine, and crud mixtures and are then combined for a total cask mixture A_2 value. Tables 4.2-2 and 4.2-3 provide the source term and A_2 values per group for PWR and BWR cask systems release rate calculations.

Maximum Allowable Leak Rates

On the basis of the methodology discussed above, the maximum allowable leak rates for the casks containing design basis PWR and BWR fuel under normal conditions of transport are calculated to be 5.0×10^{-5} and 2.4×10^{-5} ref cm^3/sec, respectively (Table 4.2-4).

The maximum allowable release rates are more restrictive for the cask containing BWR assemblies because of the significantly higher surface contamination (crud) on BWR assemblies. The increased surface contamination more than offsets the higher fuel mass (+3%) and burnup (+12%) of the cask containing PWR assemblies.

4.2.1.2 Correlation of Allowable Leak Rates to Air Standard

The volumetric gas leak rate, L, is independent of transport cask pressure and temperature. The maximum allowable release must be correlated with air standard leak rates, which depend on gas temperatures, pressures, and leakage path length and diameter. This correlation requires calculation of the capillary opening diameter through which the flow occurs. Depending on pressure and condition of the flow, a combination of continuum and molecular flow occurs.

Continuum flow and molecular flow equations are obtained from NUREG/CR-6487, Section 2. Both continuum and molecular flow rate equations presented below are adjusted to upstream flow rate in accordance with NUREG/CR-6487 and ANSI N14.5-1997.

The continuum volumetric flow rate of the gas (cm^3/sec), L_c, is given by:

$$L_c = \frac{2.48 \times 10^6 D^4}{a\mu}(P_u - P_d) * \frac{P_a}{P_u} = F_c * (P_u - P_d) * \frac{P_a}{P_u}$$

where:

F_c	=	coefficient for continuum flow [cm^3/atm-s]
D	=	capillary diameter [cm]
a	=	capillary length [cm]
μ	=	fluid viscosity [cP]
P_u	=	upstream pressure [atm] - pressure inside containment
P_d	=	downstream pressure [atm] - pressure outside containment

and, the molecular volumetric flow rate of the gas (cm^3/sec), L_m, is given by:

$$L_m = \frac{3.81x10^3 D^3 \sqrt{\frac{T}{M}}}{aP_a}(P_u - P_d) * \frac{P_a}{P_u} = F_m * (P_u - P_d) * \frac{P_a}{P_u}$$

where:

L_m	=	is the volumetric flow rate of gas at P_a [cm^3/sec]
F_m	=	is the coefficient for molecular flow [cm^3/atm-s]
D	=	is the capillary diameter [cm]
T	=	is the gas temperature [K]
M	=	is the gas molecular weight [g/mole]
P_a	=	is the average pressure $(P_u+P_d)/2$ [atm]
P_u	=	is the upstream pressure [atm]
P_d	=	is the downstream pressure [atm].
a	=	capillary diameter [cm]

For this analysis, the gas temperature used for molecular flow analysis is identical to the upstream temperature. Pressures and temperatures for PWR and BWR system normal operating conditions are summarized in Table 4.2-5. Based on the pressure, temperature and allowable leakage rate (L_N) the capillary diameter of the leak is determined. The calculated capillary diameter is then used to determine the air standard leak rate and helium test leak rate. Air standard condition leak rates are determined for air leaking from 1 atmosphere to 0.01 atmosphere at a temperature of 298K. The test gas is helium leaking from 1 atmosphere (0 psig) to a vacuum. Table 4.2-4 provides the standard and test leak rates for the Universal Transport Cask loaded with PWR or BWR fuel. The sensitivity for these tests is one-half the air standard leak rate as recommended by ANSI-N14.5-1997. The number of moles of cavity gases for normal and accident conditions are tabulated in Table 4.2-6. Key PWR and BWR containment analysis parameters are summarized in Table 4.2-7.

4.2.2 Pressurization of Containment Vessel

The maximum pressure in the cask during normal conditions of transport is calculated by using the methodology presented in Section 3.4.4. Assumptions underlying this calculation are that during normal conditions of transport, 3% of the fuel rods may fail and that 30% of the fission gases in the rods are releasable.

The cask cavity under normal conditions of transport is backfilled to 1 atm with at least 99.9% pure helium gas. To determine the limiting temperature conditions, a hypothetical replacement of the helium gas by 1 atm of air is assumed. This hypothetical situation reflects a failed canister and produces bounding canister temperatures as a result of air's lower thermal conductivity.

During pressure calculations, the gas volume (e.g. plenum and pellet to cladding gap) inside the fuel rods is conservatively neglected when calculating the cask free volume. Molar quantities of gases in the cask, the free volumes of the cask, and the cask normal operating condition pressure are calculated in Section 3.4.4.

4.2.3 Containment Criteria

The reference leak rates provided in Table 4.2-4 for PWR and BWR fuel, represent the maximum leak rate allowed if the o-rings were tested with air at 1 atm and 25°C. The maximum allowable leak rate for the containment system fabrication verification and periodic verification leak tests is described in Sections 4.1.3 and 8.1.3. This allowable leak rate is that for the BWR fuel configuration, which is more restrictive than the allowable leak rate for PWR fuel.

The sensitivity for these tests is recommended by ANSI N14.5-1997 to be one-half the allowable leak rate.

Table 4.2-1 Release Fractions: Normal and Accident Conditions

Radionuclide Origin	Fraction: Normal Conditions	Fraction: Accident Conditions
Volatiles releasable	2.00E-04	2.00E-04
Fission gas releasable	0.3	0.3
Rod mass released	3.00E-05	3.00E-05
Crud spallation factor	0.15	1.0
Fraction of fuel that fails	0.03	1.0

Table 4.2-2 Allowable Release Rate Source and A_2 Inputs for PWR Cask: Normal Conditions

B&W 15 x 15	**Crud**	**Gas**	**Volatiles**	**Fines**	**Total**
Total Activity per Assembly (Ci)	N/C[1]	3.73E+03	1.49E+05	2.28E+05	3.81E+05
Releasable Activity per Cask (Ci)	8.47E+01	8.06E+02	2.14E+01	4.92E+00	9.17E+02
Cask Volumetric Activity (Ci/cm^3)	1.20E-05	1.14E-04	3.04E-06	6.98E-07	1.30E-04
A_2 Value (Ci)	10.80	285.50	6.34	0.115	302.85
Fraction of Activity	0.092	0.879	0.023	0.005	1.000
Fraction of Activity / A_2 (1/Ci)	0.0086	0.0031	0.0037	0.0471	0.0624
			Mixture A_2 Value (Ci)		16.03

1. Not explicitly calculated.

Table 4.2-3 Allowable Release Rate Source and A_2 Inputs for BWR Cask: Normal Conditions

GE 9 x 9	Crud	Gas	Volatiles	Fines	Total
Total Activity per Assembly (Ci)	N/C[1]	1.46E+03	5.76E+04	8.82E+04	1.47E+05
Releasable Activity per Cask (Ci)	9.66E+02	7.34E+02	1.94E+01	4.44E+00	1.72E+03
Cask Volumetric Activity (Ci/cm^3)	1.40E-04	1.07E-04	2.81E-06	6.45E-07	2.50E-04
A_2 Value (Ci)	10.80	285.60	6.34	0.12	302.85
Fraction of Activity	0.56	0.43	0.01	0.003	1.00
Fraction of Activity / A_2 (1/Ci)	0.052	0.002	0.002	0.022	0.078
			Mixture A_2 Value (Ci)		12.90

1 Not explicitly calculated.

Table 4.2-4 Leak Rate and Leak Test Sensitivity: Normal Conditions

Reactor Type	Assembly Type	Operating Condition	Vol. Activity Ci/cm^3	Leak Rate (cm^3/sec) Volumetric (L)	Air Reference (L_R)	Test Sensitivity
PWR[1]	B&W 15x15	Normal	1.3E-04	3.4E-05	5.0E-05	2.5E-05
BWR[2]	GE 9x9-2	Normal	2.5E-04	1.4E-05	2.4E-05	1.2E-05

1 The corresponding helium test leak rates and leak test sensitivities for the PWR configuration are 6.7 x 10^{-5} cm^3/sec and 3.3 x 10^{-5} cm^3/sec, respectively, at standard conditions.

2 The corresponding helium test leak rates and leak test sensitivities for the BWR configuration are 3.3 x 10^{-5} cm^3/sec and 1.6 x 10^{-5} cm^3/sec, respectively, at standard conditions.

Table 4.2-5 Cask Free Volumes and Pressures: Normal and Accident Conditions

Reactor Type	**PWR**		**BWR**	
Cask Operating Condition	Normal	Accident[1]	Normal	Accident[1]
Free Gas Volume (liters)[2]	7,060	7,060	6,890	6,890
Pressure (atm)[2]	1.50	5.1	1.4	3.7
Average Gas Temperature (K)	528.7	696.3	495.3	644.1

1 The accident condition for this analysis is 100% rod failure in combination with a fire accident raising cask temperature. This hypothetical dual failure accident conservatively maximizes both available releasable material and cask pressure.

2 Free volume is rounded down to the nearest 10 liters and pressure is rounded up to the nearest 0.1 atmosphere. This conservatively minimizes free volume and capillary diameter.

Table 4.2-6 Cask Cavity Gases: Normal and Accident Conditions

Contributor	**PWR Cask (Number of Moles)**	**BWR Cask (Number of Moles)**
Total rod back-fill	153.31	25.01
Total fission gas	808.0	775.67
Canister backfill	185.96	209.87
Cask backfill	45.57	10.22

Table 4.2-7 PWR and BWR Containment Parameters - Normal Conditions

Assembly Type	Crud Surface Activity (Ci/cm^2)	Containment Free Volume (cm^3)	Capillary Length (cm)	Capillary Diameter (cm)	Upstream Pressure (atm)	Gas Temperature (K)
B&W 15x15	7.2E-5	7.1E+6	0.287	6.4E-4	1.5	529
GE 9x9-2	6.5E-4	6.9E+6	0.287	5.3E-4	1.4	495

Note: 3% of the fuel rods are postulated to fail in normal transport conditions.

THIS PAGE INTENTIONALLY LEFT BLANK

4.3 Containment Requirements for Hypothetical Accident Conditions

The 10 CFR 71 requirement for the release of radioactive material under hypothetical accident conditions is met by ensuring that for the cask containing PWR fuel a reference (air) leak rate limit of 2.6×10^{-3} ref·cm^3/sec is not exceeded. The corresponding air standard leak rate limit for the cask containing BWR fuel is 2.8×10^{-3} ref·cm^3/sec. Calculations of these limits are provided in Section 4.3.2.

Assuming a simultaneous occurrence of a fire accident and a 100% rod failure, and on the basis of bulk average gas temperatures of 696K (PWR) and 644K (BWR) resulting from air in the cavity, the pressure within the cask cavity is calculated to be 5.1 atm (PWR) or 3.7 atm (BWR). The hypothetical presence of air in the cask provides an upper bound on the gas temperature. These pressures represent the maximum possible cask internal pressures.

The structural integrity of the cask containment during hypothetical accident conditions is demonstrated in Chapter 2.0. Therefore, the cask containment is maintained under hypothetical accident conditions.

4.3.1 Fission Gas Products

The calculated amounts of fission gases contained in the design basis PWR and BWR fuel assembly are reported in Tables 4.5.3-1 and 4.5.3-4. The accident conditions for maximum fission gas release assume 100% rod failure and also assume that 30% of the radioactive fission gases, primarily ^{85}Kr, tritium and ^{129}I, are available for release to the cask cavity. In addition, 100% of the ^{60}Co in the crud on the fuel assemblies is conservatively assumed to be available for release as an aerosol.

4.3.2 Containment of Radioactive Materials

The Universal Transport Cask is designed to maintain a release rate of less than 1 A_2/week for the hypothetical accident conditions, as required by 10 CFR 71.51. A_2 for a mixed gas is determined by using the method described in 10 CFR 71, Appendix A. The release fractions for the various radionuclides found in the cask are obtained from NUREG/CR-6487 and summarized in Table 4.2-1. The curie content per isotope for 5-year cooled PWR and BWR design basis fuel assemblies is provided in Section 4.5.3.

4.3.2.1 Calculation of Allowable Leak Rates

The allowable leak rates under hypothetical accident conditions are calculated by using the method described in Section 4.2.1.1 for normal conditions of transport. The total inventory of fission product gases, volatiles, fines, and crud are calculated by using the source terms generated by SAS2H and release fractions for the PWR and the BWR fuel. Using the A_2 values from 10 CFR 71, Appendix A (Tables 4.3-1 and 4.3-2 1), the mixture A_2 values are then determined for gas, volatile, fine, and crud mixtures. Finally, the maximum allowable release rates are calculated by using the hypothetical accident conditions allowable release limit:

$$R_A = L_A C_A \leq A_2 \cdot \text{week}^{-1}$$

or

$$R_A = L_A C_A \leq A_2 \cdot 1.65 \times 10^{-6} \text{sec}^{-1}$$

where:

L_A = volumetric gas leakage rate [cm^3/s]
C_A = curies per unit volume (termed "activity density") of the radioactive material that passes through the leak path [Ci/cm^3]
R_A = release rate for accident transport conditions

Assumptions underlying the calculations for the hypothetical accident conditions are that 100% of the fuel rods fails and 100% of the crud is released (compared with the assumptions that 3% of the fuel rods fail and 15% of the crud is released in the analysis in Section 4.2.1.1 for normal conditions of transport). The mixture A_2 for gas, volatile, fine, and crud mixtures is not changed by the change in the magnitude of releasable material, but the combined A_2 changes based on the change in activity fraction in each group.

The calculated maximum permissible release rates for the casks containing design basis PWR and BWR fuel under hypothetical accident conditions are tabulated in Table 4.3-3.

4.3.2.2 Correlation of Allowable Leak Rates to Air Standard

The maximum allowable leak rates for the hypothetical accident conditions are correlated with standard leak rates by using the methodology described in Section 4.2.1.2. The results for casks containing PWR or BWR fuel as shown in Table 4.3-3.

4.3.3 Containment Criteria

The allowable leak rates calculated for the hypothetical accident conditions are much greater than those for the normal conditions of transport calculated in Section 4.2.1. Because the cask containment is demonstrated to be maintained under hypothetical accident conditions (Section 2.7), the maximum permissible leak rates for normal conditions of transport are more limiting and are therefore used for the establishment of the maximum allowable leak rates for the containment system fabrication and periodic verification leak test calculations and test acceptance criteria.

Table 4.3-1 Allowable Release Rate Source and A_2 Inputs for PWR Cask: Accident Conditions

B&W 15 x 15	**Crud**	**Gas**	**Volatiles**	**Fines**	**Total**
Total Activity per Assembly (Ci)	N/C[1]	3.73E+03	1.49E+05	2.28E+05	3.81E+05
Releasable Activity per Cask (Ci)	5.65E+02	2.69E+04	7.15E+02	1.64E+02	2.83E+04
Cask Volumetric Activity (Ci/cm^3)	7.99E-05	3.80E-03	1.01E-04	2.33E-05	4.01E-03
A_2 Value (Ci)	10.80	285.50	6.39	0.114	302.8
Fraction of Activity	0.020	0.949	0.025	0.006	1.00
Fraction of Activity / A_2 (1/Ci)	0.002	0.003	0.004	0.051	0.06
			Mixture A_2 Value (Ci)		16.68

1 Not explicitly calculated.

Table 4.3-2 Allowable Release Rate Source and A_2 Inputs for BWR Cask: Accident Conditions

GE 9 x 9	**Crud**	**Gas**	**Volatiles**	**Fines**	**Total**
Total Activity per Assembly (Ci)	N/C[1]	1.46E+03	5.76E+04	8.82E+04	1.47E+05
Releasable Activity per Cask (Ci)	6.44E+03	2.45E+04	6.45E+02	1.48E+02	3.17E+04
Cask Volumetric Activity (Ci/cm^3)	9.35E-04	3.55E-03	9.36E-05	2.15E-05	4.60E-03
A_2 Value (Ci)	10.80	285.60	6.34	0.115	302.85
Fraction of Activity	0.203	0.772	0.0204	0.005	1.00
Fraction of Activity / A_2 (1/Ci)	0.019	0.003	0.003	0.041	0.065
			Mixture A_2 Value (Ci)		15.31

1 Not explicitly calculated.

Table 4.3-3 Standard Leak Rates: Accident Conditions

Reactor Type	Assembly Type	Operating Condition	Vol. Activity (Ci/cm^3)	Leak Rates (cm^3/sec) Volumetric (L)	Air Reference (L_R)
PWR	B&W 15x15	Accident	4.0E-03	6.9E-03	2.6E-03
BWR	GE 9x9-2	Accident	4.6E-03	5.5E-03	2.8E-03

Table 4.3-4 PWR and BWR Containment Parameters - Accident Conditions

Assembly Type	Crud Surface Activity (Ci/cm^2)	Containment Free Volume (cm^3)	Capillary Length (cm)	Capillary Diameter (cm)	Upstream Pressure (atm)	Gas Temperature (K)
B&W 15x15	7.2E-5	7.1E+6	0.287	1.8E-3	5.1	696
GE 9x9-2	6.5E-4	6.9E+6	0.287	1.8E-3	3.7	644

Note: 100 % of the fuel rods are postulated to fail in the accident condition.

THIS PAGE INTENTIONALLY LEFT BLANK

4.4 Special Requirements

The Universal Transport Cask is not used for plutonium shipments and thus is not subject to the special requirements of 10 CFR 71.63.

THIS PAGE INTENTIONALLY LEFT BLANK

4.5 Appendices

THIS PAGE INTENTIONALLY LEFT BLANK

4.5.1 Containment Evaluation for Site Specific Contents

4.5.1.1 Containment Evaluation for Maine Yankee Contents

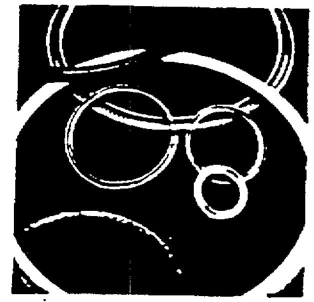
Pressure and radionuclide content of the Maine Yankee 14x14 fuel assemblies are bounded by the larger B&W 15x15 assembly employed in the containment evaluations presented in Sections 4.2 and 4.3. The larger fission mass of the B&W assembly produces higher fission gas inventories for a fixed burnup. The Maine Yankee fuel assemblies, with non-fuel components or in consolidated form (up to four consolidated assemblies per canister) displaces less free volume than the B&W fuel assembly forming the design basis for the containment analysis, and, therefore, results in lower pressures at a fixed decay heat.

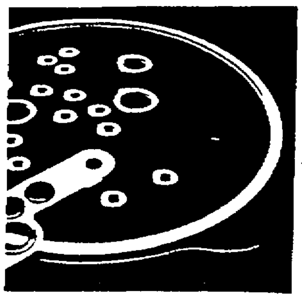
THIS PAGE INTENTIONALLY LEFT BLANK

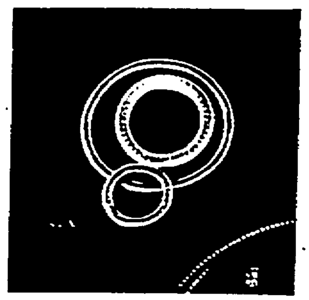
4.5.2 Technical Information on EPDM O-Rings

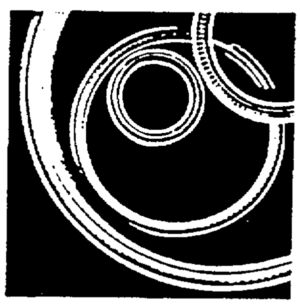
This appendix contains the manufacturer's technical bulletins for the O-rings and seals.

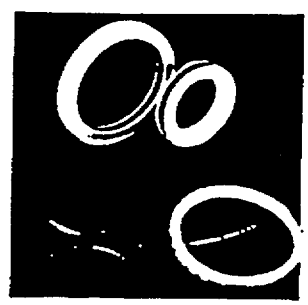
FLUOROCARBON MECHANICAL SEAL DIVISION

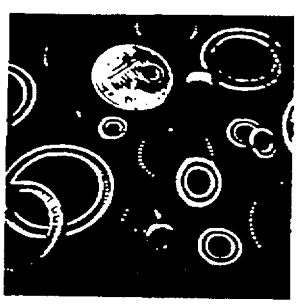








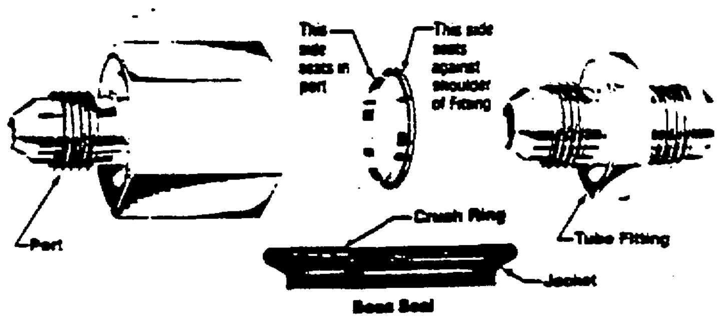


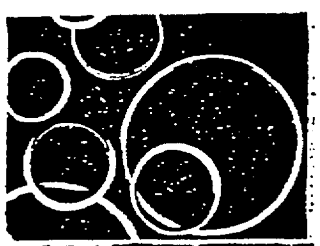


Omniseal® Design Handbook

Boss Seal

Description:
The Metal Boss Seal is a pressure loaded, positive sealing device designed for service in the AN (MS, MC) port-to-tube fitting in applications beyond the capabilities of available elastomer O-rings. The initial seal is accomplished by means of the "crush-ring" which mates the coating of the seal to the surfaces of the port and tube fitting. Once the seal is installed, it is spring loaded and pressure sensitive to insure sealing over a wide range of temperature and pressure. The advantages of both "crush" and "pressure sensitive" type seals are inherent in the design of the Boss Seal.





Advantages
1. Unlimited shelf life.
2. Temperature capability from -423° F. 1500° F
3. Designed to operate within wide pressure range.
4. Compatible with corrosive and radioactive fluids.
5. Sealing force increases with increased pressure.
6. Allows metal-to-metal seat of fittings.
7. Permits full thread engagement.
8. Non-permeable.
9. Fits standard AN, MS and MC hardware.
10. Seal pressure limit exceeds proof pressure of hardware.

Ordering Information
Seal Part Number should be written as follows:

10061-XX-X-X

Basic Part No.

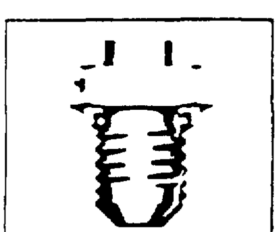
Material
0 321 Cres Jacket
302 Cres Crush Ring
1 Inconel X-718 Jacket (Special Order)
Inconel X-718 Crush Ring

	Coating	Temp. Range[1]
0	Teflon	-423° F. to 500°
1	Silver	-320° F. to 1300°
2	Gold	-423° F. to 1400°
3	Nickel	-453° F. to 1500°
4	Copper	-423° F. to 1500°
5	No Coating[2]	-453° F. to 1500°

[1]Temperatures given are achieved with Cres 321/302. Inconel seals are available for higher temperatures and corrosion resistance.

[2]Uncoated seals may require polishing of mating surfaces.

Size Dash Number
02 AND 10050-2/MS33649-2
03 AND 10050-3/MS33649-3
04 AND 10050-4/MS33649-4
05 AND 10050-5/MS33649-5
06 AND 10050-6/MS33649-6
08 AND 10050-8/MS33649-8
10 AND 10050-10/MS33649-10
12 AND 10050-12/MS33649-12
16 AND 10050-16/MS33649-16
Larger sizes available on special order.

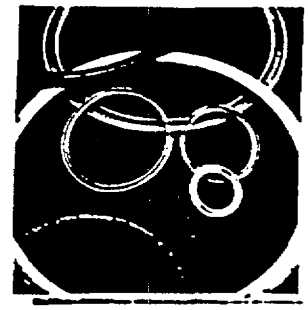


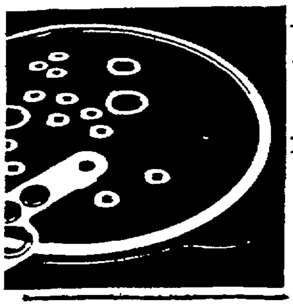
Installation
1. Lightly lubricate fitting threads, boss cavity, and top of seal with suitable lubricant except where prohibited by cleaning requirements.
2. Place seal over threads of the fitting and assemble into mating port.
3. Apply recommended torque. If no lubricant is used a higher torque may be required to bottom the fitting against the boss.

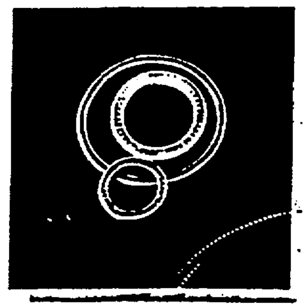
Recommended Torque

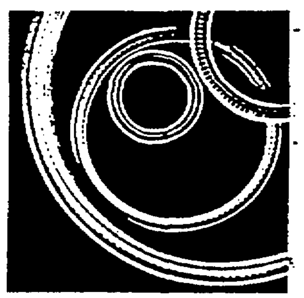
Boss Seal	Stainless Steel (in Lbs.)	Aluminum (in Lbs.)
10061 - 02	120 - 185	[illegible]
10061 - 03	200 - 250	[illegible]
10061 - 04	300 - 350	[illegible]
10061 - 05	360 - 420	[illegible]
10061 - 06	420 - 480	[illegible]
10061 - 08	600 - 720	[illegible]
10061 - 10	980 - 1080	[illegible]
10061 - 12	1380 - 1500	[illegible]
10061 - 16	1440 - 1580	[illegible]

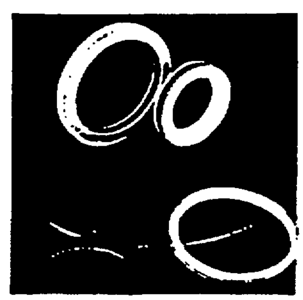
Torque values based on lubricated threads and seal surfaces.

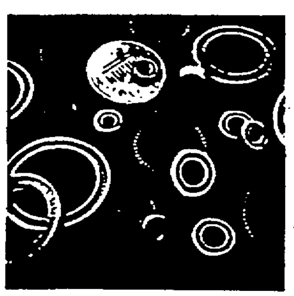






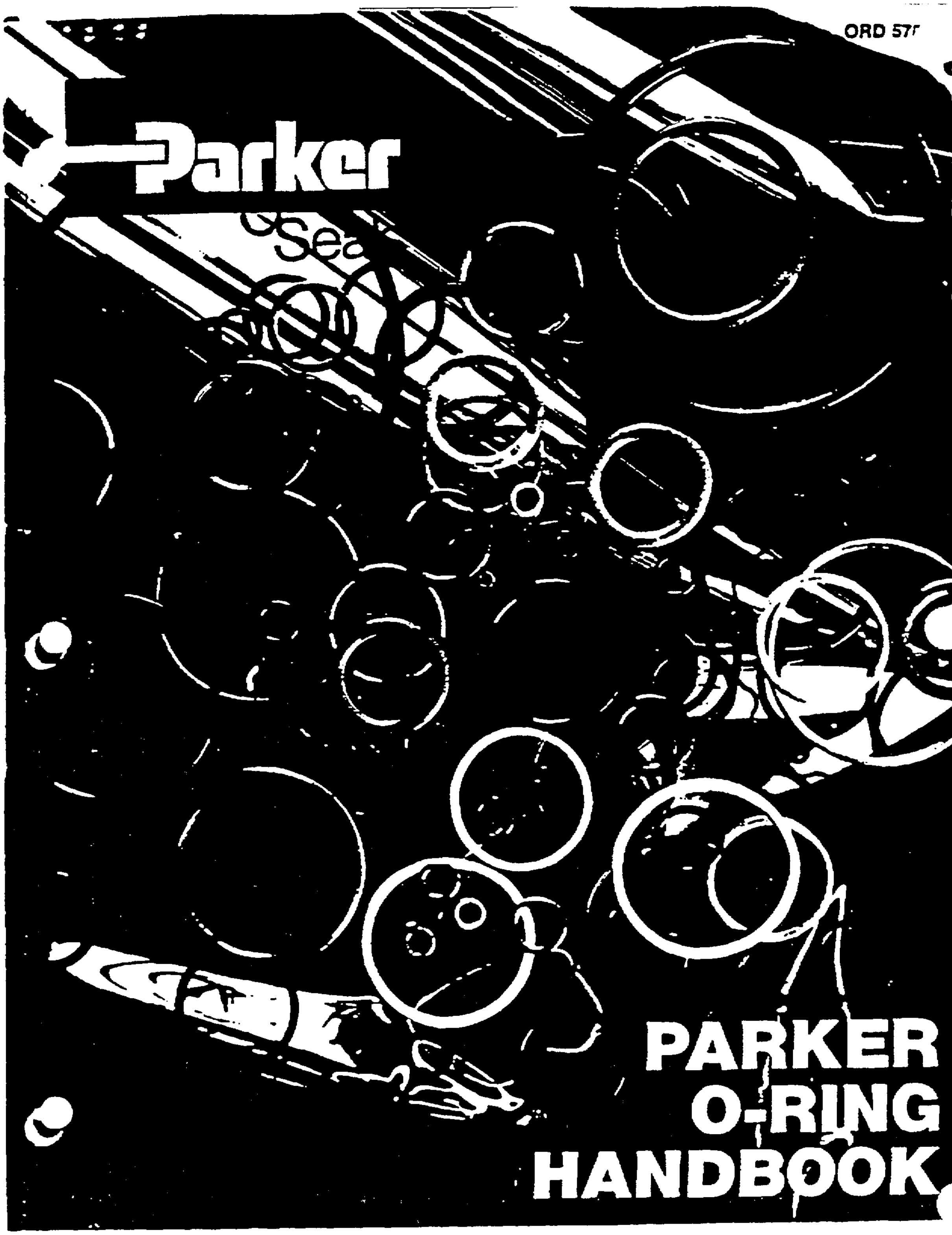






FLUOROCARBON

MECHANICAL SEAL DIVISION
4412 Corporate Center Drive
P.O. Box 520
Los Alamitos, CA 90720
(213) 594-0941 (714) 995-1818
FAX (714) 761-1270
TWX 910-341-7672


ORD 57
Parker
PARKER
O-RING
HANDBOOK

ETHYLENE PROPYLENE RUBBER (EPM, EPDM)

Typical Trade Names:

Nordel....................E. I. duPont de Nemours Co.
Royalene .... Uniroyal
Vistalon......................Exxon Chemical Co. USA
Epsyn .... Copolymer Rubber & Chemical Corp.
Epcar .... B. F. Goodrich Co.

Ethylene propylene rubber is an elastomer prepared from ethylene and propylene monomers (ethylene propylene copolymer) and at times with a small amount of a third monomer (ethylene propylene terpolymers). It was introduced to the rubber industry in 1961 and quickly won broad acceptance in the sealing world because of its excellent resistance to Skydrol and other phosphate ester type hydraulic fluids.

Ethylene propylene has a temperature range of −65 to +300°F (−54 to +149°C) for most applications.

EP is recommended for:

Phosphate ester base hydraulic fluids (Skydrol, Fyrquel, Pydraul).
Steam (to 400°F) (204°C).
Water.
Silicone oils and greases.
Dilute acids.
Dilute alkalies.
Ketones (MEK, acetone).
Alcohols.
Automotive brake fluids.

EP is not recommended for:

Petroleum oils.
Di-ester base lubricants.

FLUOROCARBON RUBBER (FKM)

Typical Trade Names:

Fluorel 3M
Kalrez (high temp) ......E.I. duPont de Nemours Company
Kel-F ....3M (formerly Kellogg)
Viton ....E.I duPont de Nemours Company

Fluorocarbon elastomers were first introduced in the mid 1950's. Since then they have grown to major importance in the seal industry. Due to its wide spectrum chemical compatibility and temperature range and its low compression set, fluorocarbon rubber is the most significant single elastomer development in recent history.

Its working temperature range is considered to be −15 to +400°F (−29 to +204°C), but it will take temperatures up to 600°F (316°C) for short periods of time, and Du Pont's Kalrez is normally recommended up to 500°F (260°C). On the low temperature end, Parker's compound V835-75 will seal down to −40°F (−40°C) in a static seal. Though the standard compounds have been known to seal at −65°F (−54°C) in some special static applications, the normal low temperature limit is −15°F (−26°C).

Special formulations having extra chemical resistance are also available, and new types are being developed constantly.

Fluorocarbon O-rings should be considered for seal use in aircraft, automobile and other mechanical devices requiring maximum resistance to elevated temperature and to many functional fluids.

FKM is recommended for:

Petroleum oils.
Di-ester base lubricants (MIL-L-7808, MIL-L-6085).
Silicate ester base lubricants (MLO 8200, MLO 8515, OS-45.)
Silicone fluids and greases.
Halogenated hydrocarbons (carbon tetrachloride, trichloroethylene).
Selected phosphate ester fluids.
Acids.

FKM is not recommended for:

Ketones (MEK, acetone).
Skydrol fluids.
Amines (UDMH), anhydrous ammonia.
Low molecular weight esters and ethers.
Hot hydrofluoric or chlorosulfonic acids.

FLUOROSILICONE (FSI)

Typical Trade Name:

Silastic LS. ....Dow Corning Corp.

Fluorosilicone combines the good high- and low-temperature properties of silicone with basic fuel and oil resistance. The primary uses of fluorosilicones are in fuel systems at temperatures up to 350°F (177°C), and in applications where the dry-heat resistance of silicone is required, but the seal may be exposed to petroleum oils and/or hydro-carbon fuels. In some fuels and oils, however, the high temperature limit is more conservative because temperatures approaching 350°F may degrade the fluid, producing acids which attack fluorosilicone elastomers.

On the other end of the temperature scale, fluorosilicones typically seal at temperatures as low as −100°F (−73°C). High strength type fluorosilicones are available. Certain of these exhibit much improved resistance to compression set.

ISOPRENE RUBBER-SYNTHETIC (IR)

Typical Trade Names:

Natsyn ....Goodyear Tire & Rubber Co.

Polyisoprene has the distinction of being a synthetic elastomer which has the same chemical composition as natural rubber. For a guide to its chemical and physical properties, refer to Natural Rubber below.

NATURAL RUBBER—NATURAL POLYISOPRENE (NR)

Crude natural rubber is found in the juices of many plants, including the shrub guayule, Russian dandelion, goldenrod and dozens of other shrubs, vines and trees. The principal source is the tree Hevea Brasiliensis which is native to Brazil. Petroleum oils are the greatest enemy of natural rubber compounds. The synthetics have all but completely replaced natural rubber for seal use.

NR is recommended for:

Automotive brake fluid.

NR is not recommended for:

Petroleum products.

NEOPRENE RUBBER (CHLOROPRENE, CR)

Trade Name:
Neoprene (formerly E. I. duPont de Nemours Company)
Butaclor .... Distugil
Denka .... Denka Chem .Co.

Neoprenes are homopolymers of chloroprene (chlorobutadiene) and were among the earliest of the synthetic rubbers available to the seal manufacturers.

Neoprene can be compounded for service at temperatures of −65 to +300°F (−54 to +149°C). Most elastomers are either resistant to deterioration from exposure to petroleum lubricants or oxygen. Neoprene is unusual in having limited resistance to both. This, combined with broad temperature range and moderate cost accounts for its desirability in many seal applications.

Chloroprene is recommended for:
Refrigerants (Freons, ammonia)
High aniline point petroleum oils.
Mild acid resistance.
Silicate ester lubricants.

Chloroprene is not recommended for:
Phosphate ester fluids.
Ketones (MEK, acetone).

NITRILE OR BUNA N (NBR)

Typical Trade Names:
Chemigum .... Goodyear Tire & Rubber Co.
Paracril .... Uniroyal
Hycar .... Goodrich Chemical Co.
Krynac .... Polysar, Ltd.
Ny Syn .... Copolymer Rubber & Chem. Corp.

Nitrile, chemically, is a copolymer of butadiene and acrylonitrile. Acrylonitrile content is varied in commercial products from 18% to 48%. As the nitrile content increases, resistance to petroleum base oils and hydrocarbon fuels increases, but low temperature flexibility decreases.

Due to its excellent resistance to petroleum products, and its ability to be compounded for service over a temperature range of −65 to +275°F (−54 to +135°C), nitrile is the most widely used elastomer in the seal industry today. Most military rubber specifications for fuel and oil resistant MS and AN O-rings require nitrile base compounds. It should be mentioned, however, that to obtain good resistance to low temperature with nitrile compounding, it is almost always necessary to sacrifice some high temperature fuel and oil resistance.

Nitrile compounds are superior to most elastomers with regard to compression set or cold flow, tear and abrasion resistance. Inherently, they do not possess good resistance to ozone, sunlight or weather but this can be substantially improved through compounding. However, since ozone and weather resistance are not always built in, seals from nitrile bases should not be stored near electric motors or other equipment which may generate ozone, or in direct sunlight.

Nitrile is recommended for:
General purpose sealing.
Petroleum oils and fluids.
Cold Water.
Silicone greases and oils.
Di-ester base lubricants (MIL-L-7808).
Ethylene glycol base fluids (Hydrolubes).

Nitrile is not recommended for:
Halogenated hydrocarbons (carbon tetrachloride, trichloroethylene).
Nitro hydrocarbons (nitrobenzene, aniline).
Phosphate ester hydraulic fluids (Skydrol, Fyrquel, Pydraul).
Ketones (MEK, acetone).
Strong acids.
Ozone.
Automotive brake fluid.

POLYPHOSPHAZENE FLUOROELASTOMER (FZ)

Trade Name:
EYPEL-F .... Ethyl Corp.

EYPEL-F elastomer should effectively solve many difficult sealing problems due to its combination of physical properties, fluid resistance and temperature range.

The base polymer was developed for the U.S. Army by Firestone Tire and Rubber Company, and it has much the same temperature range (-85 to +325/350°F) and fluid resistance (especially petroleum products) as fluorosilicone elastomers but physical properties are definitely better — enough so that polyphosphazene compounds have performed adequately in dynamic and extrusion tests. Major disadvantage is its resistance to water which is only fair to poor.

POLYACRYLATE RUBBER (ACM)

Typical Trade Names:
Cyanacryl .... American Cyanamid Co.
Hycar .... B. F. Goodrich Chemical Co.

This material has outstanding resistance to petroleum fuel and oil. In addition, it possesses complete resistance to oxidation, ozone, and sunlight, combined with an ability to resist flex cracking. Compounds from this base polymer have been developed which are adaptable for continuous service in hot oil over the temperature range 0° to +350°F (−18° to +177°C). Resistance to hot air is slightly superior to nitrile polymers, but strength, compression set and water resistance are inferior to many of the other polymers.

There are several polyacrylate types available commercially, but all are polymerization products of acrylic acid esters.

Greatest usage of polyacrylate is by the automotive industry in automatic transmissions and power steering gears using Type A fluid.

Elastomers

A3-6

TABLE A3-8

PARKER SIZE NUMBER	CLASS III				CLASS IV				CLASS V				CLASS VI			
	ID	TOL ±	W	TOL ±	ID	TOL ±	W	TOL ±	ID	TOL ±	W	TOL ±	ID	TOL ±	W	TOL ±
2-457	13.751	.120	.271	.008	13.682	.137	.269	.009	13.612	.154	.268	.009	13.542	.170	.266	.009
2-458	14.243	.122			14.171	.139			14.099	.157			14.026	.174		
2-459	14.735	.124	.271	.008	14.661	.142	.269	.009	14.586	.160	.268	.009	14.511	.181	.266	.009
2-460	15.227	.126	.271	.008	15.150	.144	.269	.009	15.073	.163	.268	.009	14.995	.181	.266	.009
2-461	15.700	.132			15.620	.152			15.540	.171			15.460	.190		
2-462	16.192	.134			16.109	.154			16.027	.174			15.945	.193		
2-463	16.684	.141			16.599	.161			16.514	.182			16.429	.202		
2-464	17.176	.148			17.088	.169			17.001	.190			16.914	.211		
2-465	17.668	.150			17.578	.171			17.488	.193			17.398	.214		
2-466	18.160	.151			18.067	.174			17.975	.196			17.883	.218		
2-467	18.652	.158			18.557	.181			18.462	.204			18.367	.226		
2-468	19.144	.160			19.046	.183			18.949	.207			18.852	.230		
2-369	19.636	.167			19.536	.191			19.436	.215			19.336	.239		
2-470	20.620	.170			20.515	.196			20.410	.221			20.305	.246		
2-471	21.604	.179			21.494	.205			21.384	.232			21.274	.258		
2-472	22.573	.188			22.458	.215			22.344	.243			22.229	.270		
2-473	23.557	.196			23.437	.225			23.318	.254			23.198	.282		
2-474	23.541	.205			24.416	.235			24.292	.265			24.167	.296		
2-475	25.525	.213	.271	.008	25.395	.245	.269	.009	25.266	.276	.268	.009	25.136	.307	.266	.009
3-901	.182	.006	.055	.004	.181	.006	.055	.004	.180	.006	.066	.004	.179	.006	.064	.004
3-902	.235	.006	.063		.234	.006	.063		.233	.006	.062		.232	.007	.063	
3-903	.296	.006	.063		.295	.006	.063		.293	.007	.062		.292	.007	.062	
3-904	.345	.006	.071		.344	.007	.070		.342	.007	.070		.340	.008	.070	
3-905	407	006	.071		405	007	.070		.403	.007	.070		.401	.008	.070	
3-906	461	.007	.077		458	.007	.076		.456	.008	.076		.453	.008	.076	
3-907	.522	.009	.081		.519	.010	.080		.516	.010	.080		.514	.011	.079	
3 908	.634	.011	.086		.630	.012	.085		.627	.013	.085		.624	.014	.084	
3-909	.695	.012	.095		.691	.012	.095		.688	.013	.094		.684	.014	.094	
3-910	743	012	.095	004	739	.013	095	.004	.735	.014	.094	.004	.732	.014	.094	.004
3-911	.849	012	.114	.005	845	013	114	.005	.841	.014	.113	.006	.836	.015	.112	.006
3-912	.909	.012	114		905	013	114		.900	.015	.113		.895	.016	.113	
3-913	.970	.014	114		965	015	114		.960	.016	.113		.955	.017	.113	
3-914	1 030	.014	.114		1 025	.015	114		1.020	.016	.113		1.015	.018	.113	
3-916	1 152	014	114		1 146	.016	114		1.141	.017	.113		1.135	.018	.113	
3-918	1 333	017	114		1 327	019	114		1.320	.020	.113		1.313	.022	.113	
3-920	1 451	019	116		1 444	021	116		1.437	.023	.115		1.429	.025	.114	
3 924	1 692	.020	116		1 684	022	116		1.675	.024	.115		1.667	.026	.114	
3 928	2 057	026	116		2 046	028	116		2.036	.031	.115		2.025	.033	.114	
3 932	2.300	.026	116	005	2 288	029	116	.005	2.276	032	.115	.006	2.265	.035	114	.006

A3-31

Elastomers

The high temperature limits assigned to compounds in this handbook are conservative estimates of the maximum temperature for 1000 hours of continuous service in the media the compounds are most often called on to seal. Since the top limit for any compound varies with the medium, the high temperature limit for many compounds is shown as a range rather than a single figure. This range may be reduced or extended in unusual fluids.

Since some fluids decompose at a temperature lower than the maximum temperature limit of the elastomer, the temperature limits of both the seal and the fluid must be considered in determining limits for a system.

Low temperature service ratings in the past have been based on values obtained by ASTM test methods D736 and D746. The present ASTM D2000/SAE 200 specification still calls for the ASTM D746 low temperature test (ASTM D736 is obsolete). For O-rings and other compression seals, however, the TR-10 value per ASTM D1414 provides a better means of approximating the low temperature capability of an elastomer compression seal, the low temperature sealing limit being generally about 15°F below the TR-10 value. This is the formula that has been used, with a few exceptions, to establish the recommended low temperature limits for Parker Seal Group compounds in tables A3-13, B5, and B10.

This is the lowest temperature normally recommended for static seals. In dynamic use, or static applications with pulsing pressure, sealing may not be accomplished below the TR10 temperature, or 15°F higher than the low limit recommendation in the Handbook.

These recommendations are based on Parker tests. Some manufacturers use a less conservative method to arrive at low temperature recommendations, but similar compounds with the same TR10 temperature would be expected to have the same actual low temperature limit regardless of catalog recommendations.

A few degrees may sometimes be gained by increasing the squeeze on the O-ring section, while insufficient squeeze may cause O-ring leakage before the recommended low temperature limit is reached.

The low temperature limit on an O-ring seal may be compromised if the seal is previously exposed to extra high temperature or a fluid that causes it to take a set, or to a fluid that causes the seal compound to shrink. Conversely, the limit may be lowered significantly if the fluid swells the compound.

With decreasing temperature, elastomers shrink approximately ten times as much as surrounding metal parts. In a rod type assembly, whether static or dynamic, this effect causes the sealing element to hug the rod more firmly as the temperature goes down. Therefore, an O-ring may seal below the recommended low temperature limit when used as a rod type seal.

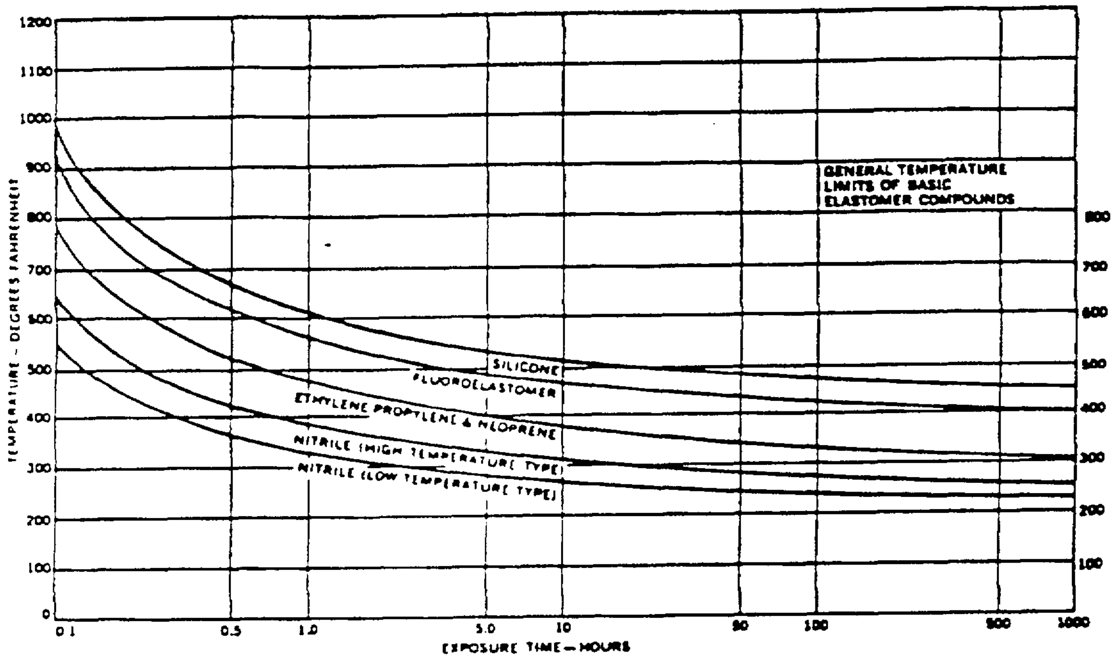


FIGURE A3-6 SEAL LIFE AT TEMPERATURE

This chart is intended only as a rough guide. It cannot be used for precise predictions of seal life. Results will vary with compound and fluid medium.

A3-36

TABLE B6

PARKER SERIES 2-XXX O-RINGS SIZE CROSS REFERENCE TABLE

1	2	3			4				5	6				7
	Size Only	NOMINAL SIZE			STANDARD O-RING SIZE (Units Are in Inches)					METRIC O-RING SIZE (Units Are in Millimeters)				
	(Size Only)	(Inches)			Actual (b) Per AS 568A					Actual (b) Per AS 568A				
PARKER SIZE NO. (Size Only) (a)	AS 568A Uniform Dash No.	(REF. ONLY) I.D.	O.D.	W.	I.D.	Tolerance + -	(REF. ONLY) W	±	Basic Volume Cu. In	I.D	Tolerance + -	W	±	PARKER SIZE NO (Size Only) (a)
2-175	-175	9	9 3/16	3/32	8.987	.050	.103	.003	.2379	228.27	1.27	2.62	0.08	2-175
2-176	-176	9 1/4	9 7/16	3/32	9.237	.055	.103	.003	.2445	234.62	1.40	2.62	0.08	2-176
2-177	-177	9 1/2	9 11/16	3/32	9.487	.055	.103	.003	.2510	240.97	1.40	2.62	0.08	2-177
2-178	-178	9 3/4	9 15/16	3/32	9.737	.055	.103	.003	.2576	247.32	1.40	2.62	0.08	2-178
2-201	-201	3/16	7/16	1/8	.171	.005	.139	.004	.0148	4.34	0.13	3.53	0.10	2-201
2-202	-202	1/4	1/2	1/8	.234	.005	.139	.004	.0178	5.94	0.13	3.53	0.10	2-202
2-203	-203	5/16	9/16	1/8	.296	.005	.139	.004	.0207	7.52	0.13	3.53	0.10	2-203
2-204	-204	3/8	5/8	1/8	.359	.005	.139	.004	.0237	9.12	0.13	3.53	0.10	2-204
2-205	-205	7/16	11/16	1/8	.421	.005	.139	.004	.0267	10.69	0.13	3.53	0.10	2-205
2-206	-206	1/2	3/4	1/8	.484	.005	.139	.004	.0297	12.29	0.13	3.53	0.10	2-206
2-207	-207	9/16	13/16	1/8	.546	.007	.139	.004	.0327	13.87	0.18	3.53	0.10	2-207
2-208	-208	5/8	7/8	1/8	.609	.009	.139	.004	.0357	15.47	0.23	3.53	0.10	2-208
2-209	-209	11/16	15/16	1/8	.671	.010	.139	.004	.0386	17.04	0.23	3.53	0.10	2-209
2-210	-210	3/4	1	1/8	.734	.010	.139	.004	.0416	18.64	0.25	3.53	0.10	2-210
2-211	-211	13/16	1 1/16	1/8	.796	.010	.139	.004	.0446	20.22	0.25	3.53	0.10	2-211
2-212	-212	7/8	1 1/8	1/8	.859	.010	.139	.004	.0476	21.82	0.25	3.53	0.10	2-212
2-213	-213	15/16	1 3/16	1/8	.921	.010	.139	.004	.0505	23.39	0.25	3.53	0.10	2-213
2-214	-214	1	1 1/4	1/8	.984	.010	.139	.004	.0535	24.99	0.25	3.53	0.10	2-214
2-215	-215	1 1/16	1 5/16	1/8	1.046	.010	.139	.004	.0565	26.57	0.25	3.53	0.10	2-215
2-216	-216	1 1/8	1 3/8	1/8	1.109	.012	.139	.004	.0595	28.17	0.30	3.53	0.10	2-216
2-217	-217	1 3/16	1 7/16	1/8	1.171	.012	.139	.004	.0624	29.74	0.30	3.53	0.10	2-217
2-218	-218	1 1/4	1 1/2	1/8	1.234	.012	.139	.004	.0654	31.34	0.30	3.53	0.10	2-218
2-219	-219	1 5/16	1 9/16	1/8	1.296	.012	.139	.004	.0684	32.92	0.30	3.53	0.10	2-219
2-220	-220	1 3/8	1 5/8	1/8	1.359	.012	.139	.004	.0714	34.52	0.30	3.53	0.10	2-220
2-221	-221	1 7/16	1 11/16	1/8	1.421	.012	.139	.004	.0744	36.09	0.30	3.53	0.10	2-221
2-222	-222	1 1/2	1 3/4	1/8	1.484	.015	.139	.004	.0774	37.69	0.38	3.53	0.10	2-222
2-223	-223	1 5/8	1 7/8	1/8	1.609	.015	.139	.004	.0833	40.87	0.38	3.53	0.10	2-223
2-224	-224	1 3/4	2	1/8	1.734	.015	.139	.004	.0893	44.04	0.38	3.53	0.10	2-224
2-225	-225	1 7/8	2 1/8	1/8	1.859	.018	.139	.004	.0952	47.22	0.46	3.53	0.10	2-225
2-226	-226	2	2 1/4	1/8	1.984	.018	.139	.004	.1012	50.39	0.46	3.53	0.10	2-226
2-227	-227	2 1/8	2 3/8	1/8	2.109	.018	.139	.004	.1072	53.57	0.46	3.53	0.10	2-227
2-228	-228	2 1/4	2 1/2	1/8	2.234	.020	.139	.004	.1131	56.74	0.51	3.53	0.10	2-228
2-229	-229	2 3/8	2 5/8	1/8	2.359	.020	.139	.004	.1191	59.92	0.51	3.53	0.10	2-229
2-230	-230	2 1/2	2 3/4	1/8	2.484	.020	.139	.004	.1250	63.09	0.51	3.53	0.10	2-230
2-231	-231	2 5/8	2 7/8	1/8	2.609	.020	.139	.004	.1310	66.27	0.51	3.53	0.10	2-231
2-232	-232	2 3/4	3	1/8	2.734	.024	.139	.004	.1370	69.44	0.61	3.53	0.10	2-232
2-233	-233	2 7/8	3 1/8	1/8	2.859	.024	.139	.004	.1429	72.62	0.61	3.53	0.10	2-233
2-234	-234	3	3 1/4	1/8	2.984	.024	.139	.004	.1489	75.79	0.61	3.53	0.10	2-234
2-235	-235	3 1/8	3 3/8	1/8	3.109	.024	.139	.004	.1548	78.97	0.61	3.53	0.10	2-235
2-236	-236	3 1/4	3 1/2	1/8	3.234	.024	.139	.004	.1608	82.14	0.61	3.53	0.10	2-236
2-237	-237	3 3/8	3 5/8	1/8	3.359	.024	.139	.004	.1668	85.32	0.61	3.53	0.10	2-237

(a) The rubber compound must be added when ordering by the 2- size number (i.e. 2-007 N674-70). See table B10 for standard Parker O-ring compounds.

(b) Table B6 provides dimensions for standard (AN) shrinkage materials ONLY. These correspond to AS568A dimensions. O-rings manufactured out of compounds with different shrinkage rates (other than AN) will produce slightly different dimensions and tolerances. See pages A3-15 through A3-31 for more information.

.103 AREA = .008392
.139 AREA = .015175

Sizes


Parker
Seals
PARKER SEAL GROUP
RING DIVISION
360 PALUMBO DRIVE
P.O. 11751
LEXINGT N, KY 40512
(606) 269- 351
GL-9/92

THIS PAGE INTENTIONALLY LEFT BLANK

4.5.3 SAS2H Input, Output and Group A_2 Values for B&W 15x15 and GE 9x9 Assemblies

SAS2H output is provided on a per assembly basis for B&W 15x15 and GE 9x9 fuel assemblies, based on a burnup of 50,000 MWD/MTU, and initial enrichment of 1.9 wt% ^{235}U and a 5-year cool time. Radionuclides are grouped into fission gases, volatiles, and fines. Volatiles and gases are defined as those radionuclides listed in NUREG/CR-6487. All remaining radionuclides are grouped as fines. A_2 values were obtained from 10 CFR 71.

Figure 4.5-1 PWR SAS2H Input File for Design Basis Fuel Source Terms

```
=SAS2H    PARM=(HALT03,SKIPSHIPDATA)
B&W 15X15 1.9 W/O U235, 50000 MWD/MTU 5-10 YEAR COOLING
27GROUPNDF4 LATTICECELL
UO2      1 0.95 900 92235 1.9 92238 98.1 END
ZIRCALLOY  2 1.0 620 END
H2O      3 DEN=0.725 1.0 580 END
ARBM-BORMOD 0.725 1 1 0 0 5000 100 3 550.0E-6 580 END
END COMP
SQUAREPITCH 1.4427 0.9362 1 3 1.0922 2 0.9576 0 END
NPIN=208 FUEL=365.76 NCYC=3 NLIB=1 PRIN=6 LIGH=5
INPL=1 NUMH=16 NUMI=1 ORTU=0.6731 SRTU=0.6325 END
POWER=17.16 BURN=466.9 DOWN=60.0 END
POWER=17.16 BURN=466.9 DOWN=60.0 END
POWER=17.16 BURN=466.9 DOWN=1461.00 END
FE 0.672 CR 0.190 NI 0.115 MN 0.020 CO 0.0012
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING TO 10 YEARS AND FISSION PRODUCT GAMMA REBIN
3$$  21 0 1 A33 -86  E
54$$  A8 1 E T
35$$  0 T
56$$  0 6 A13 -2 5 3 E
57**  4.0 E T
COOLING TO 10 YEARS AND FISSION PRODUCT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60**  5.0 6.0 7.0 8.0 9.0 10.0
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F1E-6
81$$  2 51 26 1 E
82$$  F6 T
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
56$$ F0  T
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING TO 10 YEARS AND ACTINIDE GAMMA REBIN
3$$  21 0 1 A33 -86  E
54$$  A8 1 E T
35$$  0 T
56$$  0 6 A13 -2 5 3 E
57**  4.0 E T
```

Figure 4.5-1 PWR SAS2H Input File for Design Basis Fuel Source Terms (Continued)

```
COOLING TO 10 YEARS AND ACTINIDE GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0
65$$ A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61** F1E-6
81$$ 2 51 26 1 E
82$$ F5 T
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
56$$ F0 T
END
=ORIGENS
0$$ A4 21 A8 26 A10 51 71 E
1$$ 1 1T

COOLING TO 10 YEARS AND LIGHT ELEMENT GAMMA REBIN
3$$ 21 0 1 A33 -86 E
54$$ A8 1 E T
35$$ 0 T
56$$ 0 6 A13 -2 5 3 E
57** 4.0 E T
COOLING TO 10 YEARS AND LIGHT ELEMENT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0
65$$ A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61** F1E-6
81$$ 2 51 26 1 E
82$$ F4 T
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
56$$ F0 T
END
```

Figure 4.5-2 BWR SAS2H Input File for Design Basis Fuel Source Terms

```
=SAS2H    PARM=(HALT03,SKIPSHIPDATA)
GE/4-6 9x9-2 1.90 W/O U235 50,000 MWD/MTU, 40% VOID, 5-15 YEARS COOLING
27GROUPNDF4 LATTICECELL
UO2    1 0.95 840 92235 1.9 92238 98.1 END
ZIRCALLOY 2 1.0 620. END
H2O 3 DEN=0.446 1.0 562. END
H2O 4 DEN=0.743 1.0 553. END
ZIRCALLOY 5 1.0 553 END
H2O 6 DEN=0.446 1.0 562. END
END COMP
SQUAREPITCH 1.44 0.955 1 3 1.120 2 0.978 0 END
NPIN/ASSM=79 FUELENGTH=381.00 NCYCLES=3 NLIB/CYC=1 PRINTLEVEL=6
LIGHTEL=5 INPLEVEL=2 NUMZONES=7 END
 4 0.692 5 0.792 6 1.149 500 7.313 6 7.564 5 7.793 4 8.598
POWER=4.95 BURN=666.3 DOWN=60 END
POWER=4.95 BURN=666.3 DOWN=60 END
POWER=4.95 BURN=666.3 DOWN=1461 END
FE 0.672 CR 0.190 NI 0.115 MN 0.020 CO 0.0012
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING TO 10 YEARS AND FISSION PRODUCT GAMMA REBIN
3$$  21 0 1 A33 -86  E
54$$  A8 1 E T
35$$  0 T
56$$  0 6 A13 -2 5 3 E
57**  4.0 E T
COOLING TO 10 YEARS AND FISSION PRODUCT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60**  5.0 6.0 7.0 8.0 9.0 10.0
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F1E-6
81$$  2 51 26 1 E
82$$  F6 T
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
56$$ F0  T
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING TO 10 YEARS AND ACTINIDE GAMMA REBIN
3$$  21 0 1 A33 -86  E
54$$  A8 1 E T
35$$  0 T
```

Figure 4.5-2 BWR SAS2H Input File for Design Basis Fuel Source Terms (Continued)

```
56$$ 0 6 A13 -2 5 3 E
57** 4.0 E T
COOLING TO 10 YEARS AND ACTINIDE GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0
65$$ A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61** F1E-6
81$$ 2 51 26 1 E
82$$ F5 T
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
56$$ F0 T
END
=ORIGENS

0$$ A4 21 A8 26 A10 51 71 E
1$$ 1 1T
COOLING TO 10 YEARS AND LIGHT ELEMENT GAMMA REBIN
3$$ 21 0 1 A33 -86 E
54$$ A8 1 E T
35$$ 0 T
56$$ 0 6 A13 -2 5 3 E
57** 4.0 E T
COOLING TO 10 YEARS AND LIGHT ELEMENT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0
65$$ A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61** F1E-6
81$$ 2 51 26 1 E
82$$ F4 T
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
56$$ F0 T
END
```

Table 4.5.3-1 B&W 15x15 SAS2H Output and Group A_2 Values (Gas)

Isotope	Fuel Gas Activities / Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2 Value	Isotope Fraction/A_2	Group A_2
H3	2.70E+02	7.24E-02	1.08E+03	6.70E-05	
I129	2.48E-02	6.65E-06	1.00E+60	6.65E-66	
KR 85	3.46E+03	9.28E-01	2.70E+02	3.44E-03	
Total	3.73E+03			3.50E-03	2.86E+02

Table 4.5.3-2 B&W 15x15 SAS2H Output and Group A_2 Values (Volatiles)

Isotope	Fuel Volatiles/ Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2 Value	Isotope Fraction/A_2	Group A_2
SR 90	3.56E+04	2.39E-01	2.70E+00	8.90E-02	
CS134	2.87E+04	1.93E-01	1.40E+01	1.43E-02	
CS135	2.37E-01	1.59E-06	2.43E+01	6.55E-08	
CS137	6.90E+04	4.63E-01	1.40E+01	3.43E-02	
RU106	1.56E+04	1.05E-01	5.41E+00	1.94E-02	
Total	1.49E+05			1.57E-01	6.39E+00

Table 4.5.3-3 B&W 15x15 SAS2H Output and Group A_2 Values (Fuel Fines)

Isotope	Fuel Fine Activities/Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2 Value	Isotope Fraction/A_2	Group A_2
TL208	1.17E-02	1.17E-02	5.13E-08	5.00E-01	
PB212	3.25E-02	3.25E-02	1.43E-07	8.11E+00	
BI212	3.25E-02	3.25E-02	1.43E-07	8.11E+00	
PO212	2.08E-02	2.08E-02	9.12E-08	5.41E-04	
PO216	3.25E-02	3.25E-02	1.43E-07	5.41E-04	
RN220	3.25E-02	3.25E-02	1.43E-07	5.41E-04	
RA224	3.25E-02	3.25E-02	1.43E-07	1.62E+00	
TH228	3.24E-02	3.24E-02	1.42E-07	1.08E-02	
TH234	1.51E-01	1.51E-01	6.62E-07	5.41E+00	
PA233	1.88E-01	1.88E-01	8.25E-07	2.43E+01	
PA234M	1.51E-01	1.51E-01	6.62E-07	5.00E-01	
U232	4.74E-02	4.74E-02	2.08E-07	8.11E-03	
U234	4.84E-02	4.84E-02	2.12E-07	2.70E-02	
U236	7.76E-02	7.76E-02	3.40E-07	2.70E-02	
U237	1.53E+00	1.53E+00	6.71E-06	5.00E-01	
U238	1.51E-01	1.51E-01	6.62E-07	1.00E+60	
NP235	1.26E-03	1.26E-03	5.53E-09	1.08E+03	
NP237	1.88E-01	1.88E-01	8.25E-07	5.41E-03	
NP238	2.50E-02	2.50E-02	1.10E-07	5.00E-01	
NP239	4.68E+01	4.68E+01	2.05E-04	1.35E+01	
PU236	3.92E-01	3.92E-01	1.72E-06	1.89E-02	
PU238	2.75E+03	2.75E+03	1.21E-02	5.41E-03	
PU239	1.46E+02	1.46E+02	6.40E-04	5.41E-03	
PU240	3.09E+02	3.09E+02	1.36E-03	5.41E-03	
PU241	6.40E+04	6.40E+04	2.81E-01	2.70E-01	
PU242	2.77E+00	2.77E+00	1.22E-05	5.41E-03	
AM241	6.48E+02	6.48E+02	2.84E-03	5.41E-03	
AM242M	5.55E+00	5.55E+00	2.43E-05	5.41E-03	

Table 4.5.3-3 B&W 15x15 SAS2H Output and Group A_2 Values (Fuel Fines) (Continued)

Isotope	Fuel Fine Activities/Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2 Value	Isotope Fraction/A_2	Group A_2
AM242	5.53E+00	5.53E+00	2.43E-05	5.00E-01	
AM243	4.68E+01	4.68E+01	2.05E-04	5.41E-03	
CM242	2.43E+01	2.43E+01	1.07E-04	2.70E-01	
CM243	2.89E+01	2.89E+01	1.27E-04	8.11E-03	
CM244	9.19E+03	9.19E+03	4.03E-02	1.08E-02	
CM245	1.07E+00	1.07E+00	4.69E-06	5.41E-03	
CM246	7.48E-01	7.48E-01	3.28E-06	5.41E-03	
BK249	7.50E-03	7.50E-03	3.29E-08	2.16E+00	
CF249	1.07E-03	1.07E-03	4.69E-09	5.41E-03	
CF250	4.55E-03	4.55E-03	2.00E-08	1.35E-02	
CF252	7.41E-03	7.41E-03	3.25E-08	2.70E-02	
SE 79	4.52E-02	4.52E-02	1.98E-07	5.41E+01	
Y 90	3.56E+04	3.56E+04	1.56E-01	5.41E+00	
ZR 93	7.17E-01	7.17E-01	3.14E-06	5.41E+00	
NB 93M	1.93E-01	1.93E-01	8.47E-07	1.62E+02	
NB 95	3.82E-03	3.82E-03	1.68E-08	2.70E+01	
TC 99	8.68E+00	8.68E+00	3.81E-05	2.43E+01	
RH102	4.89E-01	4.89E-01	2.14E-06	1.35E+01	
RH106	1.56E+04	1.56E+04	6.84E-02	5.00E-01	
PD107	1.34E-01	1.34E-01	5.88E-07	1.00E+60	
AG108M	1.41E-02	1.41E-02	6.18E-08	1.62E+01	
AG110	5.19E-01	5.19E-01	2.28E-06	5.00E-01	
AG110M	3.82E+01	3.82E+01	1.68E-04	1.08E+01	
CD113M	2.85E+01	2.85E+01	1.25E-04	2.43E+00	
SN119M	6.34E-01	6.34E-01	2.78E-06	1.08E+03	
SN121	1.65E+00	1.65E+00	7.24E-06	5.00E-01	
SN121M	2.12E+00	2.12E+00	9.30E-06	2.43E+01	
SN123	1.72E-02	1.72E-02	7.54E-08	1.35E+01	

Table 4.5.3-3 B&W 15x15 SAS2H Output and Group A_2 Values (Fuel Fines) (Continued)

Isotope	Fuel Fine Activities / Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2 Value	Isotope Fraction/A_2	Group A_2
SB125	1.79E+03	1.79E+03	7.85E-03	2.43E+01	
TE125M	4.37E+02	4.37E+02	1.92E-03	2.43E+02	
SN126	5.24E-01	5.24E-01	2.30E-06	8.11E+00	
SB126	7.34E-02	7.34E-02	3.22E-07	1.08E+01	
SB126M	5.24E-01	5.24E-01	2.30E-06	1.35E+01	
TE127	7.90E-02	7.90E-02	3.47E-07	1.35E+01	
TE127M	8.07E-02	8.07E-02	3.54E-07	1.35E+01	
BA137M	6.52E+04	6.52E+04	2.86E-01	5.00E-01	
CE144	6.14E+03	6.14E+03	2.69E-02	5.41E+00	
PR144	6.14E+03	6.14E+03	2.69E-02	5.00E-01	
PR144M	8.59E+01	8.59E+01	3.77E-04	5.00E-01	
PM145	1.01E-01	1.01E-01	4.43E-07	1.89E+02	
SM145	2.63E-02	2.63E-02	1.15E-07	5.41E+02	
PM146	1.96E+00	1.96E+00	8.60E-06	5.00E-01	
PM147	1.95E+04	1.95E+04	8.55E-02	2.43E+01	
SM151	1.94E+02	1.94E+02	8.51E-04	1.08E+02	
EU152	1.77E+00	1.77E+00	7.76E-06	2.43E+01	
GD153	8.59E-02	8.59E-02	3.77E-07	1.35E+02	
EU154	7.26E+03	7.26E+03	3.18E-02	1.35E+01	
EU155	3.37E+03	3.37E+03	1.48E-02	5.41E+01	
Total	2.28E+05				0.114

Table 4.5.3-4 GE9x9 SAS2H Output and Group A_2 Values (Gas)

Isotope	Fuel Gas Activities / Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2	Isotope Fraction/A_2	Group A_2
H 3	1.06E+02	7.28E-02	1.08E+03	6.74E-05	
I 129	1.01E-02	6.94E-06	1.00E+60	6.94E-66	
KR 85	1.35E+03	9.27E-01	2.70E+02	3.43E-03	
Total	1.46E+03			3.50E-03	2.86E+02

Table 4.5.3-5 GE9x9 SAS2H Output and Group A_2 Values (Volatiles)

Isotope	Fuel Volatiles Activities / Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2	Isotope Fraction/A_2	Group A_2
SR90	1.44E+04	2.50E-01	2.70E+00	9.26E-02	
CS134	1.03E+04	1.79E-01	1.35E+01	1.32E-02	
CS135	1.10E-01	1.91E-06	2.43E+01	7.86E-08	
CS137	2.79E+04	4.84E-01	1.35E+01	3.59E-02	
RU106	5.00E+03	8.68E-02	5.41E+00	1.60E-02	
Total	5.76E+04			1.58E-01	6.34E+00

Table 4.5.3-6 GE9x9 SAS2H Output and Group A_2 Values (Fuel Fines)

Isotope	Fuel Fines Activities/ Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2 Value	Isotope Fraction/A_2	Group A_2
TL208	4.09E-03	4.64E-08	5.00E-01	9.28E-08	
PB212	1.14E-02	1.29E-07	8.11E+00	1.59E-08	
BI212	1.14E-02	1.29E-07	8.11E+00	1.59E-08	
PO212	7.29E-03	8.27E-08	5.41E-04	1.53E-04	
PO216	1.14E-02	1.29E-07	5.41E-04	2.39E-04	
RN220	1.14E-02	1.29E-07	5.41E-04	2.39E-04	
RA224	1.14E-02	1.29E-07	1.62E+00	7.98E-08	
TH228	1.13E-02	1.28E-07	1.08E-02	1.19E-05	
TH234	6.22E-02	7.05E-07	5.41E+00	1.30E-07	
PA233	7.03E-02	7.97E-07	2.43E+01	3.28E-08	
PA234M	6.22E-02	7.05E-07	5.00E-01	1.41E-06	
U232	1.61E-02	1.83E-07	8.11E-03	2.25E-05	
U234	1.97E-02	2.23E-07	2.70E-02	8.28E-06	
U236	3.20E-02	3.63E-07	2.70E-02	1.34E-05	
U237	5.20E-01	5.90E-06	5.00E-01	1.18E-05	
U238	6.22E-02	7.05E-07	1.00E+60	7.05E-67	
NP235	3.69E-04	4.19E-09	1.08E+03	3.88E-12	
NP237	7.03E-02	7.97E-07	5.41E-03	1.47E-04	
NP238	9.83E-03	1.11E-07	5.00E-01	2.23E-07	
NP239	1.83E+01	2.08E-04	1.35E+01	1.54E-05	
PU236	1.22E-01	1.38E-06	1.89E-02	7.32E-05	
PU238	1.04E+03	1.18E-02	5.41E-03	2.18E+00	
PU239	5.04E+01	5.72E-04	5.41E-03	1.06E-01	
PU240	1.28E+02	1.45E-03	5.41E-03	2.68E-01	
PU241	2.17E+04	2.46E-01	2.70E-01	9.12E-01	
PU242	1.17E+00	1.33E-05	5.41E-03	2.45E-03	
AM241	2.26E+02	2.56E-03	5.41E-03	4.74E-01	
AM242M	2.18E+00	2.47E-05	5.41E-03	4.57E-03	

Table 4.5.3-6 GE9x9 SAS2H Output and Group A_2 Values (Fuel Fines) (Continued)

Isotope	Fuel Fines Activities/ Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2 Value	Isotope Fraction/A_2	Group A_2
AM242	2.17E+00	2.46E-05	5.00E-01	4.92E-05	
AM243	1.83E+01	2.08E-04	5.41E-03	3.84E-02	
CM242	8.97E+00	1.02E-04	2.70E-01	3.77E-04	
CM243	1.03E+01	1.17E-04	8.11E-03	1.44E-02	
CM244	3.70E+03	4.20E-02	1.08E-02	3.89E+00	
CM245	3.64E-01	4.13E-06	5.41E-03	7.63E-04	
CM246	3.14E-01	3.56E-06	5.41E-03	6.58E-04	
BK249	2.92E-03	3.31E-08	2.16E+00	1.53E-08	
CF249	4.30E-04	4.88E-09	5.41E-03	9.01E-07	
CF250	2.00E-03	2.27E-08	1.35E-02	1.68E-06	
CF252	3.56E-03	4.04E-08	2.70E-02	1.50E-06	
SE 79	1.86E-02	2.11E-07	5.41E+01	3.90E-09	
Y 90	1.44E+04	1.63E-01	5.41E+00	3.02E-02	
ZR 93	2.96E-01	3.36E-06	5.41E+00	6.21E-07	
NB 93M	8.75E-02	9.92E-07	1.62E+02	6.13E-09	
TC 99	3.53E+00	4.00E-05	2.43E+01	1.65E-06	
RH102	1.71E-01	1.94E-06	1.35E+01	1.44E-07	
RH106	5.00E+03	5.67E-02	5.00E-01	1.13E-01	
PD107	5.46E-02	6.19E-07	1.00E+60	6.19E-67	
AG108M	5.45E-03	6.18E-08	1.62E+01	3.82E-09	
AG110	1.74E-01	1.97E-06	5.00E-01	3.95E-06	
AG110M	1.28E+01	1.45E-04	1.08E+01	1.34E-05	
CD113M	1.17E+01	1.33E-04	2.43E+00	5.46E-05	
SN119M	2.06E-01	2.34E-06	1.08E+03	2.16E-09	
SN121	6.70E-01	7.60E-06	5.00E-01	1.52E-05	
SN121M	8.63E-01	9.79E-06	2.43E+01	4.03E-07	
SN123	5.06E-03	5.74E-08	1.35E+01	4.25E-09	

Table 4.5.3-6 GE9x9 SAS2H Output and Group A_2 Values (Fuel Fines) (Continued)

Isotope	Fuel Fine Activities/ Assembly 5-Yr Cool Time (Ci)	Fraction of Source	Isotope A_2 Value	Isotope Fraction/A_2	Group A_2
SB125	6.33E+02	7.18E-03	2.43E+01	2.95E-04	
TE125M	1.55E+02	1.76E-03	2.43E+02	7.23E-06	
SN126	2.15E-01	2.44E-06	8.11E+00	3.01E-07	
SB126	3.01E-02	3.41E-07	1.08E+01	3.16E-08	
SB126M	2.15E-01	2.44E-06	1.35E+01	1.81E-07	
TE127	2.34E-02	2.65E-07	1.35E+01	1.97E-08	
TE127M	2.38E-02	2.70E-07	1.35E+01	2.00E-08	
BA137M	2.63E+04	2.98E-01	5.00E-01	5.97E-01	
CE144	1.82E+03	2.06E-02	5.41E+00	3.82E-03	
PR144	1.82E+03	2.06E-02	5.00E-01	4.13E-02	
PR144M	2.55E+01	2.89E-04	5.00E-01	5.78E-04	
PM145	4.96E-02	5.63E-07	1.89E+02	2.98E-09	
SM145	1.05E-02	1.19E-07	5.41E+02	2.20E-10	
PM146	6.55E-01	7.43E-06	5.00E-01	1.49E-05	
PM147	6.94E+03	7.87E-02	2.43E+01	3.24E-03	
SM151	6.64E+01	7.53E-04	1.08E+02	6.97E-06	
EU152	7.48E-01	8.48E-06	2.43E+01	3.49E-07	
GD153	4.02E-02	4.56E-07	1.35E+02	3.38E-09	
EU154	2.84E+03	3.22E-02	1.35E+01	2.39E-03	
EU155	1.23E+03	1.40E-02	5.41E+01	2.58E-04	
Total	8.82E+04			8.68E+00	1.15E-01

THIS PAGE INTENTIONALLY LEFT BLANK

4.6 References

1. Title 10 of the Code of Federal Regulations, Part 71 (10 CFR 71), "Packaging and Transportation of Radioactive Materials," April 1996.

2. IAEA Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Materials," International Atomic Energy Agency, Vienna, Austria, 1985 Edition, as amended 1990.

3. ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers, 1995 Edition with 1995 Addenda.

4. ANSI N14.5-1997, "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," American National Standards Institute, December 1997.

5. NUREG/CR-6487 UCRL-ID-124822, Anderson, B.L., Carlson, R.W., and Fisher, L.E., "Containment Analysis for Type B Packages Used to Transport Various Contents," Lawrence Livermore National Laboratory, 1996.

6. "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," O.W. Hermann and C.V. Parks, ORNL/NUREG/CSD-2/V1/R5, Volume 1, Section S2, September 1995.

THIS PAGE INTENTIONALLY LEFT BLANK

Table of Contents

Table of Contents (Continued)

List of Figures

List of Figures (Continued)

List of Tables

List of Tables (Continued)

List of Tables (Continued)

List of Tables (Continued)

List of Tables (Continued)

THIS PAGE INTENTIONALLY LEFT BLANK

5.0 SHIELDING EVALUATION

The Universal Transport Cask meets the 10 CFR 71 [1] requirements for transportation dose rate limits. The optimized multiwall design provides an efficient shielding arrangement for the transportation of 24 PWR or 56 BWR spent fuel assemblies. This chapter describes the shielding design and the analysis used to establish bounding radiological dose rates for the transport of various PWR and BWR fuels.

The shielding design criteria for the Universal Transport Cask are in accordance with the requirements established in 10 CFR Parts 71.47 and 71.51 for normal conditions of transport and hypothetical accident conditions. The 10 CFR 71.47 requirements for the transport of spent fuel under normal conditions of transport include the following:

- The dose rate on the external surface of the cask must not exceed 200 mrem/hr.
- The dose rate on a plane two meters from the lateral surfaces of the railcar must not exceed 10 mrem/hr.
- The dose rate in any normally occupied positions of the railcar must not exceed 2 mrem/hr.

The 10 CFR 71.51 requirements state that the dose rate under hypothetical accident conditions must not exceed 1,000 mrem/hr at 1 meter from the surface of the cask. A summary description of the modeling methodology and dose rate results is provided in Section 5.1.

The shielding analysis is performed on the basis of design basis fuel descriptions for both PWR and BWR fuel. The design basis PWR fuel is a Westinghouse 17×17 assembly with a burnup of 45,000 MWD/MTU, an initial enrichment of 3.7 wt % ^{235}U, and a 10-year cooling time. The design basis BWR fuel is a GE 9×9 assembly design with a burnup of 40,000 MWD/MTU, an initial enrichment of 3.25 wt % ^{235}U, and a 10-year cooling time. A detailed description of the source term specification is provided in Section 5.2.

In the Universal Transport Cask design, the spent fuel assemblies are surrounded by a multiwalled arrangement of shielding materials. However, structural design requirements lead to cask extremity regions, such as rotation pockets, in which shield materials are reduced or

penetrated. Detailed analytical treatment of these shield transition regions is required to assess the radiological consequences of the design. Section 5.3 describes the three-dimensional shielding models employed in this analysis.

Dose rate results are obtained for both normal conditions of transport and hypothetical accident conditions. Under accident conditions, the cask is analyzed for the simultaneous effects of complete loss of radial neutron shielding including loss of the outer neutron shield shell; loss of impact limiters; and combined radial and axial lead slumps resulting from postulated cask side and end drops. Analytical details and dose rate results are given in Section 5.4. Under all postulated conditions, the fully loaded cask is shown to meet regulatory radiological limits.

The design basis fuel descriptions characterize Universal Transport Cask dose and heat generation rates at nominal conditions of burnup and initial enrichment for a particular PWR and BWR fuel type. In order to extend the results to other fuel types and various combinations of initial enrichment and burnup, a detailed analysis is conducted that determines, for any given fuel type, enrichment, and burnup combination, the cool time required for radiation dose and heat generation rates to fall below the design basis values. The analysis explicitly models the source term for the various fuel combinations based on SAS2H results, and the dose rate evaluation is made on the basis of computed one-dimensional dose rates which explicitly consider the effects of radiation spectrum and cask shielding properties rather than a simple source rate comparison.

For clarity, the various fuel assembly types considered are classified according to array size, with the results for the most limiting fuel type within each array size taken as the minimum required cool time for all assemblies within that size. The result of the analysis is a fuel assembly loading table showing the required cool time for any combination of fuel array size, initial enrichment, and burnup. Results are summarized in Section 5.1.3.

An analysis is performed for fuel at various combinations of burnup and initial enrichment to determine the required cool time for the fuel to fall below the design basis dose and decay heat levels. The bounding cool time is determined considering the cask normal and accident condition dose rates and total decay in the cask. This analysis is presented in Section 5.4.3.

Site specific fuel and GTCC waste is evaluated in Section 5.5.1.

5.1 Discussion and Results

The radiation protection provided by the Universal Transport Cask is in the form of solid multiwalled shielding materials that completely surround the fuel. These shielding materials include steel and lead for gamma shielding and a borated polymer (NS-4-FR) for neutron shielding. The multiwalled arrangement of steel and lead in the cask provides an optimal gamma shield. The NS-4-FR neutron shielding material, with a hydrogen density close to that of water, moderates neutrons and facilitates their capture in boron.

The multiwalled shielding arrangement is employed in both radial and axial shields. The configuration of the radial gamma shielding in the cask body is a 2.0 in. thick stainless steel inner shell and a 2.75 in. thick stainless steel outer shell with a 2.75 in. lead-filled annulus between the two. The radial neutron shield is arranged around the outer steel shell with a 4.5 in. thick NS-4-FR layer which is covered by a 0.25 in. thick stainless steel neutron shield shell. The bottom of the cask contains a steel/NS-4-FR/steel shield arrangement in which the two stainless steel components provide a total of 9.25 in. of gamma shielding with a 1 in. layer of NS-4-FR neutron shielding between them. The top of the cask provides additional shielding in the form of a 6.50 in. thick stainless steel closure lid.

The Universal Transport Cask is designed to accommodate a variety of fuel assembly types. Because assembly lengths vary, fuel assemblies are segregated into five distinct classes, three for PWR fuel and two for BWR fuel. A particular canister design is associated with each class, and for each reactor type, the canister designs differ only in their internal cavity length and basket structure. To accommodate the shorter canister classes, a spacer assembly is placed beneath the canister to ensure proper canister positioning within the cask cavity. For each reactor type, the shielding analysis develops bounding dose rates for a specific design basis fuel assembly type and its associated canister.

These design basis fuel assemblies are determined on the basis of the results of one-dimensional shielding analyses conducted for a number of candidate assembly designs. These candidate assembly types are selected on the basis of their radiation source terms as determined by detailed SAS2H isotopic depletion calculations performed for each assembly. Representative PWR and BWR fuel assembly designs are selected for the source term analysis on the basis of initial uranium loading and non-fuel hardware compositions and masses. Hence, the following analytical procedure is used to identify the design basis PWR and BWR fuel assemblies. First,

among all assembly types for which the Universal Transport Cask is intended, those assemblies expected to have the highest source terms are first identified on the basis of initial loading of heavy metal and other factors. Then, detailed source descriptions of these assemblies are developed by using the SCALE SAS2H code [2]. To identify the bounding PWR and BWR assemblies, the resulting source descriptions are employed in one-dimensional shielding calculations. Full three-dimensional analysis of the resulting design basis assemblies is then conducted to establish licensing basis dose rates.

5.1.1 Fuel Assembly Classification

5.1.1.1 PWR Fuel Assembly Classes

The Universal Transport Cask is designed to transport up to 24 intact PWR design basis fuel assemblies. As discussed in Chapters 1.0 and 6.0 of this SAR, the PWR fuel assemblies to be transported in the cask are compiled into three classes on the basis of similarity of their lengths. Of the PWR assemblies to be transported in the cask, the following four are identified as representative assemblies on the basis of their computed radiation source terms:

- Westinghouse 15×15 Std (Class 1)
- Westinghouse 17×17 Std (Class 1)
- Babcock & Wilcox 15×15 Mark B (Class 2)
- Combustion Engineering 16×16 System 80 (Class 3)

These assemblies constitute the candidate design basis PWR fuel assemblies for the shielding analysis of the cask. One-dimensional shielding calculations are performed for each to identify a single assembly type, which is then selected as the design basis assembly for a subsequent detailed three-dimensional shielding analysis. The candidate PWR fuel assemblies are analyzed on the basis of an assumed initial enrichment of 3.7 wt % ^{235}U, a burnup of 45,000 MWD/MTU, and a cooling time of 10 years. The initial enrichment assumed in the shielding analysis is significantly less than the criticality analysis design basis value of 4.2 wt % so that the calculated neutron source rate bounds that of lower enrichment fuel which may reach the design basis burnup of 45,000 MWD/MTU. This assumption produces a neutron source that is 29% higher than that calculated assuming a 4.2 wt % ^{235}U initial enrichment.

The source specifications for the design basis fuel are discussed in Section 5.2.

5.1.1.2 BWR Fuel Assembly Classes

On the basis of similarity of length, the BWR fuel assemblies to be transported in the Universal Transport Cask are compiled into two classes (Class 4 corresponds to BWR/2–3 assemblies and Class 5 corresponds to BWR/4–6 assemblies). Of these assemblies, the following are chosen as candidate design basis assemblies for the cask shielding analysis on the basis of their computed radiation source terms:

- GE 7×7 BWR/2–3 version GE-2b (Class 4)
- GE 8×8–2 BWR/2–3 version GE-5 (Class 4)
- GE 8×8–4 BWR/2–3 version GE-8 (Class 4)
- GE 7×7 BWR/4–6 version GE-2 (Class 5)
- GE 8×8–2 BWR/4–6 version GE-5 (Class 5)
- GE 8×8–4 BWR/4–6 version GE-10 (Class 5)
- GE 9×9–2 BWR/4–6 version GE-11 (Class 5)

These assemblies constitute the candidate design basis BWR fuel assemblies for the cask shielding analysis. One-dimensional shielding calculations performed for each assembly identify a single assembly type that is selected as the design basis assembly for subsequent detailed three-dimensional shielding analysis. The candidate BWR fuel assemblies are analyzed on the basis of an initial enrichment of 3.25 wt % ^{235}U, a burnup of 40,000 MWD/MTU, and a cooling time of 10 years.

5.1.2 Codes Employed

The SCALE 4.3PC [3] code system is used in the analysis of the Universal Transport Cask. Source terms are generated by using the SAS2H [4] sequence as described in Section 5.2. One-dimensional radial and axial SAS1 [5] analyses are performed in order to identify design basis PWR and BWR fuel types. With these design basis source descriptions, a detailed three-dimensional analysis is performed by using the SAS4 [2] Monte Carlo

shielding analysis sequence. Modifications to SAS4 permit computation of dose rate profiles along surface detectors. These changes are further described in Section 5.4.1.

The 27 group neutron, 18 group gamma, coupled cross section library (27N-18COUPLE) derived from ENDF/B-IV [6] data is used in all shielding evaluations. Source terms include fuel neutron, fuel gamma, and gamma contributions from activated hardware. The effects of subcritical neutron multiplication and secondary gamma production due to neutron capture are included in the analysis. Dose rate evaluations include the effect of axial fuel burnup variation on fuel neutron and gamma source terms as described in Section 5.2.6.

5.1.3 Results of Analysis

The calculated normal transport and hypothetical accident condition dose rates are discussed in the following sections. The cask surface dose rates calculated at the fuel midplane include (1) neutrons and gammas originating from the fuel; (2) neutrons resulting from subcritical multiplication; (3) secondary gammas resulting from neutron capture in the neutron shield; and (4) gammas from activated materials in the grid spacers, end-fittings, and plenum springs.

The results of the shielding analysis demonstrate that computed dose rates remain below 200 mrem/hr at all accessible locations on the surface of the cask. Computed dose rates slightly above 200 mrem/hr occur in the narrow, inaccessible gap between the neutron shield and the lower impact limiter in the BWR case and in an analysis of the PWR case with no spacer employed. The dose rates remain below 10 mrem/hr at all locations 2 m from the edge of the railcar (any point 2 m from the vertical planes projected from the outer edges of the conveyance), 2 m above or below the Transport Cask, and 2 m from the axial surfaces of the impact limiters.

In addition, the hypothetical accident conditions do not result in a dose rate that exceeds the accident condition dose rate limit of 1000 mrem/hr at 1 m from the cask surface. Therefore, the Universal Transport Cask satisfies the regulatory criteria of 10 CFR 71.47 and paragraph 469 of IAEA Safety Series No. 6 under normal conditions of transport; and 10 CFR 71.51(a) and paragraph 542 of IAEA Safety Series No. 6 for hypothetical accident conditions.

In the tabulated results, computed dose rates are reported along with the relative uncertainty associated with each computed value expressed as a percentage. The relative uncertainty corresponds to plus or minus one standard deviation in the quoted value.

5.1.3.1 Normal Conditions of Transport

The maximum radial and axial dose rates calculated for the PWR and BWR casks under normal conditions of transport are shown in Table 5.1-1 and Table 5.1-3, respectively. The locations of the maximum dose rates in normal conditions of transport relative to the cask body and transporter are shown in Figure 5.1-1. The tables present the maximum computed dose rates and corresponding relative uncertainties on radial and axial surfaces outside the cask. Dose rates indicated at the personnel barrier correspond to the maximum values computed on a cylindrical surface extending between the top and bottom impact limiters and surrounding the cask at a radius of 53.5 in. In the radial case, dose rates indicated at the "2 m position" correspond to a position 2 m from the edge of a 124 in. wide standard railcar. For axial results, this position corresponds to a dose location 2 m from the top or bottom impact limiter surface.

For the PWR cask, the maximum normal conditions surface dose rate is 167.2 (±1.8%) mrem/hr, occurring on the surface of the upper forging at the upper trunnion recess. (Values in parentheses following a dose rate result indicate the relative uncertainty in the value.) At the surface of the personnel barrier, the dose rate is much less than 200 mrem/hr with a maximum computed dose rate of 46.6 (±1.6%) mrem/hr. In addition, the 10 mrem/hr criterion is met at all locations 2 m from the railcar, 2 m above or below the cask, and 2 m from the axial surfaces of the impact limiters.

For the BWR cask, the maximum normal conditions surface dose rate is 84.8 (±2.0%) mrem/hr, occurring on the surface of the lower rotation pocket. The dose rate at the outer shell surface in the inaccessible 1.25-in. wide gap between the neutron shield shell and lower impact limiter, is computed to be 225.6 (±3.5%) mrem/hr. However, this dose rate is not considered significant due to the inaccessibility of the location. Furthermore, the dose rate at the gap opening is well below 200 mrem/hr as demonstrated in Section 5.4.2.2. All other cask surface dose rates are less than 200 mrem/hr. At the personnel barrier, the maximum dose rate is much less than 200 mrem/hr, with a maximum computed dose rate of 40.4 (±1.5%) mrem/hr. The 10 mrem/hr criterion is met at all locations 2 m from the railcar and from the axial surfaces of the impact limiters.

5.1.3.2 Hypothetical Accident Conditions

Table 5.1-2 and Table 5.1-4 provide accident dose rates that could occur in the event of the loss of the neutron shield, shield shell, and impact limiters in the PWR and BWR casks, respectively. The location of the maximum hypothetical accident dose rates relative to the transport cask body are shown in Figure 5.1-2. An accident involving the complete loss of the impact limiters or neutron shielding is not credible for the Universal Transport Cask, although some of the neutron shielding capability may be lost as a result of a fire. Nonetheless, the shielding analysis conservatively assumes a complete loss of radial neutron shielding. In addition, axial and radial lead slumps resulting from postulated cask side and end drop accidents are modeled in the accident condition analysis.

In the event of a cask end drop, the lead gamma shielding could slump and fill the annular gap (if one exists) created by the cooling of the lead after fabrication. This accident could create a 3.05 in. gap at the top of the lead annulus. The major radiation concern in this event is the "shine through" of the activated end-fittings. If the cask is subjected to a side drop, the lead gamma shielding could slump and create a void on the upper side of the cask. An evaluation of this side drop accident shows that the lead may sag at the opposite side by a maximum 0.91 in.

The dose rates presented here and in more detail in Section 5.4 show that neither the loss of the neutron shielding nor the lead slump conditions will result in a dose rate that exceeds the hypothetical accident dose rate limit of 1000 mrem/hr at 1 m from the surface of the cask.

Figure 5.1-1 Location of Maximum Dose Rates for Normal Conditions of Transport

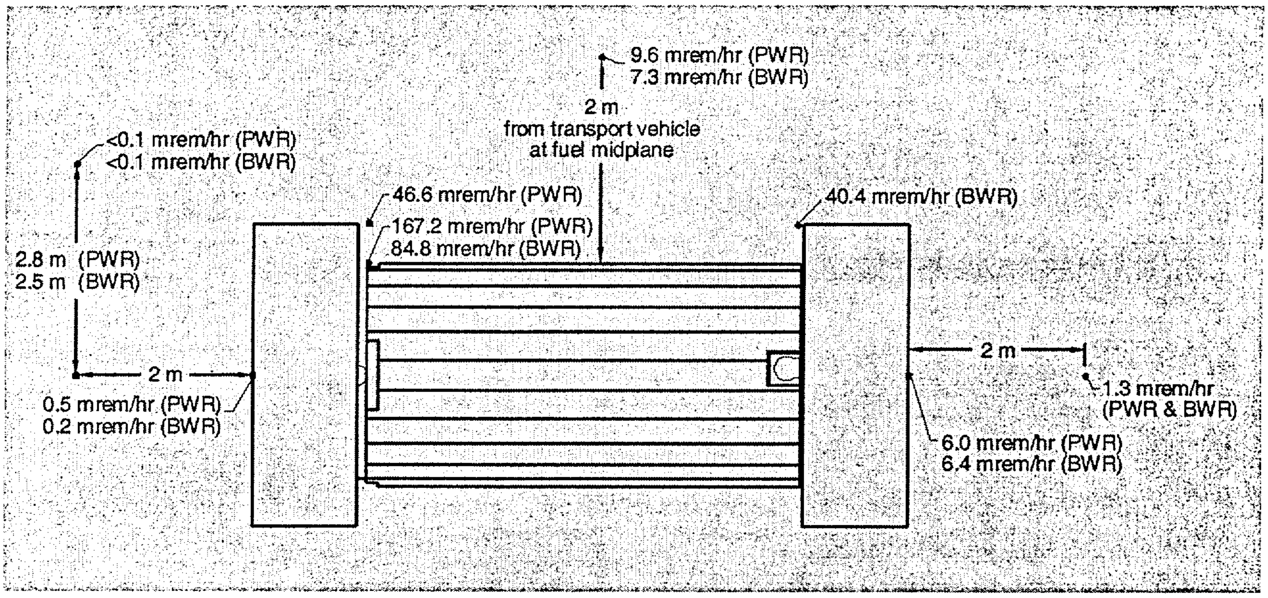


Figure 5.1-2 Location of Maximum Dose Rates for Hypothetical Accident Conditions

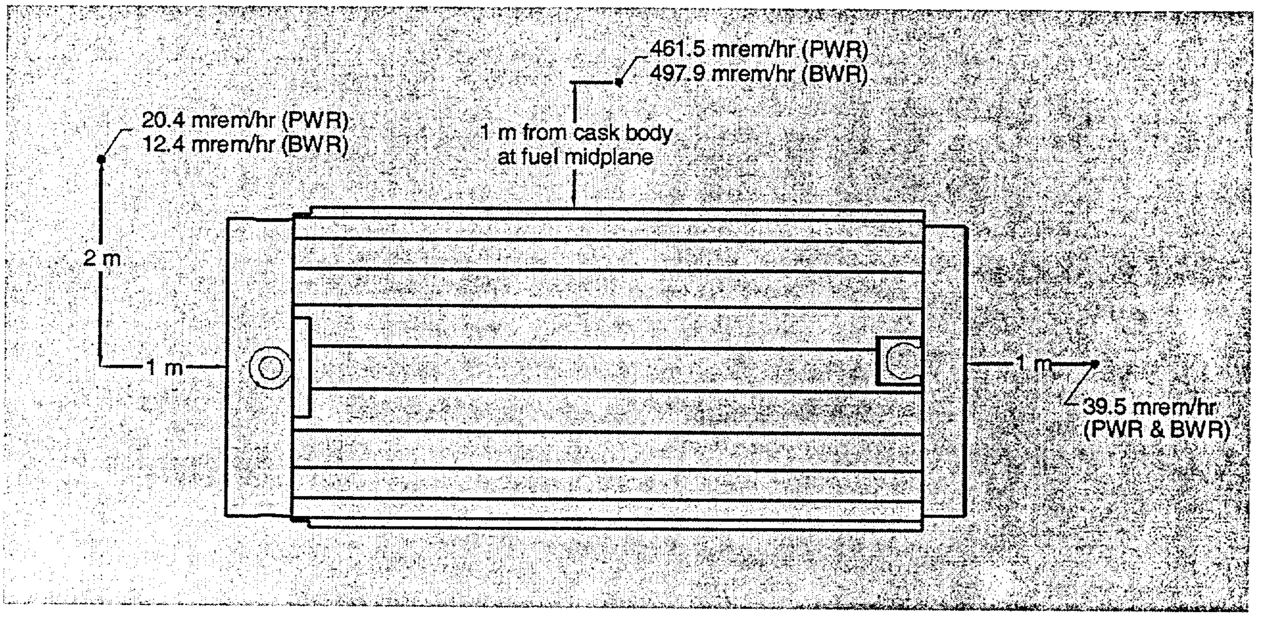


Table 5.1-1 PWR Maximum Total Dose Rate Summary – Normal Conditions [mrem/hr]

Location	Dose Type	Radial		Top Axial		Bottom Axial	
Surface	Gamma	66.2	(0.6%)	0.5	(2.0%)	6.0	(0.4%)
	Neutron	101.1	(2.9%)	<0.1	(1.4%)	<0.1	(1.7%)
	Total	167.2	(1.8%)	0.5	(1.9%)	6.0	(0.4%)
Personnel Barrier	Gamma	23.5	(0.5%)	-		-	
	Neutron	23.2	(3.1%)	-		-	
	Total	46.6	(1.6%)	-		-	
2 m Position	Gamma	6.4	(0.3%)	<0.1	(3.5%)	1.3	(1.0%)
	Neutron	3.2	(0.9%)	<0.1	(14.9%)	<0.1	(3.2%)
	Total	9.6	(0.4%)	<0.1	(11.4%)	1.3	(1.0%)

Table 5.1-2 PWR Maximum Total Dose Rate Summary – Accident Conditions [mrem/hr]

Location	Dose Type	Radial		Top Axial		Bottom Axial	
1 m Position	Gamma	44.8	(0.5%)	<0.1	(8.6%)	28.7	(0.4%)
	Neutron	416.7	(0.2%)	20.3	(2.9%)	10.9	(0.6%)
	Total	461.5	(0.2%)	20.4	(2.8%)	39.5	(0.4%)

Table 5.1-3 BWR Maximum Total Dose Rate Summary – Normal Conditions [mrem/hr]

Location	Dose Type	Radial		Top Axial		Bottom Axial	
Surface	Gamma	50.4	(1.1%)	0.2	(2.8%)	6.4	(0.4%)
	Neutron	34.5	(4.6%)	<0.1	(1.6%)	<0.1	(2.0%)
	Total	84.8 (1)	(2.0%)	0.2	(2.7%)	6.4	(0.4%)
Personnel Barrier	Gamma	20.7	(0.5%)	-		-	
	Neutron	19.8	(3.0%)	-		-	
	Total	40.4	(1.5%)	-		-	
2 m Position	Gamma	4.0	(0.4%)	<0.1	(3.9%)	1.3	(1.0%)
	Neutron	3.4	(0.8%)	<0.1	(24.1%)	<0.1	(4.3%)
	Total	7.3	(0.4%)	<0.1	(15.3%)	1.3	(1.0%)

(1) Dose rate at forging surface inside gap between lower limiter and shield shell is 225.6 (3.5%)

Table 5.1-4 BWR Maximum Total Dose Rate Summary – Accident Conditions [mrem/hr]

Location	Dose Type	Radial		Top Axial		Bottom Axial	
1 m Position	Gamma	25.3	(0.8%)	<0.1	(8.9%)	30.8	(0.4%)
	Neutron	472.7	(0.2%)	12.3	(3.9%)	8.7	(0.7%)
	Total	497.9	(0.2%)	12.4	(3.9%)	39.5	(0.4%)

5.2 Source Specification

In this section, the procedure used to identify a design basis fuel description for PWR and BWR fuel types is described. Each of the candidate fuel assemblies described in Section 5.1.1 is represented in one-dimensional radial and axial models of the fully loaded Transport Cask. The results of this one-dimensional shielding analysis are then used to identify a limiting fuel design for PWR and BWR fuel types.

The one-dimensional analysis is used only as a basis for identifying PWR and BWR design basis fuel descriptions. The detailed three-dimensional analysis described in Section 5.4.2 provides the licensing basis enveloping dose rates for the Transport Cask.

The SAS2H code sequence [4] is used to generate source terms for the shielding analysis. This code sequence is part of the SCALE 4.3 [3] code package for the PC. SAS2H includes an XSDRNPM [7] neutronics model of the fuel assembly and ORIGEN-S [8] fuel depletion and source terms calculations. Source terms are generated for both UO_2 fuel and fuel assembly hardware. The hardware activation is calculated by light element transmutation using the incore neutron flux spectrum produced by the SAS2H neutronics model. The hardware is assumed to be Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity. The effects of axial flux spectrum and magnitude variation on hardware activation are estimated by flux ratios determined from empirical data [9].

In the PWR case, consideration is made for the loading of activated burnable poison assemblies and thimble plugs. The additional hardware is assumed to be activated for the one reactor cycle immediately preceding fuel discharge.

5.2.1 Gamma Source

The fuel gamma source contains contributions from both fission products and actinides. The spectra are presented in the 18-group structure consistent with the SCALE 4.3 27N-18COUPLE cross section library. The hardware gamma spectra contain contributions primarily from ^{60}Co due to the activation of Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity and with minor contributions from ^{59}Ni and ^{58}Fe. The activated hardware spectrum is determined by the irradiation of 1 kg of stainless steel in the incore flux spectrum produced by the SAS2H

neutronics calculation. The source strength is determined by scaling this spectrum by both the mass of hardware present and an activation ratio as described below.

The activated fuel assembly hardware source terms are found by multiplying the source strength from 1 kg by the mass of steel or inconel material in the plenum, upper end fitting, or lower end fitting regions, and by then multiplying this result by a regional flux ratio. This regional flux ratio accounts for the effects of both magnitude and spectrum variation on hardware activation. These ratios are determined from empirical data [9]. A flux ratio of 0.2 is applied to hardware regions directly adjacent to the active core region, *e.g.*, upper and lower plenum (if present), and a flux ratio of 0.1 is applied to hardware regions once removed from the active core region, *e.g.*, upper and lower end fitting region.

The default gamma energy group spectrum of the ORIGEN-S code differs from that of the SCALE 27N18G library employed in the shielding analysis. To account for the differences in the energy group structure, the source spectra from the fuel and hardware gamma sources are rebinned to the SCALE 27N18G library structure by using the ORIGEN-S code directly. This method regroups the source on the basis of the actual gamma lines of each specific nuclide. A more accurate result is thus obtained than that achieved by simply rebinning the coarse energy group spectrum onto the SCALE 27N18G library group structure.

5.2.2 Neutron Source

Light water reactor (LWR) spent fuel neutron sources result from actinide spontaneous fission and from (α,n) reactions. The isotopes ^{242}Cm and ^{244}Cm characteristically produce all but a few percent of the spontaneous fission neutrons and (α,n) source in LWR fuel. The next largest contribution is from (α,n) reactions of ^{238}Pu with oxygen. The neutron spectra for each emission type are included in the ORIGEN-S nuclear data libraries of the SCALE 4.3 code package. The spectra are collapsed from the energy group structure of the data library into that of the SCALE 27-group neutron cross section library [6].

5.2.3 PWR Fuel Assembly Descriptions

The radiation source in the Universal Transport Cask consists of 24 design basis PWR spent fuel assemblies. The design basis PWR fuel has an average burnup of 45,000 MWD/MTU and an initial enrichment of 4.2 wt % ^{235}U. However, to bound the neutron source produced by lower

enrichment fuel which may achieve this burnup, the design basis PWR source terms are calculated with an initial enrichment of 3.7 wt % ^{235}U. This assumption produces a neutron source 29% higher than that obtained by assuming 4.2 wt % ^{235}U initial enrichment. Assembly power density and cycle parameters are selected such that the assembly is activated at a power level 10% greater than a typical PWR assembly to allow for assembly power peaking during core residence. This treatment results in conservatively higher source rates due to enhanced actinide production and a shorter activation period.

Source term spectra and source region elevations are determined for the following four major PWR fuel assembly types (see Table 5.2-1):

• Westinghouse 15×15 Std	Class 1
• Westinghouse 17×17 Std	Class 1
• Babcock & Wilcox 15×15 Mark B	Class 2
• Combustion Engineering 16×16 System 80	Class 3

These assembly types produce the limiting source terms. These assembly types are referred to in this report by the abbreviated names given in Table 5.2-1. Fuel assembly physical characteristics are given in Table 5.2-2, and hardware masses are given in Table 5.2-3. The results of the source term analysis for the fuel types given here are summarized in Table 5.2-4. Fuel assembly activated hardware source terms are shown in Table 5.2-5. These non-fuel source terms are determined on the basis of the hardware source per kilogram given in Table 5.2-4 and the hardware masses given in Table 5.2-3. The hardware activation is based on a stainless steel Type 304 composition with an assumed ^{59}Co impurity level of 1.2 g/kg.

In order to account for spectral differences in the activating neutron flux, a flux ratio of 0.2 is applied to hardware regions directly adjacent to the active core region, e.g., the lower end-fitting and upper plenum. A flux ratio of 0.1 is applied to the upper end-fitting region, except for the CE 16×16 upper end-fitting for which a 0.05 flux ratio is used. The lower end fitting region in the B&W fuel assembly model uses a 0.1 flux ratio since the model explicitly includes a lower plenum region adjacent to the fuel region. The ORIGEN-S code is used directly to calculate hardware activation spectra by activating the fuel assembly components in the SAS2H-calculated flux spectrum for each assembly type.

5.2.4 BWR Fuel Assembly Descriptions

The Universal Transport Cask can transport up to 56 intact BWR fuel assemblies. BWR fuel is analyzed on the basis of an initial enrichment of 3.25 wt % ^{235}U, 40,000 MWD/MTU burnup, and a post irradiation cooling time of 10 years. Assembly power density and cycle parameters are selected such that the assembly is activated at a power level 10% greater than a typical BWR assembly to allow for assembly power peaking during core residence. This treatment results in conservatively higher source rates due to enhanced actinide production and a shorter activation period.

Source term spectra and source region elevations are determined for the following major BWR fuel assembly types (see Table 5.2-1):

- GE 7×7 BWR/2-3 Reactor Type, Version GE-2b — Class 4
- GE 8×8 BWR/2-3 Reactor Type, Version GE-5, 2 water holes — Class 4
- GE 8×8 BWR/2-3 Reactor Type, Version GE-10, 1 large water hole — Class 4
- GE 7×7 BWR/4-6 Reactor Type, Version GE-2 — Class 5
- GE 8×8 BWR/4-6 Reactor Type, Version GE-5, 2 water holes — Class 5
- GE 8×8 BWR/4-6 Reactor Type, Version GE-10, 1 large water hole — Class 5
- GE 9×9 BWR/4-6 Reactor Type, Version GE-11, 2 water holes, 79 fuel rods — Class 5

These assembly types are referred to in this report by the abbreviated names given in Table 5.2-1. The physical characteristics of the two classes of BWR fuel are given in Table 5.2-6 and Table 5.2-7. For fuel assemblies with a burnup of 40,000 MWD/MTU, the fuel requires a minimum of 10 years of cooling after discharge to meet the neutron and gamma source values, and the decay heat values specified in Table 5.2-8 and Table 5.2-9. The GE BWR/2-3 8×8 fuel assembly designs are analyzed on the basis of a 144 in. active fuel length in order to provide a consistent basis for comparison with the other BWR/2-3 fuel assembly designs. The GE-2b version of the GE BWR/2-3 7x7 fuel assembly is selected over the older GE-2a design since it has been discharged more recently, although the GE-2a assembly has a marginally higher (0.4%) initial heavy metal loading.

5.2.5 Design Basis Fuel Assemblies

For the cask shielding analysis, the WE 17×17 and GE 9×9–2 fuel assembly types are selected as the design basis PWR and BWR fuel assemblies, respectively. These assembly designs are selected on the basis of the one-dimensional shielding analysis results presented in Table 5.2-10 and Table 5.2-11. To facilitate comparison, the results for each fuel assembly type are shown on a normalized basis relative to the design basis fuel assembly dose rates. With the exceptions discussed below, the computed dose rates vary over a narrow range.

In the PWR case, the one-dimensional radial model results at the cask surface and 2 m position show that the WE 17×17 assembly ranks second in both normal and accident conditions with dose rates 4% and 2% below the maximum values, respectively. These differences are well below the dose rate margins ultimately computed in the three-dimensional analysis. In addition, the WE 17×17 assembly shows the highest dose rates in the top and bottom axial configurations under both normal and accident conditions, except in the case of the CE 16×16 assembly discussed below.

Ten-year cooled source spectra for the PWR design basis WE 17×17 fuel assembly are shown in Table 5.2-12 through Table 5.2-14.

The CE 16×16 bottom axial one-dimensional dose rates are nearly double those of the design basis PWR results. This effect is due to the absence of a spacer assembly in the Class 3 canister to which the CE 16×16 fuel is assigned and the 23% higher source rate in the CE 16×16 lower end-fitting. The three-dimensional analysis treats this situation in two ways. First, no credit is taken for the shielding effectiveness of the spacer assembly in the three-dimensional analysis. Second, a separate analysis is performed in which the design basis PWR fuel and canister are positioned in the cask in the same geometry as the Class 3 canister. Refer to Section 5.4.2.2.

In the BWR case, the GE 7×7 BWR/2–3 and GE 7×7 BWR/4–6 fuel assembly types show the highest radial model dose rates. However, these assemblies are rejected as design basis assemblies because they are no longer in common use, and the U.S. spent fuel inventory does not contain a significant number of these assemblies with burnup, initial enrichment, and decay time combinations leading to source rates as high as those of the GE 9×9–2 [9]. In the axial cases, the GE 9×9-2 assembly shows bounding dose rates in all cases except the GE 7×7 BWR/4–6 and is within 1% of that value.

Ten-year cooled source spectra for the BWR design basis GE 9×9–2 fuel assembly are shown in Table 5.2-15 through Table 5.2-17.

5.2.6 Axial Source Profile

5.2.6.1 Axial Burnup Profile

For PWR fuel with burnup exceeding 30 GWD/MTU, an enveloping axial burnup profile with a 1.08 uniform peaking factor can be justified on the basis of calculated PWR data from Seabrook Station and Maine Yankee and from measured Turkey Point gamma data [12, 13, 14, 15, 16]. This normalized enveloping shape is shown in Figure 5.2-1. A uniform peaking factor of 1.08 is applied from 0.15 to 0.85 fraction of core height, and on either side of this envelope, the burnup/decay heat goes to 0.547 at 0.0 and 1.0 fraction of core height.

For BWR fuel with burnup exceeding 30 GWD/MTU, an enveloping burnup profile with a 1.22 maximum peaking factor can be justified on the basis of calculated BWR data from Washington Public Power BWR/4-6 data [17]. This normalized enveloping shape is shown in Figure 5.2-2. Uniform peaking factors of 1.22 and 1.18 are applied from 0.15 to 0.55, and 0.55 to 0.80 fraction of core height, respectively. On either side of this envelope, the burnup goes to 0.045 at 0.0 and 1.0 fraction of core height.

5.2.6.2 Axial Source Profiles

In the three-dimensional analyses, axial radiation source rate profiles are related to the axial burnup profile described above. Source rates are assumed to vary with burnup according to

$$S = aB^{b}$$

where S is the source rate for a particular radiation type, B is the burnup at a given axial elevation, a is a normalization factor, and b is the exponent given in Table 5.2-18 for each radiation type. The exponent b is determined from fits to SAS2H-computed source rates at various burnups for both PWR and BWR fuels. The numeric value of a is not computed explicitly. Instead, a scale factor r is computed that relates the total source rate (SAS4 input parameter) to the source rate at the average burnup (as computed from SAS2H analyses).

$$r = \frac{\overline{S}}{S(\overline{B})} = \frac{\frac{a}{H}\int B^b dz}{a\overline{B}^b}$$

where H is the height of the fuel region. With the burnup profile normalized to unity, this becomes

$$r = \frac{1}{H}\int B^b dz .$$

The integral is evaluated numerically by using the trapezoid rule, and the resulting scale factors are shown in Table 5.2-20 for PWR and BWR neutron source rates. The scale factor is unity for photon sources because photon source rates vary linearly with burnup.

The fuel neutron and fuel gamma source rate profile for the design basis PWR fuel assembly is tabulated in Table 5.2-20 and shown graphically in Figure 5.2-3. Corresponding BWR profiles are given in Table 5.2-21 and Figure 5.2-4.

Figure 5.2-1 Enveloping Axial Burnup Profile for PWR Design Basis Fuel

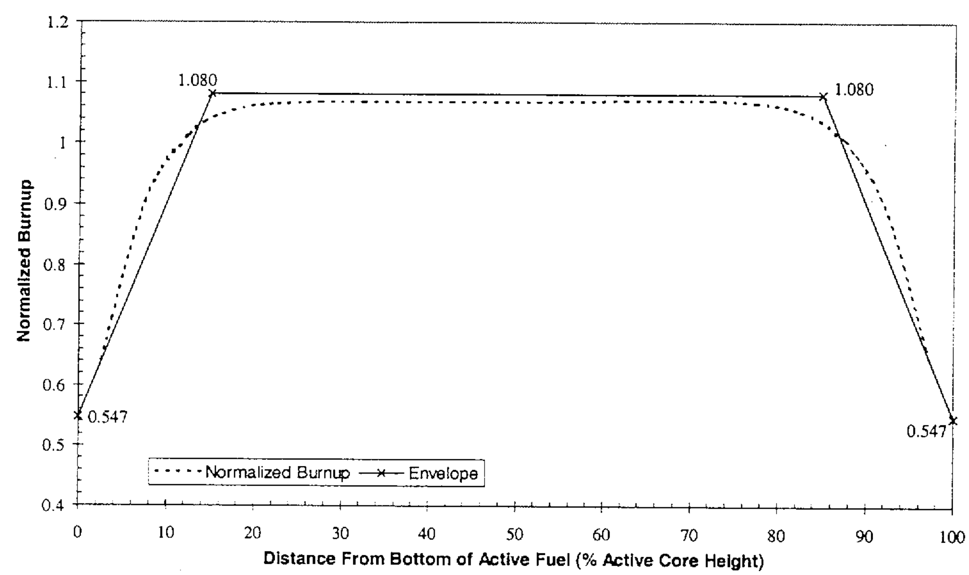


Figure 5.2-2 Enveloping Axial Burnup Profile for BWR Design Basis Fuel

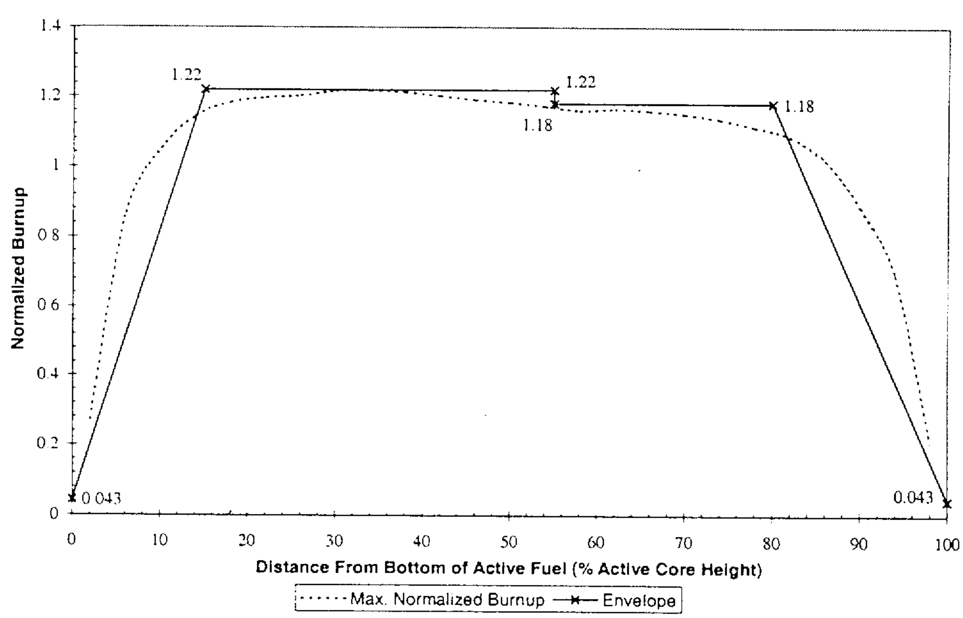


Figure 5.2-3 PWR Photon and Neutron Axial Source Profiles

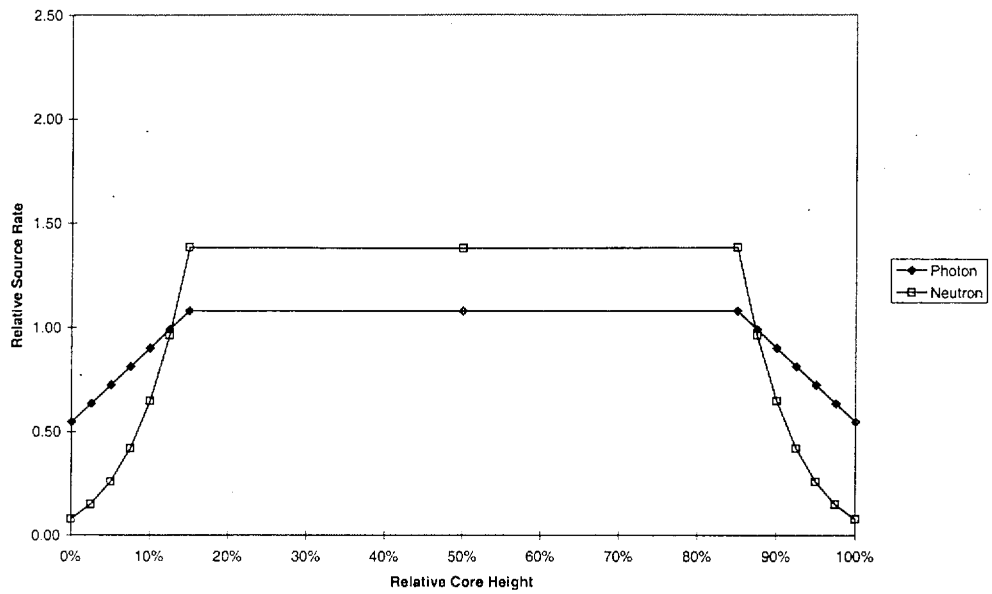


Figure 5.2-4 BWR Photon and Neutron Axial Source Profiles

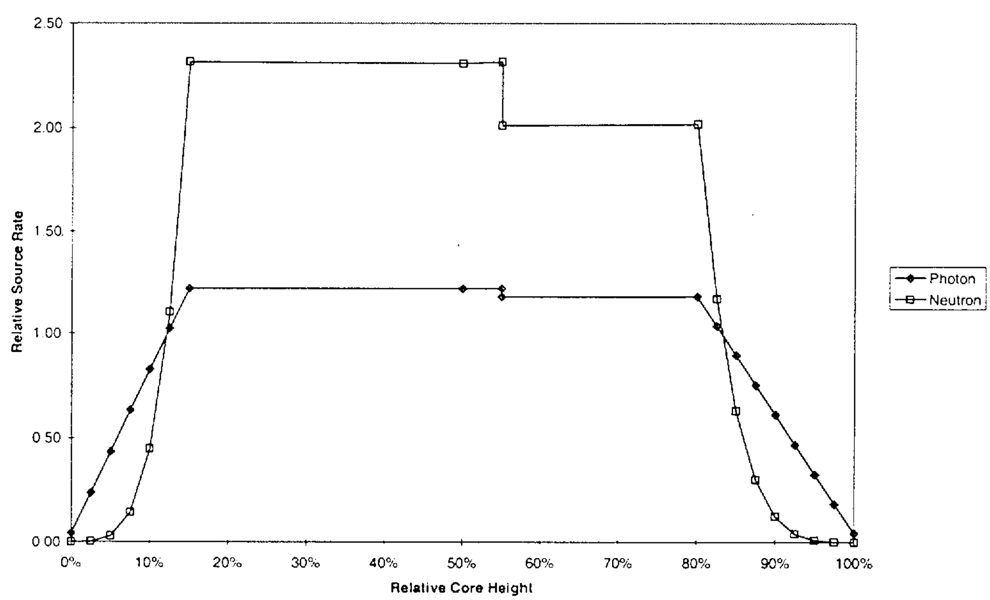


Table 5.2-1 Description of Fuel Assembly Types

Fuel Type	**Class**	**Description**
WE 15×15	Class 1	Westinghouse 15×15
WE 17×17	Class 1	Westinghouse 17×17
BW 15×15	Class 2	Babcock & Wilcox 15×15
CE 16×16	Class 3	Combustion Engineering 16×16
GE 7×7	Class 4	General Electric 7×7 BWR/2-3 Reactor Type
GE 8×8-2	Class 4	General Electric 8×8 BWR/2-3 Reactor Type, 2 water holes
GE 8×8-4	Class 4	General Electric 8×8 BWR/2-3 Reactor Type, 1 water hole
GE 7×7	Class 5	General Electric 7×7 BWR/4-6 Reactor Type
GE 8×8-2	Class 5	General Electric 8×8 BWR/4-6 Reactor Type, 2 water holes
GE 8×8-4	Class 5	General Electric 8×8 BWR/4-6 Reactor Type, 1 water hole
GE 9×9-2	Class 5	General Electric 9×9 BWR/4-6 Reactor Type, 2 water holes, 79 fuel rods

Table 5.2-2 Representative PWR Fuel Assembly Physical Characteristics

Fuel Parameter	WE 15×15 Std (Class 1)	WE 17×17 Std (Class 1)	BW 15×15 Mark B (Class 2)	CE 16×16 System 80 (Class 3)
ASSEMBLY DATA				
Rod array	15×15	17×17	15×15	16×16
Assembly length, in	159.71	159.77	165.63	178.25
Assembly width, in	8.43	8.43	8.54	8.10
Assembly weight, lbs	1,472	1,482	1,515	1,430
Active fuel length, in	144	144	144	150
Max U loading, kg	464.6	467.1	480.7	441.7
FUEL ROD DATA				
No. of fuel rods	204	264	208	236
Rod pitch, in	0.563	0.496	0.568	0.506
Rod diameter, in	0.422	0.374	0.430	0.382
Cladding material	Zirc–4	Zirc–4	Zirc–4	Zirc–4
Cladding thickness, in	0.0242	0.0225	0.0265	0.0250
Pellet diameter, in	0.3659	0.3225	0.3686	0.3250
Init. Enrich, wt %	3.7	3.7	3.7	3.7
GUIDE TUBE DATA				
No. tubes	16	24	16	5
Tube diameter, in.	0.545	0.482	0.530	0.98
Tube thickness, in.	0.015	0.016	0.016	0.035
Tube material	Zirc	Zirc	Zirc	Zirc
INSTRUMENT TUBE DATA				
No. tubes	1	1	1	0
Tube diameter, in.	0.545	0.482	0.493	–
Tube thickness, in.	0.015	0.016	0.026	–
Tube material	Zirc	Zirc	Zirc	–

Table 5.2-3 Representative PWR Fuel Assembly Hardware Data Per Assembly

Assembly Region	WE 15×15 Std (Class 1)	WE 17×17 Std (Class 1)	BW 15×15 Mark B (Class 2)	CE 16×16 System 80 (Class 3)
	Material Mass [kg/assembly]			
Upper End Fitting	Inconel/SS 11.80	Inconel/SS 7.85	Inconel/SS 10.76	Inconel/SS 15.90
Lower End Fitting	Inconel/SS 5.44	Inconel/SS 5.90	Inconel/SS 8.31	Inconel/SS 7.30
Upper Plenum Springs	Inconel 4.07	Inconel 4.43	Inconel 1.98	Inconel 10.70
Upper Plenum Grid	Inconel 1.07	Inconel/SS 0.88	Zirc 1.04	Zirc 0.82
Lower Plenum Springs	–	–	Inconel 1.98	–
Lower Plenum Grid	–	–	Zirc 1.3	–
Incore Grid	Inconel/SS 8.06	Inconel/SS 5.44	Inconel 4.9	Zirc 7.35
Guide Tubes	Zirc 9.39	Zirc 9.53	Zirc 8.64	Zirc 11.3
Upper End Fitting CC Hardware(1)	SS 2.47	SS 2.95	SS 3.41	–
Upper Plenum CC Hardware(1)	SS 4.23	SS 4.04	SS 3.64	–
Fuel Region CC Hardware(1)	SS 11.39	SS 11.00	–	–

(1) Control component (CC) hardware consists of activated stainless steel from burnable poison and thimble plug hardware.

Table 5.2-4 Nuclear and Thermal Parameters of PWR Fuel Assemblies with 3.7 wt % ^{235}U Enrichment, 45,000 MWD/MTU Burnup, 10 Years Cooling Time

Assembly	Decay Heat [kW]	Neutron Source [n/s]	Gamma Source [γ/s]	3-Cycle Hardware Source [γ/kg/s]	1-Cycle Hardware Source [γ/kg/s]
WE 15×15	0.780	2.718E+08	3.707E+15	4.045E+12	1.409E+12
WE 17×17	0.791	2.783E+08	3.755E+15	4.079E+12	1.422E+12
BW 15×15	0.812	2.887E+08	3.832E+15	4.008E+12	1.417E+12
CE 16×16	0.745	2.595E+08	3.538E+15	4.051E+12	1.417E+12

Table 5.2-5 PWR Fuel Assembly Activated Hardware Comparison [γ/s], 10 Year Cooling Time

Fuel Type	Upper End-Fitting	Lower End-Fitting	Upper Plenum	Lower Plenum	Fuel Hardware
WE 15×15	5.121E+12	4.401E+12	5.049E+12	–	4.865E+13
WE 17×17	3.621E+12	4.813E+12	5.231E+12	–	3.782E+13
BW 15×15	4.795E+12	3.331E+12	2.619E+12	1.587E+12	1.964E+13
CE 16×16	3.403E+12	5.914E+12	8.669E+12	–	–

Table 5.2-6 Representative BWR Fuel Physical Characteristics

Assembly	GE 7×7		GE 8×8–2		GE 8×8–4		GE 9×9–2
BWR Reactor	2–3	4–6	2–3	4–6	2–3	4–6	4–6
Canister Class	4	5	4	5	4	5	5
Assembly Version	GE-2b	GE-2	GE-5	GE-5	GE-10	GE-10	GE-11
ASSEMBLY DATA							
Assembly length, in.⁽²⁾	171.3	176.2	171.3	176.2	171.3	176.2	176.2
Assembly width, in.	5.44	5.44	5.44	5.44	5.44	5.44	5.44
Assembly pitch, in.	6.0	6.0	6.0	6.0	6.0	6.0	6.0
Active fuel length, in.	144	144	144⁽¹⁾	150	144⁽¹⁾	150	150
Max U loading, kg	197.7	197.7	177.3⁽¹⁾	184.7	171.7⁽¹⁾	178.7	197.9
Assembly power, MW	3.85	4.95	3.85	4.95	3.85	4.95	4.95
FUEL ROD DATA							
No. fuel rods	49	49	62	62	60	60	79
Rod pitch, in.	0.738	0.738	0.640	0.640	0.640	0.640	0.567
Rod diameter, in.	0.563	0.563	0.483	0.483	0.484	0.484	0.441
Cladding material	Zirc–2	Zirc–2	Zirc–2	Zirc–2	Zirc–4	Zirc–4	Zirc–4
Cladding thick, in.	0.032	0.032	0.032	0.032	0.032	0.032	0.028
Pellet diameter, in.	0.487	0.487	0.410	0.410	0.410	0.410	0.376

Note: (1) Active fuel length normalized to 144 in.
(2) Modeled assembly length standardized to 171.3 in for BWR/2-3 fuel and 176.2 in. for BWR/4-6 fuel

Table 5.2-7 Representative BWR Fuel Assembly Hardware Data

Array	Reactor Type	Upper End Mass	Lower End Mass	Plenum Spring Mass	Incore Grid Mass [kg]		Channel Mass
		[kg]	[kg]	[kg]	Zirc	Inconel	[kg]
GE 7×7	BWR/2–3	2.05	4.36	1.85	1.70	0.32	30
GE 7×7	BWR/4–6	2.05	4.36	1.85	1.70	0.32	30
GE 8×8–2	BWR/2–3	2.1	4.83	2.0	2.20	0.29	30
GE 8×8–2	BWR/4–6	2.1	4.83	2.0	2.20	0.29	35
GE 8×8-4	BWR/2–3	2.56	4.75	1.3	2.20	0.29	30
GE 8×8-4	BWR/4–6	2.56	4.75	1.3	2.20	0.29	35
GE 9×9–2	BWR/4–6	2.08	4.74	1.68	2.50	0.12	35

Table 5.2-8 Nuclear and Thermal Parameters of BWR Fuel with 3.25 wt % ^{235}U Enrichment, 40,000 MWD/MTU Burnup and 10 Years Cooling Time

Assembly	**Reactor**	**Decay Heat [kW]**	**Neutron Source [n/s]**	**Gamma Source [γ/s]**	**Hardware Source [γ/kg/s]**
GE 7×7	BWR/2–3	0.280	8.675E+07	1.335E+15	3.616E+12
GE 7×7	BWR/4–6	0.284	8.753E+07	1.366E+15	3.876E+12
GE 8×8–2	BWR/2–3	0.247	7.136E+07	1.206E+15	3.765E+12
GE 8×8–2	BWR/4–6	0.260	7.485E+07	1.278E+15	3.969E+12
GE 8×8-4	BWR/2–3	0.239	6.737E+07	1.170E+15	3.839E+12
GE 8×8-4	BWR/4–6	0.250	7.035E+07	1.239E+15	4.035E+12
GE 9×9–2	BWR/4–6	0.282	8.533E+07	1.364E+15	3.841E+12

Table 5.2-9 BWR Fuel Assembly Activated Hardware Comparison [γ/s] at 40,000 MWD/MTU Burnup, 10 Year Cooled

Fuel Type	**Reactor**	**Upper End-Fitting**	**Lower End-Fitting**	**Plenum**	**Fuel Hardware**
7×7	BWR/2–3	7.413E+11	2.365E+12	1.338E+12	1.157E+12
7×7	BWR/4–6	7.946E+11	2.535E+12	1.434E+12	1.240E+12
8×8–2	BWR/2–3	7.907E+11	2.728E+12	1.506E+12	1.092E+12
8×8–2	BWR/4–6	8.335E+11	2.876E+12	1.588E+12	1.151E+12
8×8-4	BWR/2–3	9.784E+11	2.745E+12	1.002E+12	1.117E+12
8×8-4	BWR/4–6	1.028E+12	2.884E+12	1.052E+12	1.174E+12
9×9–2	BWR/4–6	7.989E+11	2.731E+12	1.291E+12	4.609E+11

Table 5.2-10 One-Dimensional Dose Rate Results Relative to PWR Design Basis

Class	Fuel	Normal Conditions					
		Radial		Top Axial		Bottom Axial	
		Surface	2.4 m	Surface	2 m	Surface	2 m
Class 1	WE 15×15	1.04	1.04	0.95	0.95	0.97	0.97
	WE 17×17	1.00	1.00	1.00	1.00	1.00	1.00
Class 2	BW 15×15	0.93	0.91	0.94	0.94	0.95	0.97
Class 3	CE 16×16	0.79	0.78	0.59	0.59	1.97	1.97
Class	**Fuel**	**Accident Conditions**					
		Radial		**Top Axial**		**Bottom Axial**	
		Surface	**1 m**	**Surface**	**1 m**	**Surface**	**1 m**
Class 1	WE 15×15	0.99	0.99	0.94	0.94	0.98	0.98
	WE 17×17	1.00	1.00	1.00	1.00	1.00	1.00
Class 2	BW 15×15	1.01	1.01	0.95	0.95	0.83	0.85
Class 3	CE 16×16	0.92	0.93	0.59	0.59	1.83	1.85

Table 5.2-11 One-Dimensional Dose Rate Results Relative to BWR Design Basis

Class	Fuel	Normal Conditions					
		Radial		Top Axial		Bottom Axial	
		Surface	2.4 m	Surface	2 m	Surface	2 m
Class 4	GE 7×7	1.03	1.01	0.98	0.98	0.11	0.08
	GE 8×8-2	0.91	0.89	0.81	0.81	0.09	0.07
	GE 8×8-4	0.88	0.87	0.78	0.78	0.09	0.07
Class 5	GE 7×7	1.05	1.03	0.89	0.89	1.01	1.01
	GE 8×8-2	0.92	0.93	0.88	0.88	1.01	1.02
	GE 8×8-4	0.89	0.90	0.84	0.84	1.01	1.02
	GE 9×9-2	1.00	1.00	1.00	1.00	1.00	1.00
Class	**Fuel**	**Accident Conditions**					
		Radial		**Top Axial**		**Bottom Axial**	
		Surface	**1 m**	**Surface**	**1 m**	**Surface**	**1 m**
Class 4	GE 7×7	1.04	1.03	0.98	0.98	0.25	0.22
	GE 8×8-2	0.89	0.88	0.82	0.82	0.21	0.18
	GE 8×8-4	0.85	0.85	0.78	0.78	0.20	0.18
Class 5	GE 7×7	1.05	1.04	0.89	0.89	1.01	1.01
	GE 8×8-2	0.90	0.90	0.88	0.88	0.95	0.96
	GE 8×8-4	0.86	0.86	0.84	0.84	0.92	0.94
	GE 9×9-2	1.00	1.00	1.00	1.00	1.00	1.00

Table 5.2-12 Design Basis PWR 10 Year Fuel Neutron Source Spectrum

Group	Elow [MeV]	Ehigh [MeV]	Spectrum [n/sec/assy]
1	6.43E+00	2.00E+01	5.138E+06
2	3.00E+00	6.43E+00	5.841E+07
3	1.85E+00	3.00E+00	6.471E+07
4	1.40E+00	1.85E+00	3.645E+07
5	9.00E-01	1.40E+00	4.932E+07
6	4.00E-01	9.00E-01	5.378E+07
7	1.00E-01	4.00E-01	1.053E+07
8	1.70E-02	1.00E-01	0.0
9	3.00E-03	1.70E-02	0.0
10	5.50E-04	3.00E-03	0.0
11	1.00E-04	5.50E-04	0.0
12	3.00E-05	1.00E-04	0.0
13	1.00E-05	3.00E-05	0.0
14	3.05E-06	1.00E-05	0.0
15	1.77E-06	3.05E-06	0.0
16	1.30E-06	1.77E-06	0.0
17	1.13E-06	1.30E-06	0.0
18	1.00E-06	1.13E-06	0.0
19	8.00E-07	1.00E-06	0.0
20	4.00E-07	8.00E-07	0.0
21	3.25E-07	4.00E-07	0.0
22	2.25E-07	3.25E-07	0.0
23	1.00E-07	2.25E-07	0.0
24	5.00E-08	1.00E-07	0.0
25	3.00E-08	5.00E-08	0.0
26	1.00E-08	3.00E-08	0.0
27	1.00E-11	1.00E-08	0.0
Total			2.783E+08

Table 5.2-13 Design Basis PWR 10 Year Fuel Photon Spectrum

Group	Elow [MeV]	Ehigh [MeV]	Spectrum [γ/sec/assy]	Spectrum [MeV/sec/assy]
1	8.00E+00	1.00E+01	1.5743E+05	1.417E+06
2	6.50E+00	8.00E+00	7.4152E+05	5.376E+06
3	5.00E+00	6.50E+00	3.7803E+06	2.174E+07
4	4.00E+00	5.00E+00	9.4200E+06	4.239E+07
5	3.00E+00	4.00E+00	4.2178E+08	1.476E+09
6	2.50E+00	3.00E+00	3.7019E+09	1.018E+10
7	2.00E+00	2.50E+00	5.3258E+10	1.198E+11
8	1.66E+00	2.00E+00	1.3025E+11	2.384E+11
9	1.33E+00	1.66E+00	8.1351E+12	1.216E+13
10	1.00E+00	1.33E+00	7.6492E+13	8.911E+13
11	8.00E-01	1.00E+00	1.0567E+14	9.510E+13
12	6.00E-01	8.00E-01	1.8094E+15	1.267E+15
13	4.00E-01	6.00E-01	1.7568E+14	8.784E+13
14	3.00E-01	4.00E-01	3.7162E+13	1.301E+13
15	2.00E-01	3.00E-01	6.0583E+13	1.515E+13
16	1.00E-01	2.00E-01	2.1673E+14	3.251E+13
17	5.00E-02	1.00E-01	2.7731E+14	2.080E+13
18	1.00E-02	5.00E-02	9.8776E+14	2.963E+13
Total			3.7551E+15	1.6623E+15

Table 5.2-14 Design Basis PWR 10 Year Hardware Photon Spectrum

Group	**Elow [MeV]**	**Ehigh [MeV]**	**Spectrum [γ/sec/kg]**	**Spectrum [MeV/sec/kg]**
1	8.00E+00	1.00E+01	0.0000E+00	0.000E+00
2	6.50E+00	8.00E+00	0.0000E+00	0.000E+00
3	5.00E+00	6.50E+00	0.0000E+00	0.000E+00
4	4.00E+00	5.00E+00	0.0000E+00	0.000E+00
5	3.00E+00	4.00E+00	2.2491E-15	7.872E-15
6	2.50E+00	3.00E+00	3.2497E+04	8.937E+04
7	2.00E+00	2.50E+00	2.0958E+07	4.716E+07
8	1.66E+00	2.00E+00	7.5979E-06	1.390E-05
9	1.33E+00	1.66E+00	8.8313E+11	1.320E+12
10	1.00E+00	1.33E+00	3.1272E+12	3.643E+12
11	8.00E-01	1.00E+00	1.0399E+09	9.359E+08
12	6.00E-01	8.00E-01	3.6948E+06	2.586E+06
13	4.00E-01	6.00E-01	1.0639E+07	5.320E+06
14	3.00E-01	4.00E-01	1.6834E+08	5.892E+07
15	2.00E-01	3.00E-01	1.2830E+08	3.208E+07
16	1.00E-01	2.00E-01	2.5839E+09	3.876E+08
17	5.00E-02	1.00E-01	1.0711E+10	8.033E+08
18	1.00E-02	5.00E-02	5.4096E+10	1.623E+09
Total			4.0791E+12	4.9674E+12

Table 5.2-15 Design Basis BWR 10 Year Fuel Neutron Source Spectrum

Group	Elow [MeV]	Ehigh [MeV]	Spectrum [n/sec/assy]
1	6.43E+00	2.00E+01	1.574E+06
2	3.00E+00	6.43E+00	1.791E+07
3	1.85E+00	3.00E+00	1.987E+07
4	1.40E+00	1.85E+00	1.118E+07
5	9.00E-01	1.40E+00	1.511E+07
6	4.00E-01	9.00E-01	1.647E+07
7	1.00E-01	4.00E-01	3.224E+06
8	1.70E-02	1.00E-01	0.0
9	3.00E-03	1.70E-02	0.0
10	5.50E-04	3.00E-03	0.0
11	1.00E-04	5.50E-04	0.0
12	3.00E-05	1.00E-04	0.0
13	1.00E-05	3.00E-05	0.0
14	3.05E-06	1.00E-05	0.0
15	1.77E-06	3.05E-06	0.0
16	1.30E-06	1.77E-06	0.0
17	1.13E-06	1.30E-06	0.0
18	1.00E-06	1.13E-06	0.0
19	8.00E-07	1.00E-06	0.0
20	4.00E-07	8.00E-07	0.0
21	3.25E-07	4.00E-07	0.0
22	2.25E-07	3.25E-07	0.0
23	1.00E-07	2.25E-07	0.0
24	5.00E-08	1.00E-07	0.0
25	3.00E-08	5.00E-08	0.0
26	1.00E-08	3.00E-08	0.0
27	1.00E-11	1.00E-08	0.0
Total			8.533E+07

Table 5.2-16 Design Basis BWR 10 Year Fuel Photon Spectrum

Group	Elow [MeV]	Ehigh [MeV]	Spectrum [γ/sec/assy]	Spectrum [MeV/sec/assy]
1	8.00E+00	1.00E+01	4.8234E+04	4.341E+05
2	6.50E+00	8.00E+00	2.2719E+05	1.647E+06
3	5.00E+00	6.50E+00	1.1582E+06	6.660E+06
4	4.00E+00	5.00E+00	2.8862E+06	1.299E+07
5	3.00E+00	4.00E+00	1.2490E+08	4.372E+08
6	2.50E+00	3.00E+00	1.0814E+09	2.974E+09
7	2.00E+00	2.50E+00	1.5523E+10	3.493E+10
8	1.66E+00	2.00E+00	4.5751E+10	8.372E+10
9	1.33E+00	1.66E+00	2.7186E+12	4.064E+12
10	1.00E+00	1.33E+00	2.7140E+13	3.162E+13
11	8.00E-01	1.00E+00	3.4940E+13	3.145E+13
12	6.00E-01	8.00E-01	6.6179E+14	4.633E+14
13	4.00E-01	6.00E-01	5.6730E+13	2.837E+13
14	3.00E-01	4.00E-01	1.3773E+13	4.821E+12
15	2.00E-01	3.00E-01	2.2340E+13	5.585E+12
16	1.00E-01	2.00E-01	7.9396E+13	1.191E+13
17	5.00E-02	1.00E-01	1.0163E+14	7.622E+12
18	1.00E-02	5.00E-02	3.6365E+14	1.091E+13
Total			1.3642E+15	5.9972E+14

Table 5.2-17 Design Basis BWR 10 Year Hardware Photon Spectrum

Group	Elow [MeV]	Ehigh [MeV]	Spectrum [γ/sec/kg]	Spectrum [γ/sec/kg]
1	8.00E+00	1.00E+01	0.0000E+00	0.000E+00
2	6.50E+00	8.00E+00	0.0000E+00	0.000E+00
3	5.00E+00	6.50E+00	0.0000E+00	0.000E+00
4	4.00E+00	5.00E+00	0.0000E+00	0.000E+00
5	3.00E+00	4.00E+00	1.1331E-15	3.966E-15
6	2.50E+00	3.00E+00	3.0602E+04	8.416E+04
7	2.00E+00	2.50E+00	1.9736E+07	4.441E+07
8	1.66E+00	2.00E+00	5.6617E-06	1.036E-05
9	1.33E+00	1.66E+00	8.3162E+11	1.243E+12
10	1.00E+00	1.33E+00	2.9448E+12	3.431E+12
11	8.00E-01	1.00E+00	7.2217E+08	6.500E+08
12	6.00E-01	8.00E-01	3.4793E+06	2.436E+06
13	4.00E-01	6.00E-01	1.0019E+07	5.010E+06
14	3.00E-01	4.00E-01	1.5852E+08	5.548E+07
15	2.00E-01	3.00E-01	1.2082E+08	3.021E+07
16	1.00E-01	2.00E-01	2.4332E+09	3.650E+08
17	5.00E-02	1.00E-01	1.0086E+10	7.565E+08
18	1.00E-02	5.00E-02	5.0986E+10	1.530E+09
Total			3.8410E+12	4.6774E+12

Table 5.2-18 Source Rate Versus Burnup Fit Parameters

Radiation Type	Exponent, b
Neutron	4.22
Photon	1.00

Table 5.2-19 Scale Factors Applied to Neutron Source Rate at Average Burnup

Fuel Type	Scale Factor
PWR	1.1250
BWR	1.5821

Table 5.2-20 PWR Axial Source Profile

% Core Height	Burnup Profile	Photon Source	Neutron Source
0.00%	0.5470	0.5470	7.840E-02
2.50%	0.6358	0.6358	1.479E-01
5.00%	0.7247	0.7247	2.569E-01
7.50%	0.8135	0.8135	4.185E-01
10.00%	0.9023	0.9023	6.481E-01
12.50%	0.9912	0.9912	9.633E-01
15.00%	1.0800	1.0800	1.384E+00
50.00%	1.0790	1.0790	1.378E+00
85.00%	1.0800	1.0800	1.384E+00
87.50%	0.9912	0.9912	9.633E-01
90.00%	0.9023	0.9023	6.481E-01
92.50%	0.8135	0.8135	4.185E-01
95.00%	0.7247	0.7247	2.569E-01
97.50%	0.6358	0.6358	1.479E-01
100.00%	0.5470	0.5470	7.840E-02

Table 5.2-21 BWR Axial Source Rate Profile

% Core Height	Burnup Profile	Photon Source	Neutron Source
0.00%	0.0430	0.0430	1.711E-06
2.50%	0.2392	0.2392	2.388E-03
5.00%	0.4353	0.4353	2.991E-02
7.50%	0.6315	0.6315	1.437E-01
10.00%	0.8277	0.8277	4.501E-01
12.50%	1.0238	1.0238	1.105E+00
15.00%	1.2200	1.2200	2.314E+00
50.00%	1.2190	1.2190	2.306E+00
55.00%	1.2200	1.2200	2.314E+00
55.01%	1.1800	1.1800	2.011E+00
80.00%	1.1810	1.1810	2.018E+00
82.50%	1.0379	1.0379	1.170E+00
85.00%	0.8958	0.8958	6.284E-01
87.50%	0.7536	0.7536	3.031E-01
90.00%	0.6115	0.6115	1.255E-01
92.50%	0.4694	0.4694	4.110E-02
95.00%	0.3272	0.3272	8.970E-03
97.50%	0.1851	0.1851	8.104E-04
100.00%	0.0430	0.0430	1.711E-06

5.3 Model Specification

The licensing basis dose rates for the fully loaded Universal Transport Cask are determined from the results of a detailed three-dimensional analysis performed by using the SCALE SAS4 [2] control sequence. To determine bounding dose rates, the three-dimensional analyses are performed for limiting, design basis PWR and BWR fuel assemblies. These design basis assemblies are selected on the basis of the results of one-dimensional shielding models obtained for the candidate assembly designs described in Section 5.1.1. This section describes both the one- and three-dimensional shielding models.

5.3.1 One-Dimensional Models

One-dimensional analyses are performed to identify the limiting PWR and BWR fuel descriptions. These design basis fuel descriptions are then used in a detailed three-dimensional shielding analysis. Dose rate results for the one-dimensional models are not reported directly since the results are subsumed by the three-dimensional analysis.

5.3.1.1 One-Dimensional Radial Model

The one-dimensional radial model consists of a series of infinitely long concentric cylinders representing the cross-of the cask at the fuel midplane. A representation of the model is shown in Figure 5.3-3. The fuel assemblies and basket are modeled as two concentric regions inside the cask cavity. The innermost region consists of the fuel and that part of the basket material which lies inside the radius of the assumed fuel homogenization region. The radius of the homogenized fuel region is determined by assuming that the area of the fuel region in the model is equivalent to the cross-sectional area of the region enclosed by the periphery of the fuel assemblies in the actual basket configuration (24 in the PWR basket and 56 in the BWR basket). These equivalent circularized cylindrical sources for the PWR and the BWR baskets are shown in Figure 5.3-4 and Figure 5.3-5, respectively.

The annulus between the fuel region and the inner shell is modeled as a homogenized mixture of basket support plates and heat transfer disks (PWR) conserving the total mass of material present. The remaining cask body regions are modeled using the specified dimensions of the cask except for the radial neutron shield. Since the neutron shield thickness varies as a result of its polygonal shape, a volume-conserving equivalent thickness is used.

Note that the three-dimensional analysis described below uses the same homogenized source region description as that developed here. However, the fraction of the basket support plates which lie outside the homogenized fuel region are treated explicitly in the three-dimensional model.

The one-dimensional radial model is developed by using the following radial zones:

- Homogenized Fuel Zone
- Homogenized Basket Outside of Equivalent Fuel Radius
- Canister Wall
- Gap to Cask Inner Wall
- Cask Inner Shell
- Cask Gamma Shield
- Cask Stainless Steel Outer Shell
- Cask NS-4-FR
- Neutron Shell Expansion Space
- Cask Neutron Shield Shell

A graphical illustration of the radial shielding zones is provided in Figure 5.3-3. The dimension for each of the zones is provided in Table 5.3-1. It should be noted that the composition of the canister interior zones varies in source term and material composition depending on which assembly region (fuel, plenum, end-fitting) or fuel assembly type is represented.

5.3.1.2 One-Dimensional Top Axial Model

The one-dimensional model of the top end of the cask consists of a series of infinite slabs representing the thicknesses of the cask materials along the centerline of the cask from the bottom of the active fuel region to the top end of the cask. A typical model of the top axial region of the cask is shown in Figure 5.3-8 for the BW 15×15 fuel assembly. The shielding, including the canister lid and impact limiter materials, consists of a total of 17.0 in. of steel, 29.75 in. of redwood, and 1.5 in. of balsa wood. Refer to Table 5.3-2 for details of the top axial shield dimensions.

5.3.1.3 One-Dimensional Bottom Axial Model

The one-dimensional model of the bottom end of the cask also consists of a series of infinite slabs, representing the cask materials along the centerline of the cask from the top of the active fuel region to the bottom end of the cask. The shielding model used for the bottom region of the cask is shown in Figure 5.3-9. The shielding includes a total of 11.5 in. of steel, 1.0 in. of NS-4-FR neutron shielding, 29.75 in. of redwood, and 1.5 in. of balsa wood, as well as the self-shielding provided by the fuel and the end-fitting regions. Refer to Table 5.3-2 for details of the bottom axial shield dimensions.

5.3.2 Three-Dimensional Shielding Models

The three-dimensional analysis of Universal Transport Cask dose rates permits a high fidelity treatment of complicated cask geometric features and avoids approximations implicit in one-dimensional analyses, such as specification of buckling parameters. The three-dimensional models described here are developed for the SAS4 code, [2] (refer to Section 5.4.1).

SAS4 requires that models be developed separately for the top and bottom of the Universal Transport Cask. Therefore, the models developed here are reflected across the axial midplane of the active fuel region.

To reduce variance in computed values, the SAS4 code permits particles to be biased either axially towards the end of the cask model or radially through the cask walls. Separate runs are required for each biasing direction, although the same geometric models are employed in each case. Biasing parameters are determined automatically by the code on the basis of a simplified one-dimensional model of the problem specified by the user.

Dose rate results are obtained in terms of responses on certain detector surfaces whose location and extents are specified by the user. SAS4 detectors are arranged either as nested cylindrical surfaces in the case of a radially biased analysis or as stacked disks in the case of an axial analysis. An enhancement to the SAS4 code is the ability to subdivide these detector surfaces into subdetectors for which results are tallied individually. In the case of radial detectors, the subdetector description partitions each detector axially and azimuthally as indicated in Figure 5.3-1. For axially biased runs, the detector surfaces are subdivided radially and azimuthally as shown in Figure 5.3-2. The degree of subdivision is specified by the user.

5.3.2.1 Geometric Models

Key geometric features of the three-dimensional model include:

- Explicit treatment of basket support disks in the annular region outside the homogenized source.
- Detailed treatment of cask transition regions, such as trunnion recesses, vent ports, and impact limiters.
- Accurate representation of cask shield material axial and radial extents.

5.3.2.1.1 Source Region

The same homogenized source description used in the one-dimensional model is used in the three-dimensional model; however, the source region volume is represented by a collection of box-shaped structures and has a rectilinear cross section matching the outline of the fuel assembly periphery in the basket. Basket support disk material lying inside the homogenized source region is included with the fuel assembly material. Table 5.3-3 and Table 5.3-4 give additional geometric and source rate information for the design basis PWR and BWR assemblies. Figure 5.3-6 and Figure 5.3-7 show the source region elevations for the design basis fuel assemblies.

The analysis accounts for the neutron, fuel gamma, and fuel hardware gamma sources emanating from the fuel region. The effects of secondary gamma production due to neutron capture and subcritical multiplication of neutrons in fissionable material are considered. Gamma sources in the plenum and end-fitting region are also analyzed.

5.3.2.1.2 Basket

The fraction of basket support plates which lie outside the fuel region homogenization radius are modeled explicitly. Note that due to the requirement of a reflected model, basket support plates which lie precisely at the fuel midplane are conservatively omitted from the model.

In the PWR model, basket support disks are modeled as stainless steel Type 17-4PH disks of thickness 0.5 in. and diameter 65.50 in. with an internal spacing of 4.42 in. between plates. Similarly, the PWR basket heat transfer disks are modeled as aluminum disks of thickness 0.5 in. and diameter 65.25 in. with the same plate spacing. In the BWR model, support disks are modeled as 0.625 in. carbon steel disks with diameter 65.50 in. and an internal spacing of 3.205 in. between plates. The heat transfer disks present in the BWR basket are modeled as 0.50 in. aluminum plates with diameter 65.18 in. and centered between the support disks. For the top model, the disks are positioned by preserving their location and spacing with respect to the top weldment and including only those disks, which lie entirely above the axial position of the fuel midplane. A similar procedure is used to place the disks in the bottom model with respect to the bottom weldment. This approach preserves geometric features near cask transition regions such as trunnion recesses.

5.3.2.1.3 Canister

The vent ports in the canister shield lid are neglected in this analysis. This model simplification has little impact on computed dose rates because in all modeled scenarios, the Transport Cask lid is in place.

5.3.2.1.4 Transport Cask

Figure 5.3-10 through Figure 5.3-13 show typical slices through the PWR and BWR top and bottom three-dimensional models employed in the shielding analysis. Features such as trunnion recesses, ventilation port openings, and basket support plates are described below.

The sketches are understood to be symmetric with respect to the fuel axial midplane and the cask axial centerline. Sketches are provided only for normal conditions of transport. A description of the accident condition geometry is provided in Section 5.3.2.4.

The polygonal shape of the radial neutron shield is modeled as a cylindrical ring with the same cross sectional area. Heat transfer fins embedded in the neutron shield are explicitly included in the three-dimensional shielding model.

The ventilation port in the Universal Transport Cask lid is modeled as a cylindrical void region in the lid. The lid bolting ring is modeled as an annular void surrounding the lid (refer to Figure 5.3-14). The Universal Transport Cask lower rotation pocket model accounts for the penetration of the neutron shield shell and the void region within the pocket, as shown in Figure 5.3-15.

To permit sufficient clearance for the cask lifting yoke, the neutron shielding immediately below each of the four upper trunnions is shouldered. These features are accurately depicted in the cask model. Note that no shielding credit is taken for the trunnions themselves.

For both the upper and lower impact limiters, a simplified model is employed. The limiters are assumed to contact the cask radially, and no detailed model of the fastening system is developed.

5.3.2.2 Biasing Models

For radially biased cases, the one-dimensional biasing models are determined on the basis of the dimensions and materials along a radius taken through the cask at the axial midplane. For the top and bottom axially biased analyses, biasing models are determined on the basis of the materials and dimensions of cask components traversed by the central axis of the cask from the fuel midplane outward in each direction.

5.3.2.3 Detector Descriptions

Modifications to SAS4 permit surface detectors to be subdivided as illustrated in Figure 5.3-1 and Figure 5.3-2. Dose rates are tallied on these individual subdetectors, permitting detailed axial, radial, and azimuthal dose rate profiles to be computed. Detailed descriptions of the surface detector partitioning employed in the PWR cask analysis are given in Table 5.3-5 through Table 5.3-12 for both normal conditions of transport and hypothetical accident conditions. BWR detector locations are specified in a similar manner, with only slight elevational differences in the detector descriptions as a result of the SAS4 requirement that the model be reflected about the fuel axial midplane.

The axial and radial lead slump analyses use the same detector description as that used in the corresponding PWR or BWR normal conditions analysis.

Maximum dose rates at specified radial and axial positions are reported on the basis of the maximum subdetector dose rate computed on the surface detector corresponding to that location. The surface of the cask is defined by the outer surfaces of the impact limiters and the neutron shield shell for normal conditions of transport. In the hypothetical accident conditions, the 1 m detector surfaces are conservatively positioned with respect to the accident geometry of the cask.

5.3.2.4 Accident Conditions

The hypothetical accident condition is modeled by omitting the neutron shield, shield shell, and impact limiter components from the base case normal conditions model. In addition, simultaneous axial and radial lead slumps are assumed to occur. Note that the 1 m radial and axial surface detectors are located with respect to the accident geometry of the cask (refer to Figure 5.3-17).

For the lead slump conditions, voids are introduced in the lead gamma shielding on the basis of the redistribution of lead within the cavity, assuming a post-fabrication gap of 0.045 in. between the outer radial surface of the lead and the inner surface of the cask outer shell. As a result of the simultaneous application of radial and axial lead slumps in the accident condition model, the total modeled lead volume is conservatively less than that actually present in the cask. The radial slump configuration is illustrated in Figure 5.3-16.

In the axial case, the extent of the lead is reduced by 3.05 in. at both ends of the cask simultaneously (due to the reflected model requirement).

5.3.2.5 Neutron Shield Heat Transfer Fin Model

The Transport Cask incorporates explosively-bonded copper/stainless steel heat transfer fins in the radial neutron shield which displace neutron shielding material. The potential exists for this displaced material to provide a streaming path for neutrons to escape the shield. An explicit model of the heat transfer fins is incorporated in the shielding analysis.

The Transport Cask incorporates 24 fins evenly distributed around the cask on 15° angular increments. Each fin is radially oriented and assumed to lie entirely within the neutron shield

shell and outside the cask outer shell. The copper and stainless steel components of the fin are 0.6 cm and 0.8 cm thick, respectively. The fins are treated as a homogenized mixture of copper and stainless steel (refer to Figure 5.3-18).

Detailed azimuthal response profiles are obtained for fuel neutron and fuel gamma sources to assess the relative impact on dose rate responses for each source type. A radial surface detector is positioned on the cask neutron shield shell at the fuel axial midplane and extending 50 cm axially. This detector is subdivided into 72 subdetectors arranged azimuthally (5° increments) and oriented in such a way that every third subdetector straddles a heat transfer fin.

5.3.3 Material Compositions

Isotopic number densities for the various source region and shielding materials are provided in this section. The tables give number densities in units of atoms per barn-centimeter [atom/b-cm] for the various isotopes listed. Note that the tabulated isotope names correspond to the names listed in the Table of Contents of the SCALE 27N18G shielding library employed in this analysis.

The densities of the materials in regions where radioactive sources exist are calculated based on the rectilinear cross sectional shape of the periphery of the fuel/basket region. For one-dimensional analyses, the determination of the effective fuel radius is illustrated in Figure 5.3-4 and Figure 5.3-5. The radioactive source regions include the fuel, plenum, and top and bottom end-fittings. All fuel, plenum, and end-fitting region densities are homogenized. Fuel assembly hardware within the active fuel region is homogenized in the effective fuel region in all of the shielding models. Fuel region number densities are provided in Table 5.3-13. The plenum and end-fitting (upper and lower) densities are presented in Table 5.3-14 and Table 5.3-15.

In the three-dimensional models, the portion of the basket support plates located in the region between the homogenized source region and the canister inner wall are modeled explicitly. In the one-dimensional models, the disk material in this region is homogenized, and the resulting isotopic number densities are given in Table 5.3-16 and Table 5.3-17 for PWR and BWR canisters, respectively.

The remaining cask shielding materials and their densities are included in Table 5.3-18 and Table 5.3-19. The densities reported for the NS-4-FR neutron shielding material reflect the composition of the material after curing as obtained from information supplied by the manufacturer. The

aluminum density is reported for both a solid heterogeneous aluminum disk and for the homogenized basket. The redwood and balsa wood densities are taken from the Standard Composition Library [10] of the SCALE4.3 code package.

Figure 5.3-1 Illustration of SAS4 Axial Surface Detector Partitioning

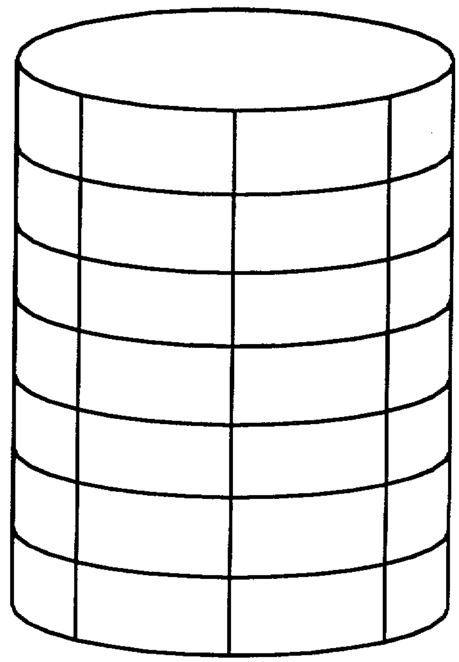


Figure 5.3-2 Illustration of SAS4 Radial Surface Detector Partitioning

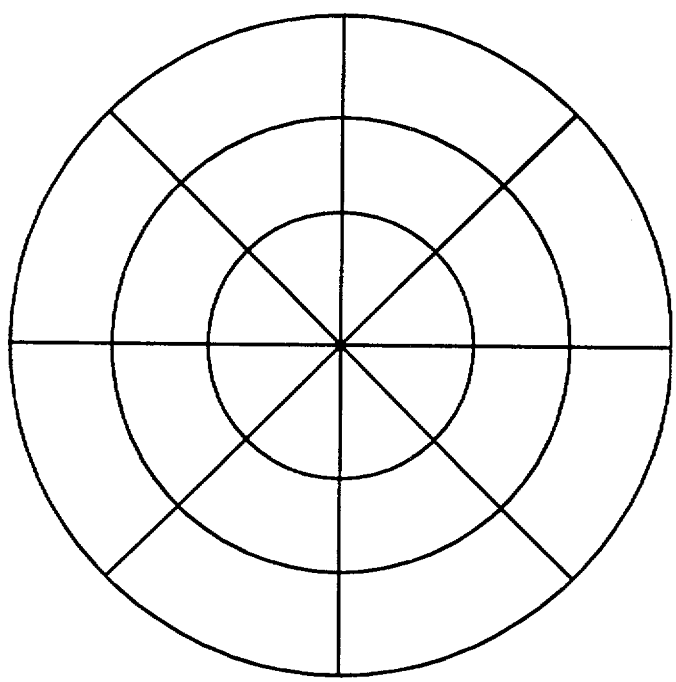


Figure 5.3-3 One-Dimensional Radial Shielding Regions

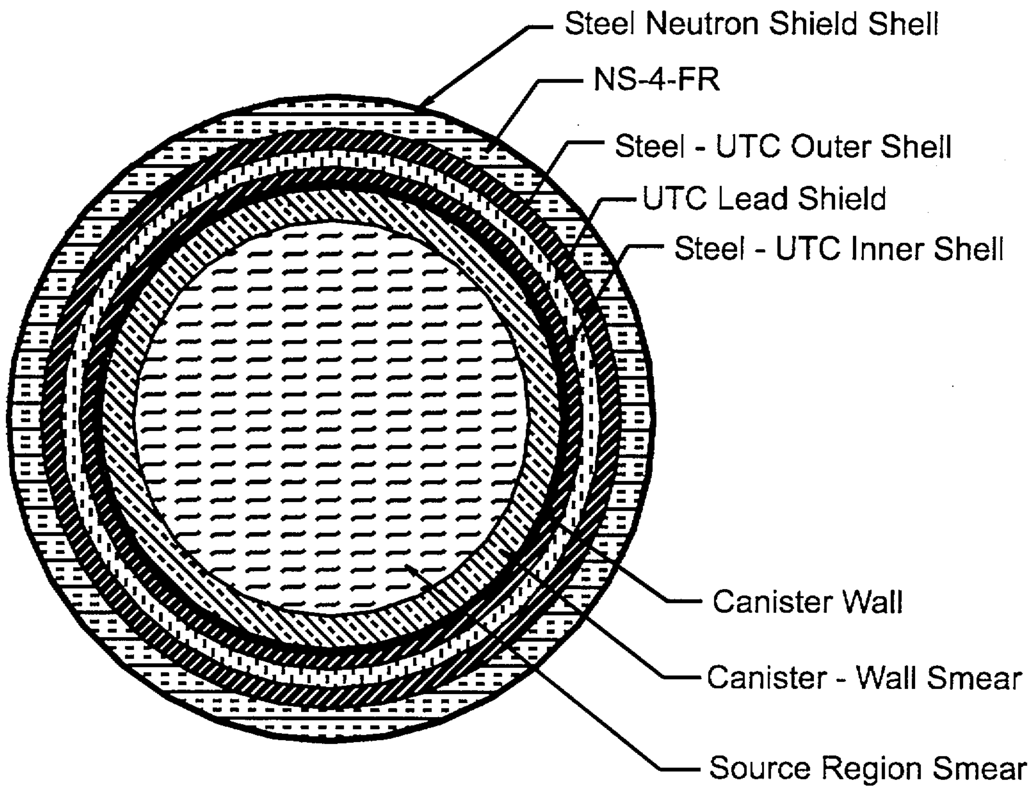


Figure 5.3-4 Equivalent Homogenized Cylindrical Source: PWR (Dimensions in cm)

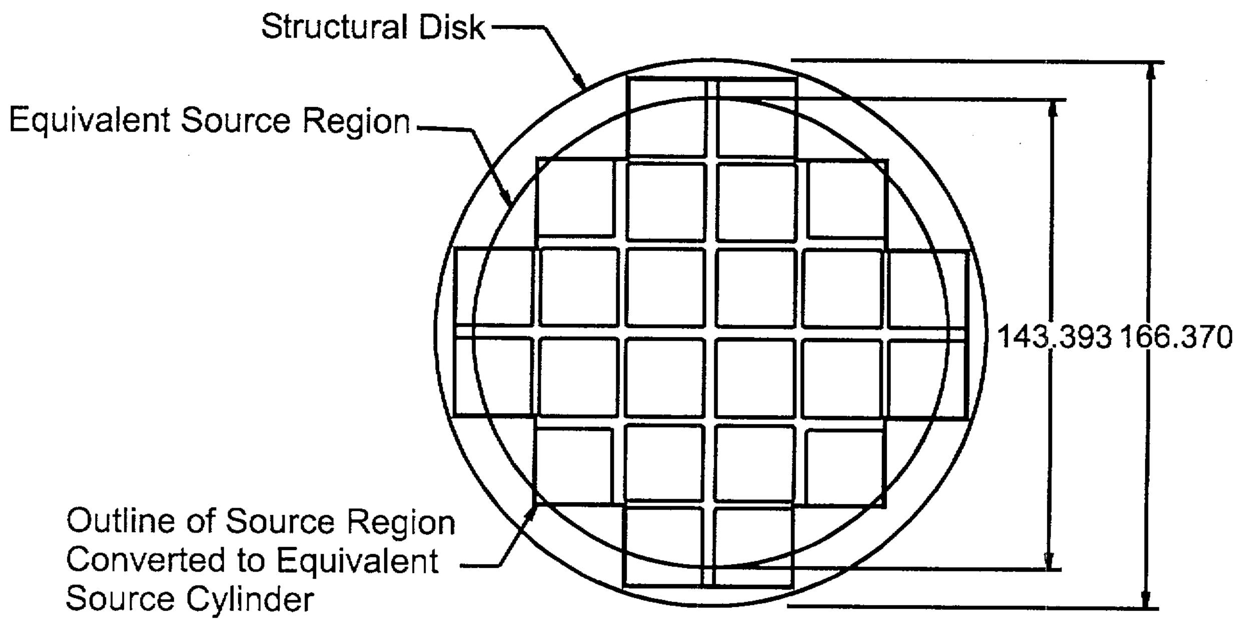


Figure 5.3-5 Equivalent Homogenized Cylindrical Source: BWR (Dimensions in cm)

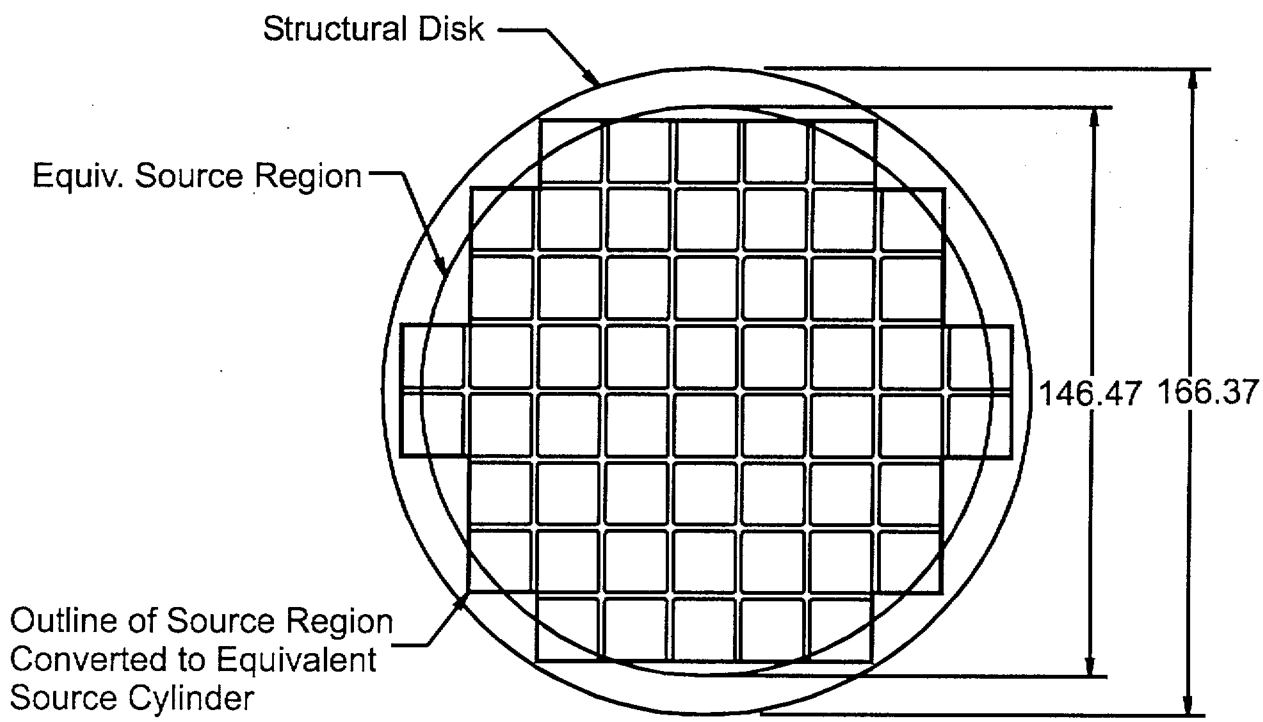


Figure 5.3-6 Design Basis PWR (WE 17×17) Fuel Assembly Source Region Elevations

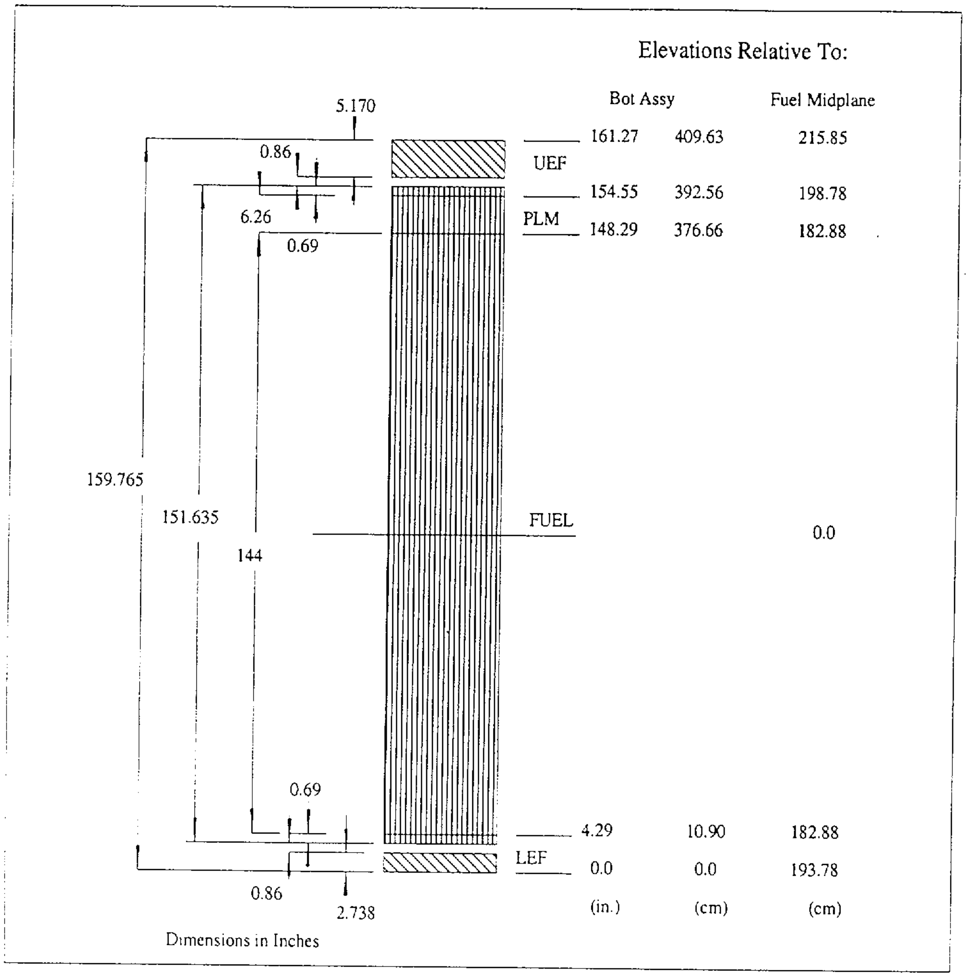


Figure 5.3-7 Design Basis BWR (GE 9×9-2) Fuel Assembly Source Region Elevations

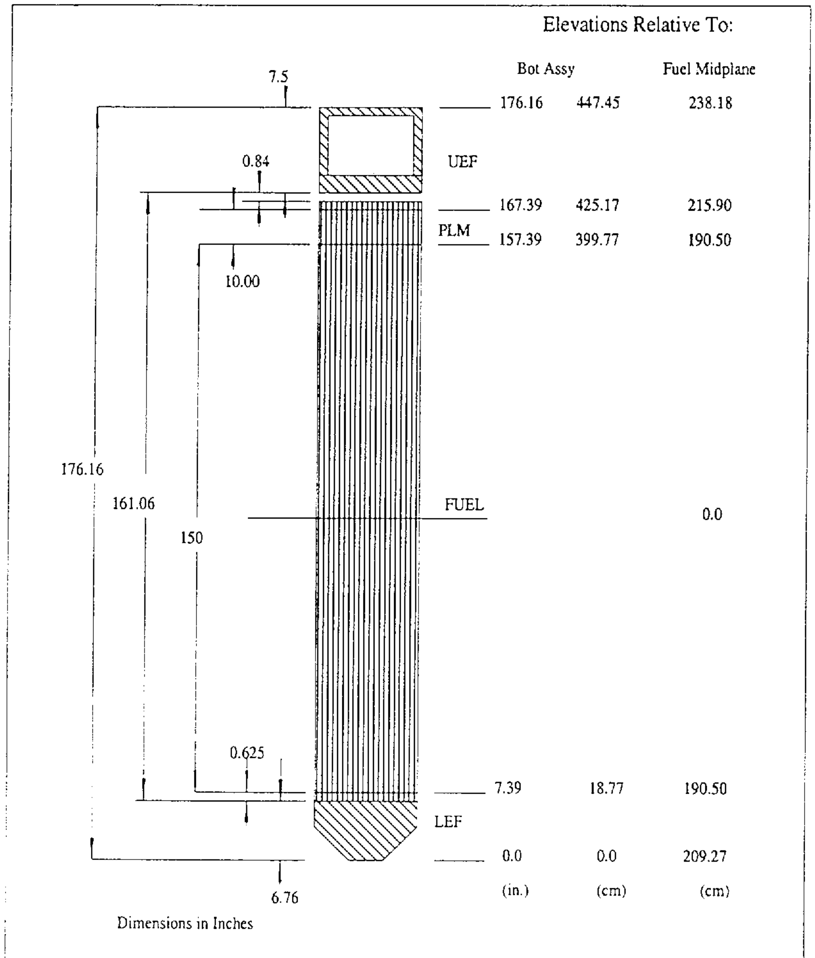


Figure 5.3-8 One-Dimensional Top Axial Model: BW 15×15 Fuel Assembly
(Dimensions in cm)

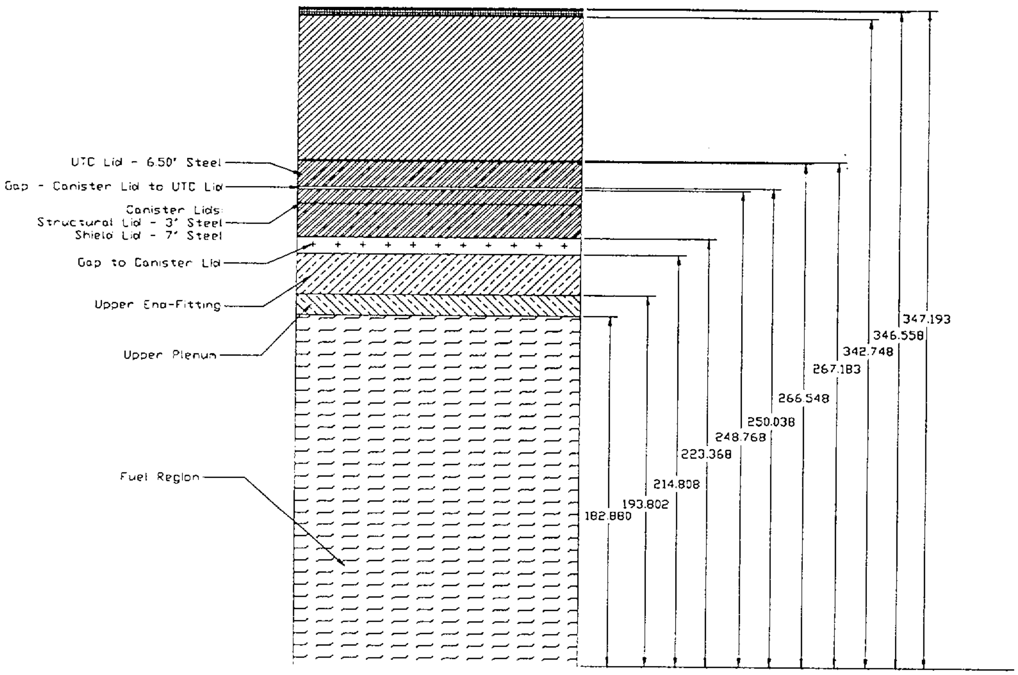


Figure 5.3-9 One-Dimensional Bottom Axial Model: BW 15×15 Fuel Assembly
(Dimensions in cm)

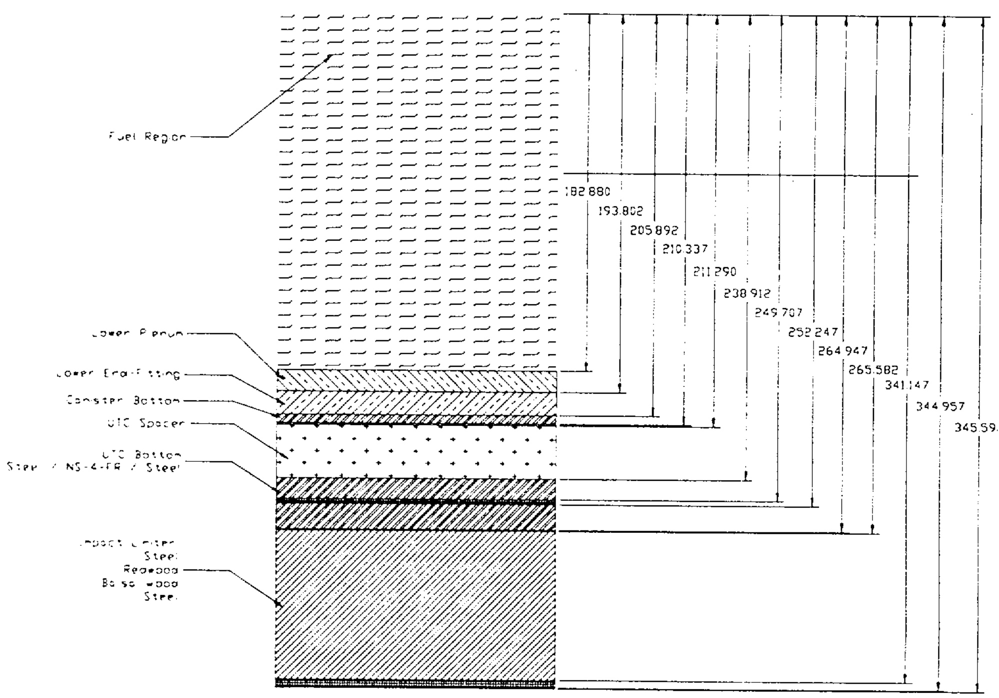


Figure 5.3-10 Transport Cask Bottom Model – PWR Design Basis (Dimensions in cm)

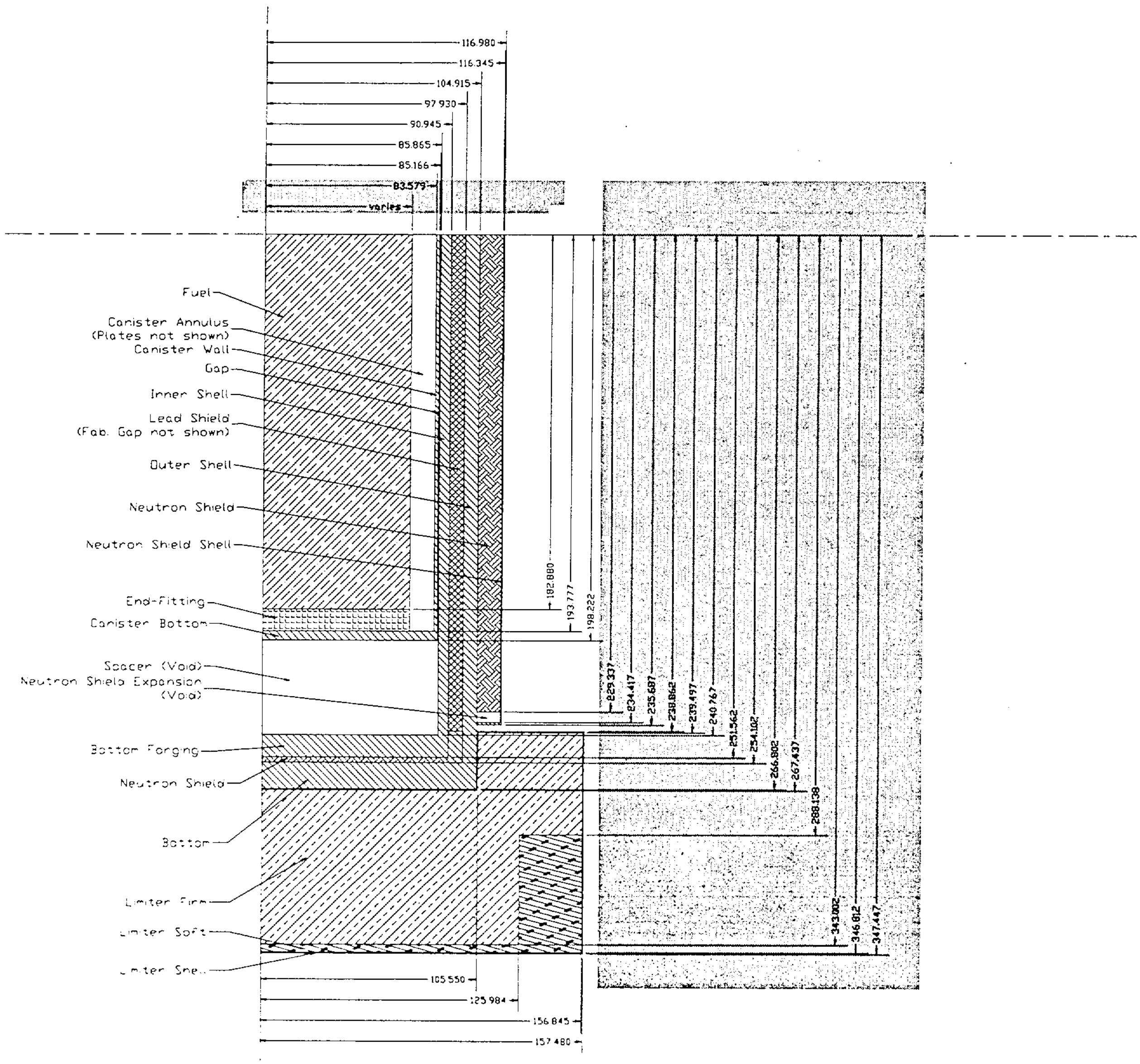


Figure 5.3-11 Transport Cask Bottom Model – BWR Design Basis (Dimensions in cm)

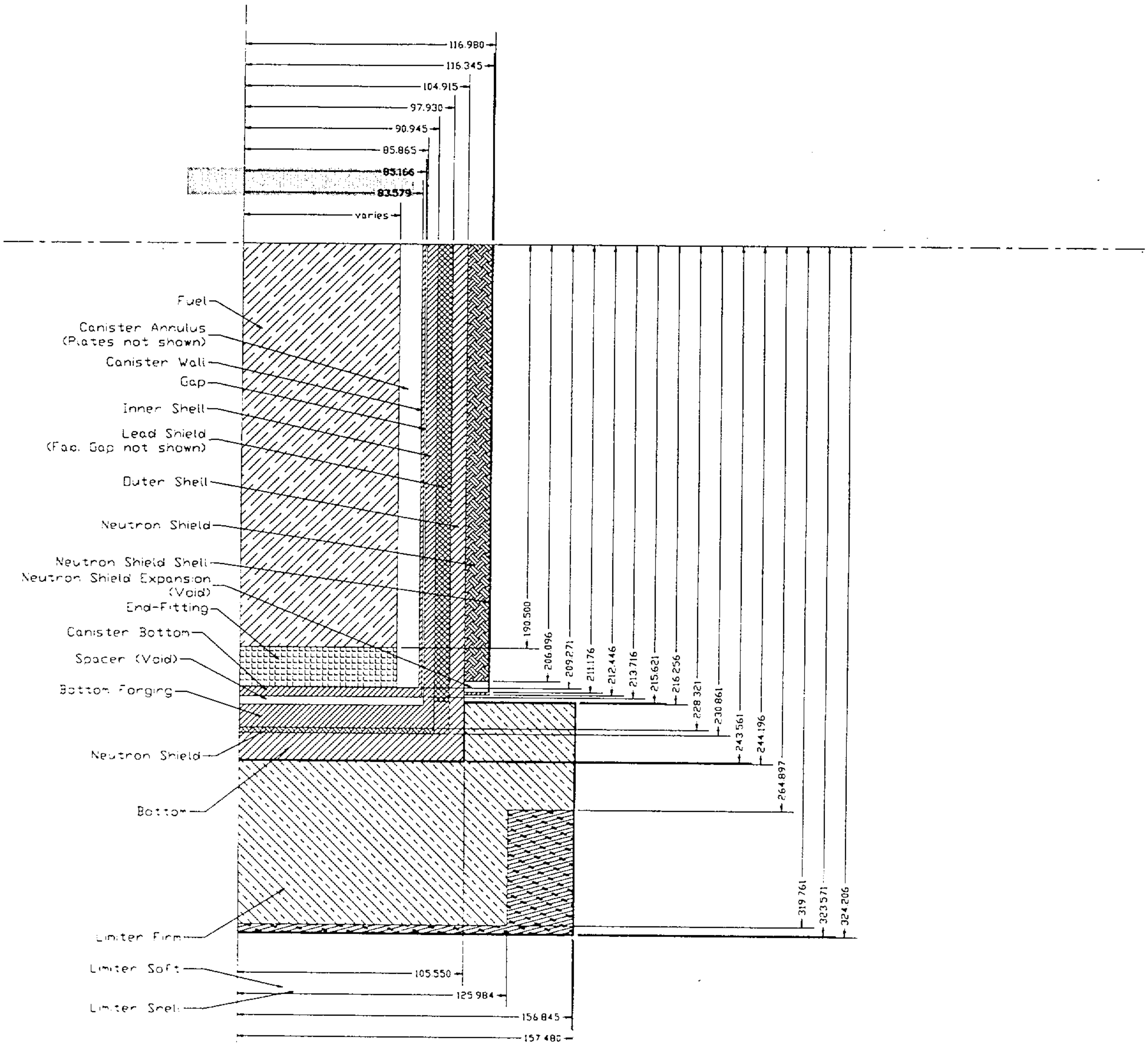


Figure 5.3-12 Transport Cask Top Model – PWR Design Basis (Dimensions in cm)

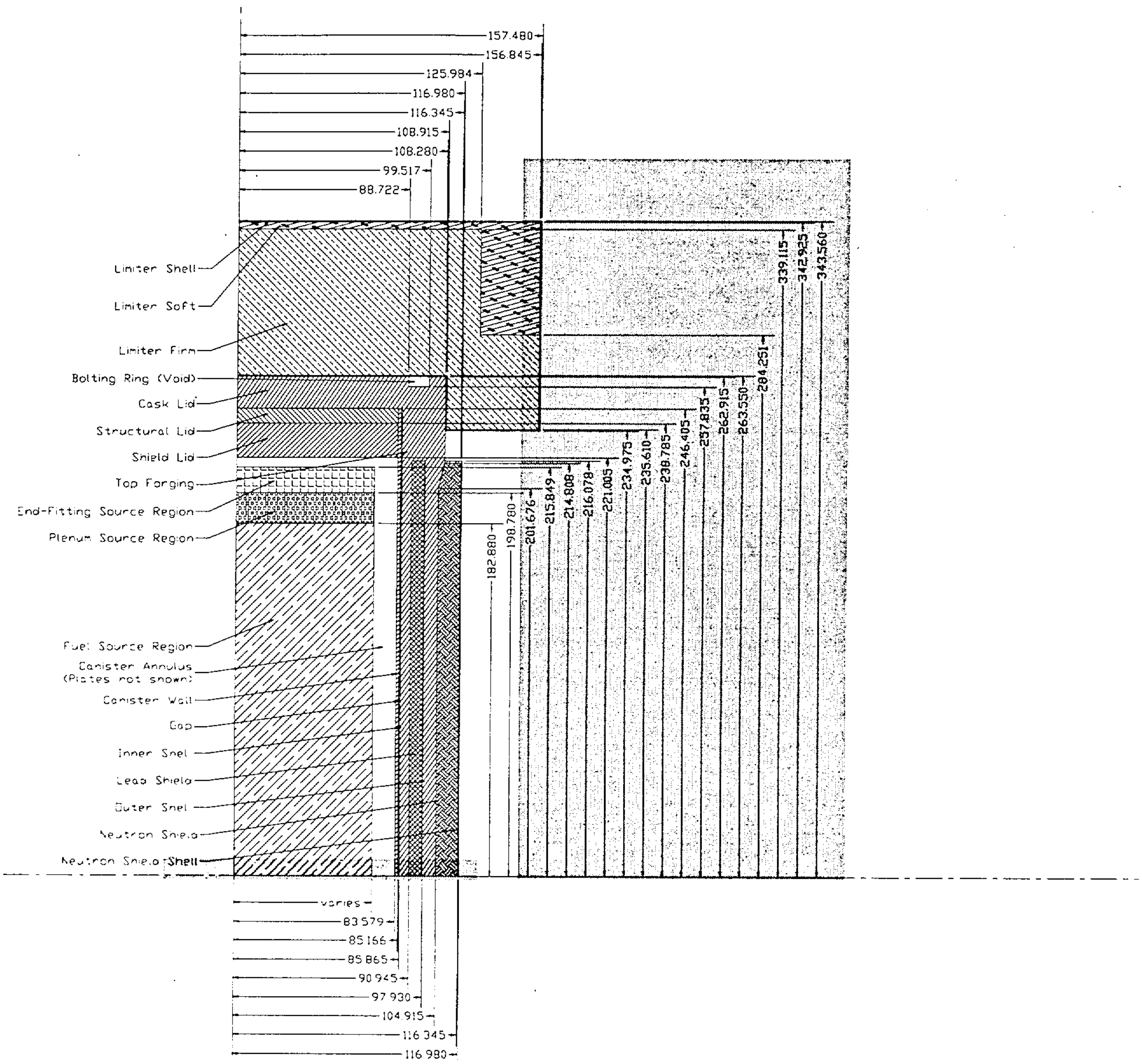


Figure 5.3-13 Transport Cask Top Model – BWR Design Basis (Dimensions in cm)

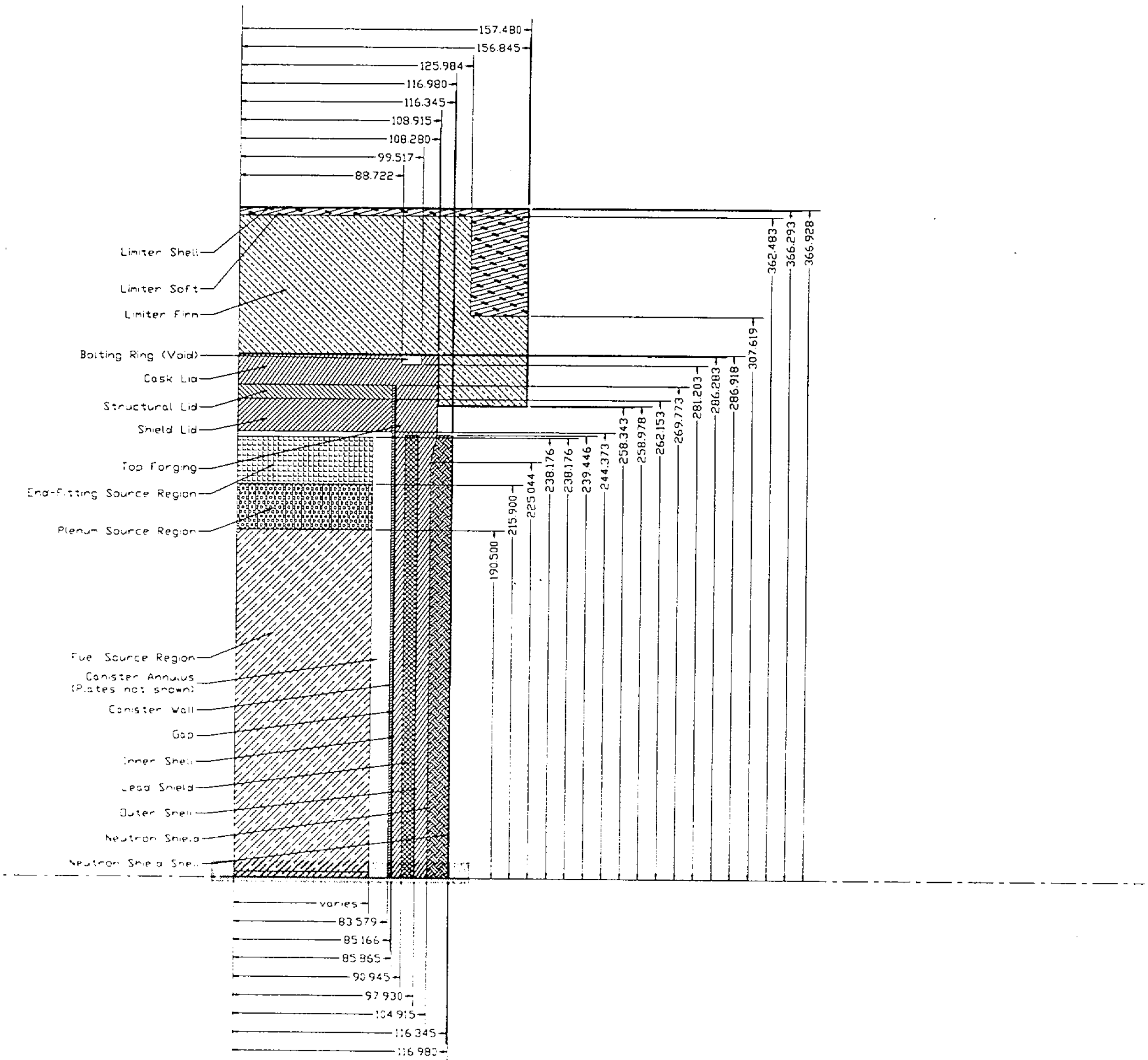


Figure 5.3-14 PICTURE Representation of PWR Top Model – Normal Conditions – Showing Trunnion Recesses and Lid Vent

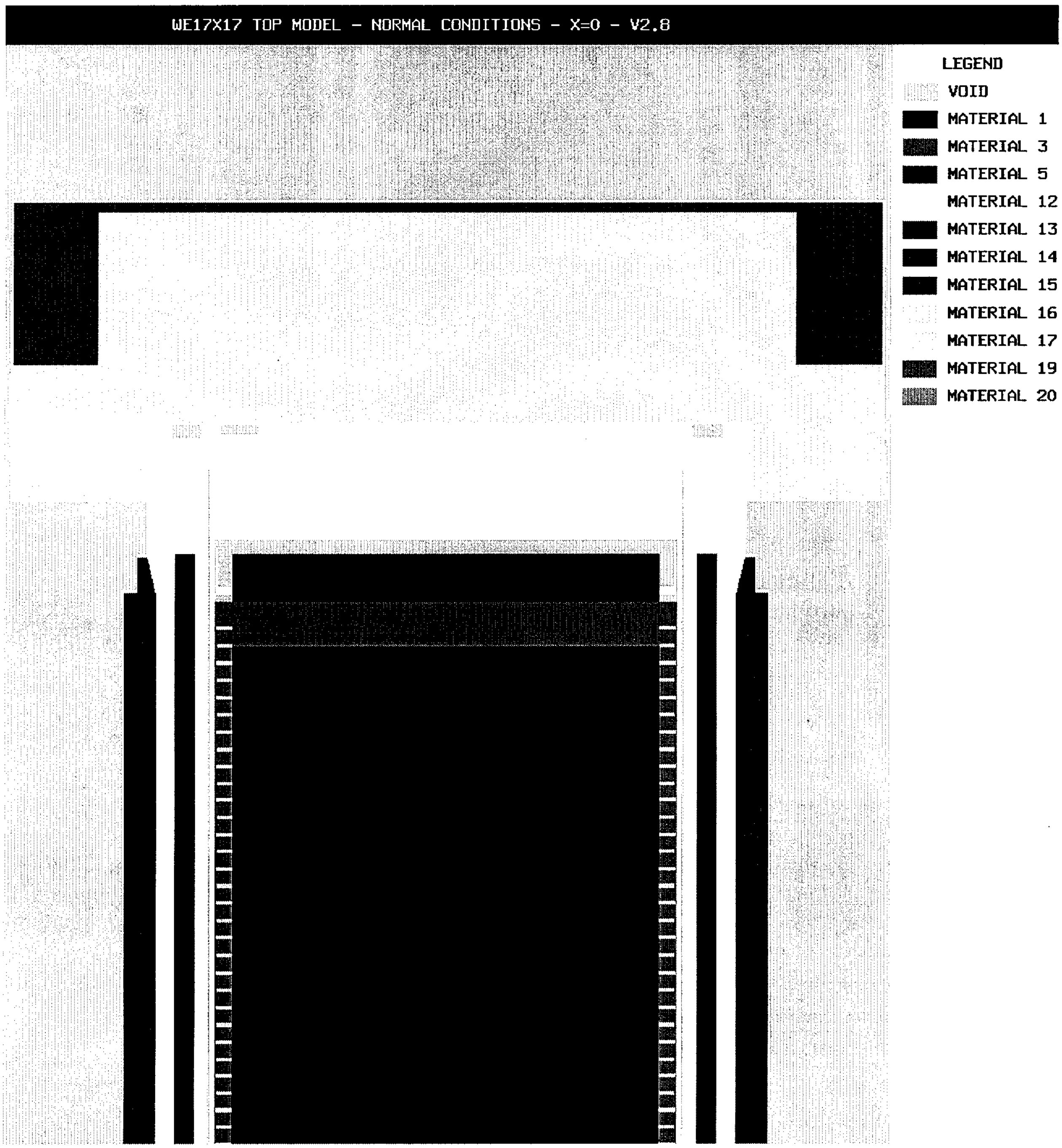


Figure 5.3-15 PICTURE Representation of PWR Bottom Model – Normal Conditions – Slice Through Lower Rotation Pockets

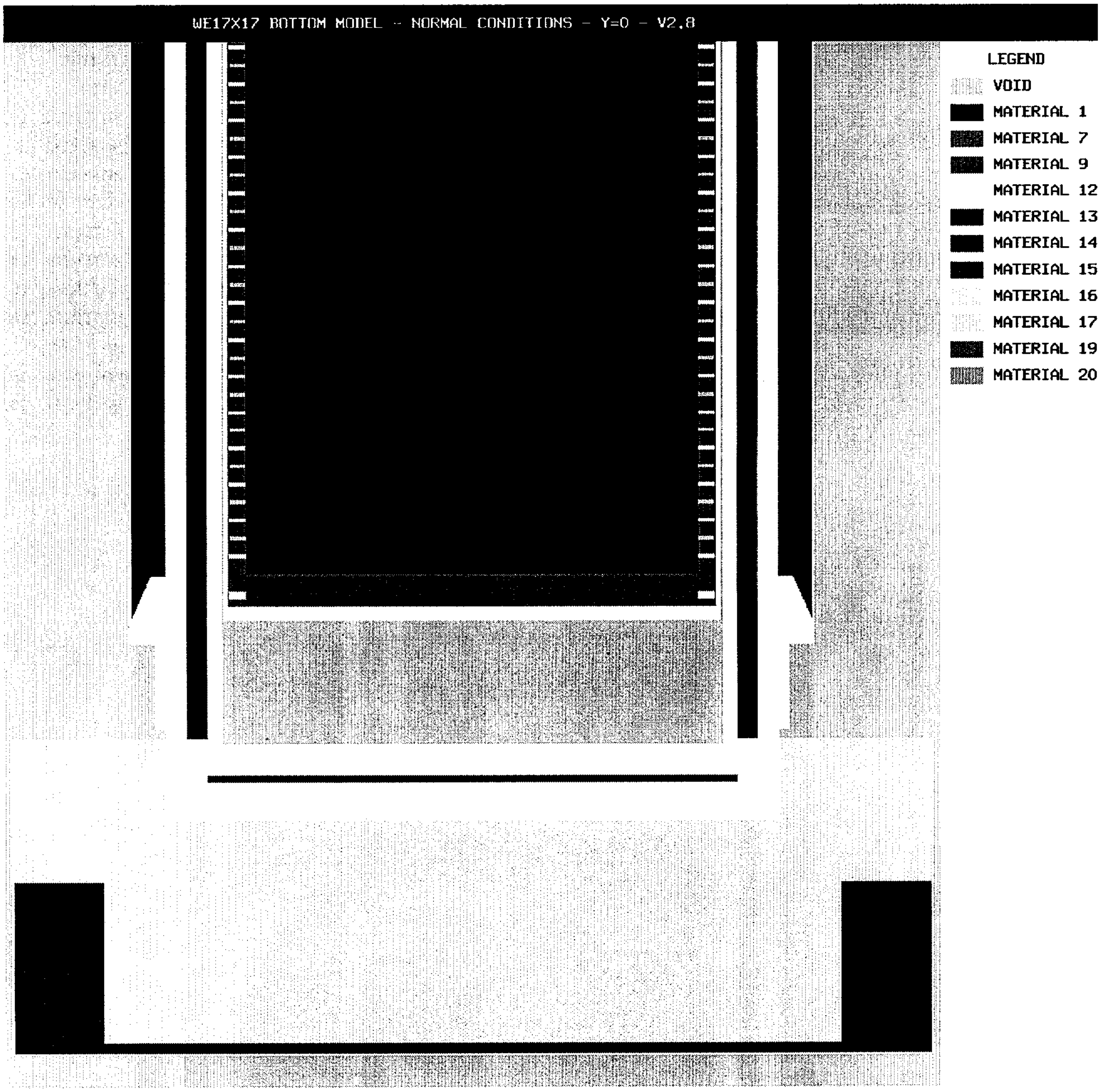


Figure 5.3-16 Radial Lead Slump Model (Dimensions in In.)

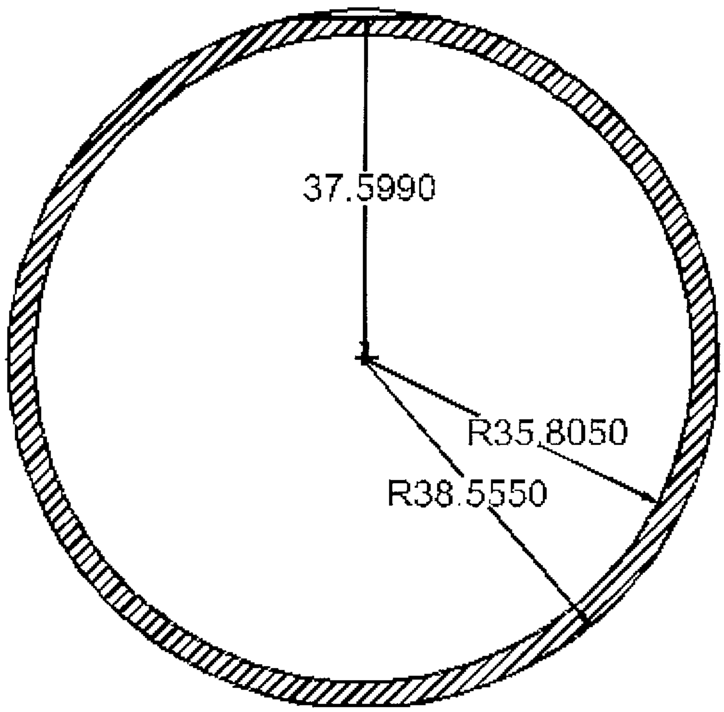


Figure 5.3-17 PICTURE Representation of PWR Top Model – Accident Conditions

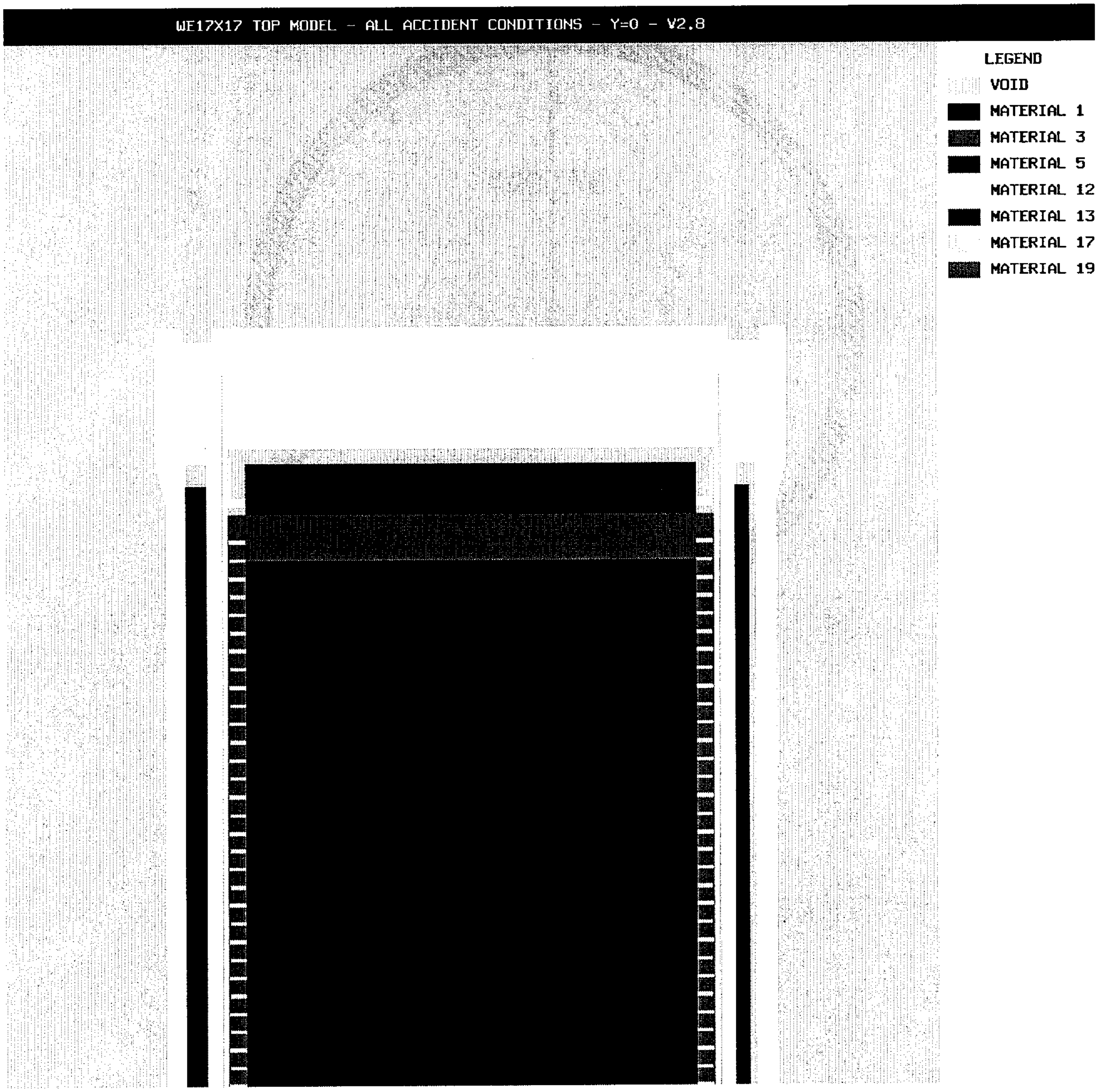


Figure 5.3-18 PICTURE Representation of BWR Fuel Region and Heat Transfer Fin Models at Fuel Axial Midplane

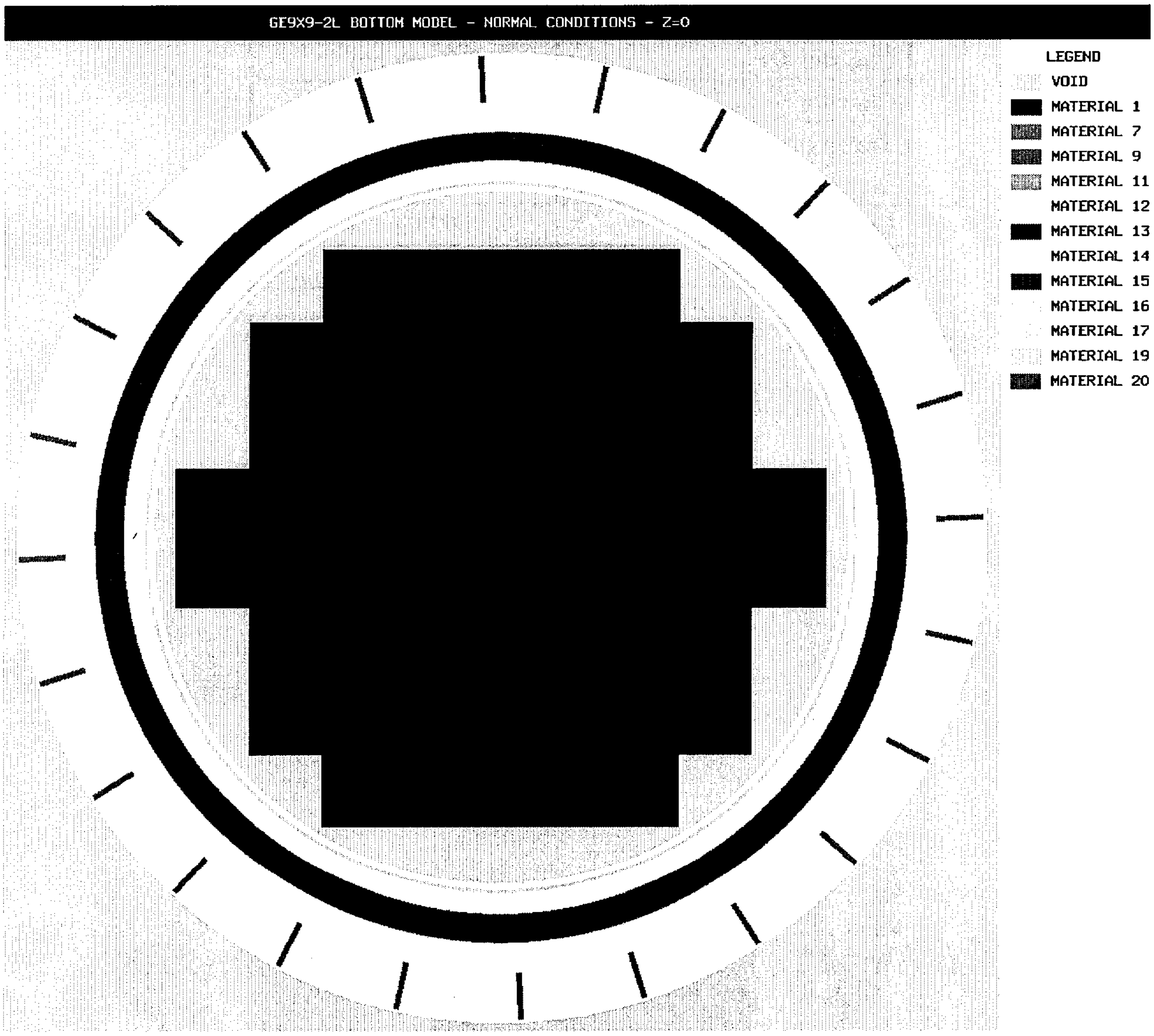


Table 5.3-1 Radial Shield Regions Employed in One-Dimensional Model

Component	Radius Of Region
Canister ID	83.58 cm
Canister Wall Thickness	85.27 cm
Transport Cask ID	85.86 cm
Inner Shell OD	90.94 cm
Lead OD	97.93 cm
Outer Shell OD	104.91 cm
Neutron Shield (NS-4-FR)	116.36 cm
Expansion Space	116.68 cm
Neutron Shield Shell	117.32 cm

Table 5.3-2 Universal Transport Cask One-Dimensional Model Axial Dimensions

Component	Inches	Centimeters
Universal Transport Cask Bottom		
Steel Layer	4.25	10.795
NS-4-FR Layer	1.0	2.54
Steel Layer	5.0	12.7
Transport Cask Lid (Steel)	6.50	16.510
Transport Cask Cavity	192.5	487.68
Cavity Spacer		
Aluminum Disk (Note 1)	1.5	3.81
Steel Plate (Note 1)	0.38	0.9652
Impact Limiter (Top/Bottom)		
Limiter Shell	0.25	0.635
Redwood (Note 2)	29.75	75.565
BALSA	1.5	3.81
Limiter Shell	0.25	0.635
Canister Bottom (Steel)	1.75	4.445
Canister Structural Lid (Steel)	7.0	17.78
Canister Shield Lid (Steel)	3.0	7.62
Canister Length (Note 3)		
Class 1 (PWR)	175.25	445.135
Class 2 (PWR)	184.35	468.249
Class 3 (PWR)	191.95	487.553
Class 4 (BWR)	185.75	471.805
Class 5 (BWR)	190.55	483.997

Note 1 – Aluminum spacer disks are used in the Class 5 (1 disk) and Class 4 (4 disks) canisters. In the Class 1 and 2 canisters a plate supported by concentric steel cylinders forms the spacer. The spacer lengths are 16.75 and 7.65 in., respectively. In the one-dimensional analysis, only the plate is included since the concentric cylinders little effective shielding in the axial direction.

Note 2 – Calculated value using the 43 in. total impact limiter height, 0.25 in. impact limiter shell thickness (two shell thicknesses), the 1.5 in. balsa wood thickness, and the 11.25 in. impact limiter recess.

Note 3 – Minor variations in actual canister length do not significantly change model results.

Table 5.3-3 Source Region Summary for PWR Three-Dimensional Model

Region	Length [cm]	Volume [cm^3]	10 Yr Total Source Rate		
			Neutron [n/sec]	Photon [γ/sec]	Hardware [γ/sec]
Fuel	365.760	5.906E+06	7.514E+09	9.012E+16	9.077E+14
Upper Plenum	15.900	2.568E+05	–	–	1.255E+14
Upper E/F	17.069	2.756E+05	–	–	8.692E+13
Lower E/F	10.897	1.760E+05	–	–	1.155E+14

Table 5.3-4 Source Region Summary for BWR Three-Dimensional Model

Region	Length [cm]	Volume [cm^3]	10 Yr Total Source Rate [/sec]		
			Neutron	Photon	Hardware
Fuel	381.000	6.420E+06	7.560E+09	7.640E+16	2.581E+13
Upper Plenum	25.400	4.280E+05	–	–	7.227E+13
Upper E/F	22.276	3.753E+05	–	–	4.474E+13
Lower E/F	18.771	3.163E+05	–	–	1.529E+14

Table 5.3-5 PWR Top Model Radial Detector Description – Normal Conditions

Detector	ID	Radius [cm]	Lower Elev [cm]	Upper Elev [cm]	Axial Divisions	Azimuthal Divisions
Forging Surface	1	108.280	216.078	234.975	2	1
Trunnion Surface	2	116.980	203.378	216.078	1	16
Cask Surface	3	116.980	0.000	203.378	20	1
Personnel Barrier	4	135.763	0.000	234.975	24	1
Limiter Surface	5	157.480	234.975	262.915	8	1
1m	6	216.980	0.000	443.560	16	1
2m	7	316.980	0.000	543.560	16	1
Railcar+2m	8	357.480	0.000	593.560	16	1
4m	9	516.980	0.000	743.560	16	1

Table 5.3-6 PWR Top Model Axial Detector Description – Normal Conditions

Detector	ID	Elevation [cm]	Inner Rad [cm]	Outer Rad [cm]	Radial Divisions	Azimuthal Divisions
Cask Surf 1	1	343.560	0.000	49.800	1	1
Cask Surf 2	2	343.560	49.800	157.480	9	1
1m 1	3	443.560	0.000	88.582	2	1
1m 2	4	443.560	88.582	216.980	10	1
2m	5	543.560	0.000	357.480	12	1
4m	6	743.560	0.000	357.480	12	1

Table 5.3-7 PWR Bottom Model Radial Detector Description – Normal Conditions

Detector	ID	Radius [cm]	Lower Elev [cm]	Upper Elev [cm]	Axial Divisions	Azimuthal Divisions
N/S Gap Inner	1	104.915	235.687	238.862	1	1
N/S Gap Outer	2	116.980	235.687	238.862	1	1
Trunnion Surface	3	116.980	202.667	235.687	3	16
Cask Surface	4	116.980	0.000	202.667	18	1
Personnel Barrier	5	135.763	0.000	238.862	20	1
Limiter Surface	6	157.480	238.862	266.802	8	1
1m	7	216.980	0.000	447.447	16	1
2m	8	316.980	0.000	547.447	16	1
Railcar+2m	9	357.480	0.000	597.447	16	1
4m	10	516.980	0.000	747.447	16	1

Table 5.3-8 PWR Bottom Model Axial Detector Description – Normal Conditions

Detector	ID	Elevation [cm]	Inner Rad [cm]	Outer Rad [cm]	Radial Divisions	Azimuthal Divisions
Cask Surf 1	1	347.447	0.000	49.800	1	1
Cask Surf 2	2	347.447	49.800	157.480	9	1
1m 1	3	447.447	0.000	88.582	2	1
1m 2	4	447.447	88.582	216.980	10	1
2m	5	547.447	0.000	357.480	12	1
4m	6	747.447	0.000	357.480	12	1

Table 5.3-9 PWR Top Model Radial Detector Description – Accident Conditions

Detector	ID	Radius [cm]	Lower Elev [cm]	Upper Elev [cm]	Axial Divisions	Azimuthal Divisions
Outer Shell	1	104.915	0.000	201.676	24	1
Forging Radius	2	108.280	0.000	262.915	24	1
1m	3	204.915	0.000	362.915	24	1

Table 5.3-10 PWR Top Model Axial Detector Description – Accident Conditions

Detector	ID	Elevation [cm]	Inner Rad [cm]	Outer Rad [cm]	Radial Divisions	Azimuthal Divisions
Outer Shell 1	1	262.915	0.000	48.424	2	1
Outer Shell 2	2	262.915	48.424	108.280	8	1
1m 1	3	362.915	0.000	91.641	2	1
1m 2	4	362.915	91.641	204.915	8	1

Table 5.3-11 PWR Bottom Model Radial Detector Description – Accident Conditions

Detector	ID	Radius [cm]	Lower Elev [cm]	Upper Elev [cm]	Axial Divisions	Azimuthal Divisions
Outer shell	1	104.915	0.000	266.802	24	1
1m	2	204.915	0.000	366.802	24	1

Table 5.3-12 PWR Bottom Model Axial Detector Description – Accident Conditions

Detector	ID	Elevation [cm]	Inner Rad [cm]	Outer Rad [cm]	Radial Divisions	Azimuthal Divisions
Outer Shell 1	1	266.802	0.000	46.919	2	1
Outer Shell 2	2	266.802	46.919	104.915	8	1
1m 1	3	366.802	0.000	91.641	2	1
1m 2	4	366.802	91.641	204.915	8	1

Table 5.3-13 Homogenized Fuel Region Isotopic Composition [atom/b-cm]

Isotope	PWR	BWR
ALUMINUM	1.9953E-03	1.9507E-03
BORON-10	1.9090E-04	5.1195E-05
BORON-11	7.6839E-04	2.0607E-04
CARBON-12	2.3982E-04	1.6127E-04
CHROMIUM(SS304)	8.3776E-04	4.9029E-04
IRON	–	2.0626E-03
IRON(SS304)	2.8532E-03	1.6698E-03
MANGANESE	8.3462E-05	4.8845E-05
NICKEL(SS304)	3.7112E-04	2.1719E-04
OXYGEN-16	9.6038E-03	8.7353E-03
URANIUM-234	2.6411E-07	2.4022E-07
URANIUM-235	3.4574E-05	3.1447E-05
URANIUM-238	4.7671E-03	4.3360E-03
ZIRCALLOY	2.9669E-03	4.4688E-03

Table 5.3-14 Isotopic Compositions of PWR Fuel Assembly Non-Fuel Source Regions [atom/b-cm]

Isotope	Upper Plenum	Upper End Fit	Lower End Fit
CHROMIUM(SS304)	2.2706E-03	2.7589E-03	3.2027E-03
IRON(SS304)	7.7330E-03	9.3961E-03	1.0908E-02
MANGANESE	2.2621E-04	2.7485E-04	3.1907E-04
NICKEL(SS304)	1.0058E-03	1.2222E-03	1.4188E-03
ZIRCALLOY	2.9669E-03	–	–

Table 5.3-15 Isotopic Compositions of BWR Fuel Assembly Non-Fuel Source Regions [atom/b-cm]

Isotope	Upper Plenum	Upper End Fit	Lower End Fit
CARBON-12	7.4574E-05	–	–
CHROMIUM(SS304)	4.8369E-04	1.2561E-03	3.1099E-03
IRON	1.5864E-03	–	–
IRON(SS304)	1.6473E-03	4.2780E-03	1.0592E-02
MANGANESE	4.8188E-05	1.2514E-04	3.0982E-04
NICKEL(SS304)	2.1427E-04	5.5644E-04	1.3776E-03
ZIRCALLOY	4.3248E-03	–	–

Table 5.3-16 Isotopic Compositions of PWR Canister Annular Region Materials (One-Dimensional Analysis Only) [atom/b-cm]

Isotope	Fuel Annulus	Upper Plenum Annulus	Upper End Fit Annulus	Lower End Fit Annulus
ALUMINUM	5.6780E-03	–	–	–
CHROMIUM(SS304)	1.6925E-03	1.3426E-03	2.5012E-03	3.9181E-03
IRON(SS304)	5.7642E-03	4.5725E-03	8.5185E-03	1.3344E-02
MANGANESE	1.6861E-04	1.3375E-04	2.4918E-04	3.9035E-04
NICKEL(SS304)	7.4975E-04	5.9475E-04	1.1080E-03	1.7357E-03

Table 5.3-17 Isotopic Compositions of BWR Canister Annular Region Materials (One-Dimensional Analysis Only) [atom/b-cm]

Isotope	Fuel Annulus	Upper Plenum Annulus	Upper End Fit Annulus	Lower End Fit Annulus
ALUMINUM	3.1336E-03	–	–	–
CARBON-12	6.1200E-04	4.7078E-04	–	–
CHROMIUM(SS304)	–	–	1.9068E-03	2.2629E-03
IRON	1.3019E-02	1.0015E-02	–	–
IRON(SS304)	–	–	6.4942E-03	7.7068E-03
MANGANESE	–	–	1.8997E-04	2.2544E-04
NICKEL(SS304)	–	–	8.4470E-04	1.0024E-03

Table 5.3-18 Isotopic Composition of Additional Shielding Materials [atom/b-cm]

Isotope	Carbon Steel	SS304	Lead	Heat Fin
CARBON-12	3.9250E-03	–	–	–
CHROMIUM(SS304)	–	1.7429E-02	–	9.9587E-03
IRON	8.3498E-02	–	–	–
IRON(SS304)	–	5.9358E-02	–	3.3917E-02
LEAD	–	–	3.2969E-02	–
MANGANESE	–	1.7363E-03	–	9.9214E-04
NICKEL(SS304)	–	7.7207E-03	–	4.4116E-03
COPPER				3.5920E-02

Table 5.3-19 Isotopic Composition of Additional Shielding Materials [atom/b-cm]

Isotope	NS-4-FR	Balsa Wood	Redwood	Aluminum
ALUMINUM	7.7630E-03	–	–	6.0307E-02
BORON-10	8.5530E-05	–	–	–
BORON-11	3.4220E-04	–	–	–
CARBON-12	2.2640E-02	2.7875E-03	8.6301E-03	–
HYDROGEN	5.8540E-02	4.6458E-03	1.4384E-02	–
NITROGEN-14	1.3940E-03	–	–	–
OXYGEN-16	2.6090E-02	2.3229E-03	7.1917E-03	–

THIS PAGE INTENTIONALLY LEFT BLANK

5.4 Shielding Evaluation

The techniques used to perform gamma and neutron dose rate calculations for the Universal Transport Cask are described in the following section. Descriptions of the computer codes and methods that are used in the shielding analyses of the cask are included.

5.4.1 Computer Code Description

The computer codes used in the shielding evaluation of the Universal Transport Cask are the SCALE sequences SAS1 [5], SAS2H [4] and SAS4 [2]. SAS1 is a driver for the XSDRNPM [7] one-dimensional transport code, and SAS2H executes the ORIGEN-S isotopic depletion code. SAS4 prepares input for and executes the MORSE Monte-Carlo shielding code.

Both one-dimensional SAS1 and three-dimensional SAS4 shielding models are used in the evaluations of the Universal Transport Cask. The SAS1 radial and axial models are used to provide a basis for selection of design basis PWR and BWR fuel assemblies. The method of solution employed in the one-dimensional analysis is XSDRNPM discrete ordinates and XSDOSE flux at a point estimation. Bucklings are applied to the SAS1 models to account for transverse leakage. One-dimensional analysis also serves as a cross check to the more complex three-dimensional model results.

The SAS4 three-dimensional shielding models are used to estimate the dose profiles along the surfaces of the cask and are used to estimate dose rates in and around streaming paths such as the rotation pockets. The method of solution is adjoint biased MORSE Monte Carlo. Since SAS4 requires model symmetry at the fuel midplane, two models are created for each cask, a top and a bottom model. Radial biasing is performed to estimate dose rate on the sides of the cask, and axial biasing is performed to estimate dose rates on the top and bottom surfaces of the cask. Modifications are made to SAS4 to tally dose rates along the radial, top and bottom surfaces of the cask as well as any cylindrical surface surrounding the cask. Thus, detailed dose rate profiles are determined that explicitly show peaks due to the fuel burnup profile, activated hardware gamma emission, and streaming paths.

5.4.2 Three-Dimensional Shielding Calculations

The SAS4 code is used to compute dose rate responses for each source region in the model. Individual runs are performed for each of the source conditions described in Table 5.4-1. To determine total dose rates, the results of these individual runs are combined statistically. That is, if individual response r_i with uncertainty σ_i represents the dose rate contribution from source type i on a given detector, then the total dose rate r_T and uncertainty σ_T on the detector are obtained by

$$r_T = \sum_i r_i$$

$$\sigma_T^2 = \sum_i \sigma_i^2$$

In general, SAS4 cases are run with a number of histories sufficient to give at most a 5% relative standard deviation in maximum dose rate on the cask surface detectors. The typical number of histories required to achieve this level of uncertainty is indicated in Table 5.4-1 for each source type.

5.4.2.1 Normal Conditions of Transport

Table 5.4-2 and Table 5.4-3 show the surface average dose rates on various surface detectors for the PWR top radial and axial models, respectively. These results show the relative contributions to dose rate from the various source types analyzed.

Peak subdetector dose rate results for each surface detector are shown in Table 5.4-4 and Table 5.4-5 for the PWR top and bottom radial models, respectively, under normal conditions of transport. A detailed description of each detector may be found by using the detector identification number provided in the applicable table and consulting the appropriate detector description table given in Section 5.3.

Axial dose rate profile results are shown in Figure 5.4-1 for the design basis PWR fuel under normal conditions of transport.

Top and bottom radial dose rate profiles are shown in Figure 5.4-2 and Figure 5.4-3, respectively, for the design basis PWR fuel. These profiles represent the radial variation of dose

rates on the surface of disk-shaped detectors located at the indicated distance from the end of the cask. The corresponding maximum detector responses are provided in Table 5.4-6 and Table 5.4-7 for the top and bottom models, respectively. Dose rate results are well below regulatory limits on the axial ends of the cask.

The BWR axial dose rate profile is shown in Figure 5.4-4 for the combined top and bottom models. The sharp increase in cask surface dose rate near the bottom of the cask is due to the adjacent position of the lower end-fitting and Transport Cask rotation pockets. Figure 5.4-5 shows the azimuthal surface dose rate profile at the elevation of the rotation pockets. Maximum dose rates for the various BWR detector configurations are provided in Table 5.4-8 through Table 5.4-11 for normal conditions of transport.

BWR radial dose rate profiles are shown in Figure 5.4-6 and Figure 5.4-7 for top and bottom models under normal conditions of transport.

The BWR model shows a computed dose rate of 225.6 (±3.5%) mrem/hr at the cask outer shell surface in the narrow circumferential gap between the neutron shield shell and the lower impact limiter. This gap is approximately 1.25 in. wide and 4.75 in. deep, and is inaccessible. The analysis in Section 5.4.2.2 shows that this dose rate is highly localized, and that the dose rate at the gap opening (at the radius of the neutron shield shell) is 82.9 (±2.7%).

The effect of the heat transfer fins in the neutron shield is illustrated by the azimuthal dose plot shown in Figure 5.4-8. Observe that near the fins a localized peak in neutron dose rate is offset by a localized depression in gamma contribution. The net result is that no significant streaming effect is observed through the fins. The peaks in the total dose profile (at 0°, 90°, 180° and 270°) are due to the geometry of the canister basket, which places the fuel closer to the canister shell at these locations.

5.4.2.2 Bounding Analysis of Class 3 Canister

The absence of a spacer assembly in the Class 3 canister configuration suggests that the design basis PWR analysis may not provide bounding dose rates near the bottom of the cask. To evaluate this situation, a special analysis is performed. In this analysis, a canister loaded with design basis PWR fuel is placed in the Transport Cask with no spacer assembly. Axial dose profiles under normal conditions are shown in Figure 5.4-9.

A maximum radial surface dose rate of 133.3 (±1.5%) mrem/hr occurs at the opening of the narrow, inaccessible gap between the neutron shield shell and the inside surface of the lower impact limiter. At the surface of the bottom forging inside this narrow gap, the dose rate is somewhat higher, 393.0 (±2.6%) mrem/hr. However, this dose rate is not considered significant because of its inaccessible location and the rapidity with which the dose rate decreases radially outward from the forging surface.

5.4.2.3 Accident Conditions

Results for the accident conditions are shown in Table 5.4-12 through Table 5.4-20 for the PWR and BWR models. Dose rates do not exceed the regulatory limit of 1000 mrem/hr at any point 1 m from the cask surface.

5.4.3 Loading Table Analysis

The design basis fuel descriptions characterize Transport Cask dose and heat generation rates at nominal conditions of burnup and initial enrichment for a particular PWR and BWR fuel types. In order to extend the results to other fuel types and various combinations of initial enrichment and burnup, a detailed analysis is conducted that determines, for any given fuel type, enrichment and burnup combination, the minimum cool time required for radiation dose and heat generation rates to fall below the design basis values. The analysis models the source term for a series of enrichment/burnup/cool time combinations based on SAS2H results, and the dose rate evaluation is made on the basis of computed one-dimensional dose rates which explicitly consider the effects of radiation spectrum and cask shielding properties rather than a simple source rate comparison.

For clarity, the various fuel assembly types considered are classified according to array size, with the results for the most limiting fuel type within each array size taken as the minimum required cool time for all assemblies within that class. The result of the analysis is a fuel assembly loading table showing the required cool time for any combination of fuel array size, initial enrichment, and burnup. An exception is made for CE 14x14 fuel, which is analyzed separately in support of the Maine Yankee. The CE 14x14 assembly envelopes the CE, Westinghouse and Exxon/ANF fuel in the Maine Yankee inventory, based on fuel and hardware mass.

5.4.3.1 Methodology

The loading table analysis extends the applicability of the initial 10-year cooled 45,000 MWD/MTU PWR and 40,000 MWD/MTU BWR shielding design basis evaluation by providing minimum cool times for 30,000 MWD/MTU to 50,000 MWD/MTU burned fuel assemblies in increments of 5,000 MWD/MTU. In addition to the burnup range the loading table evaluation also includes minimum initial enrichment limits ranging from 1.9 to 3.7 wt % ^{235}U in 0.2 wt % ^{235}U increments. Changes in the initial enrichment can have a substantial impact on the actinide neutron source of the spent nuclear fuel and, thereby, modifies the minimum cool times required to meet the decay heat and dose rate limits imposed on the transport cask.

The fuel types analyzed in the loading table analysis include the candidate design basis fuel assemblies listed in Section 5.1.1. Note that the analysis of BWR fuel types considers only the longer BWR-4/6 type fuel, which bound the shorter BWR-2/3 type fuels.

A complete set of source spectra and decay heat values for this set of limiting assemblies is then computed using the SAS2H [4] code at various initial enrichment, burnup, and cool times representative of the fuel intended for shipment in the UMS Transport Cask.

Next, the cool time required for each fuel type, initial enrichment, and burnup combination to meet limiting values of decay heat and dose rate is determined. The decay heat limits are set based on the cool-time dependent decay heat limits established in Section 3.4.6, as shown in Table 3.4-8. The dose rate limits are established based on a one-dimensional analysis of the Transport Cask containing the design basis PWR and BWR fuels under both normal and accident conditions.

With the limiting cool times identified for each fuel combination, the results are further summarized by identifying the most limiting fuel type within each array size classification. This array size classification is intended to simplify the application of the loading tables in determining the suitability of a particular fuel for shipment without need for a detailed analysis.

The SCALE computer code system is used to evaluate radiation source terms and to perform one dimensional shielding calculations. Source terms are evaluated using the SAS2H code, which provides a simplified interface to the ORIGEN-S code, including burnup-dependent cross-section processing. Source spectra at additional cool times are evaluated by direct application of the ORIGEN-S code. The SAS1 code sequence is used to determine one-dimensional dose rates and

the dose rate response functions. The SAS1 driver provides an interface to the XSDRNPM one-dimensional transport code. All SCALE analyses are conducted using the SCALE 27N18G group library.

The key analytical assumption made in this analysis is the validity of extending one-dimensional dose rate comparisons to conclusions about the dose rate field in the vicinity of the cask. This assumption is supported by the following:

1. The geometry of the cask system is essentially the same regardless of the fuel type loaded. This is particularly true when the dissipating effects on dose rate of the cask shielding materials are considered. Possible concerns about an unfavorable geometry (e.g., a fuel assembly end fitting adjacent to an area of minimal shielding) occurring for a particular fuel assembly have been considered both implicitly in the design of the system and directly in the three dimensional shielding analyses conducted for the design basis fuels.

2. The one-dimensional radial dose rates used for the comparison are themselves accurate predictions of actual dose rates. The ratio of the length of the cask to its diameter ensures that the buckling approximation implicit in a one-dimensional calculation is valid. This assumption is further justified based on the results of a three-dimensional verification study.

Furthermore, the one-dimensional dose rate comparisons are made on the basis of fuel region sources alone. This analysis is intended to consider the impact of the relationship between initial enrichment and burnup on actinide production in the active fuel region. The three-dimensional shielding calculations performed for the Transport Cask incorporate the maximum plenum and end-fitting hardware descriptions of all assemblies intended for shipment in the UMS system.

Hence, the non-fuel source regions have already been considered to the maximum extent, and it is only necessary to demonstrate that the fuel region sources are bounded by the design basis.

This argument is also applied to the suitability of considering dose rates computed only for the radial case. Since no aspect of the cask shield construction changes with varying fuel type, it is

reasonable to expect the axial dose contribution from fuel sources to vary consistently with the radial contribution.

SAS2H runs are executed for each assembly type at the following combinations of burnup, initial enrichment, and cool time:

Burnup:	30, 35, 40, 45, 50 GWD/MTU
Enrichment:	1.9, 2.1, 2.3, 2.5, 2.7, 2.9, 3.1, 3.3, 3.5, 3.7 wt % ^{235}U
Cool Time:	5, 6, 7, 8, 9, 10, 12, 14, 16, 18, 20, 22, 24, 26, 28, 30, 35, 40 years

Final cool times are established by interpolating between results calculated for each cool time listed above. This interpolation procedure is conservative due to the exponentially decreasing behavior of decay heat and radiation source rates with time. The maximum cool time (that is, the cool time which ensures all constraints are met) is always rounded up to the next whole year.

The various combinations of enrichment and burnup shown above define a discrete mesh of possible combinations. When considering the required cool time for a particular assembly, the actual enrichment of the assembly should be rounded down to the next lower analyzed value, and the assembly burnup should be rounded up to the next higher analyzed value in order to ensure that a conservative value is obtained from the loading table.

5.4.3.2 Limiting Decay Heat and Dose Rate Values

The decay heat limits applied in the analysis are the cool-time dependent decay heat limits established in Section 3.4.6 based on clad temperature limits as a function of fuel cool time.

The maximum allowable heat load, or decay heat limit as used in the context of this chapter, is based on the more limiting of (1) cladding stress and temperature limited heat loads and (2) the overall maximum decay heat limit of 20 kW (PWR) or 16 kW (BWR). Details of this analysis are presented in Section 3.4.6.

Decay heat limits for fuel burned to less than 35 GWD/MTU are not determined in Section 3.4.6. For the minimum cooling time evaluation presented here, the 30 GWD/MTU decay heat limit is conservatively set equal to the 35 GWD/MTU decay heat limit values. Since the heat load limit is based on the rod cladding stress and pressure level, which is directly related to fission gas

production, employing the 35 GWD/MTU limit for lower burnups clearly provides bounding limits.

Normal condition dose rate limits are established using the SAS1 code by computing the Transport Cask one-dimensional radial dose rate for the design basis PWR and BWR fuel descriptions. The SAS1 geometrical models are described in Section 5.3.1. The dose rate limits are established based on the computed dose rate from fuel region sources taken at a distance 2m from the edge of a 124 in. wide railcar (357.48 cm from the cask axial centerline). The resulting one-dimensional dose rate limits are 6.71 mrem/hr for PWR fuels and 4.53 mrem/hr for BWR fuels.

Since the design basis PWR and BWR fuels lead to accident condition dose rates that are much lower than regulatory limits, the approach of basing the dose rate limits on one-dimensional models of these design basis assemblies is unnecessarily conservative. Instead, a fixed one-dimensional dose rate limit of 600 mrem/hr at 1 meter from the Transport Cask is employed for the accident condition dose rate limit. This value is verified by performing three-dimensional analyses of selected fuel, burnup, and enrichment cases at the decay times resulting from the analysis. The results indicate that the 600 mrem/hr limit is sufficient to ensure that actual three-dimensional computed dose rates are below the regulatory limit of 1,000 mrem/hr at 1 meter from the cask.

It is not the intent to compare one-dimensional dose rates directly with the three-dimensional design basis values. Instead, the one-dimensional dose rates computed for each fuel combination are compared with the corresponding one-dimensional dose rates evaluated for the design basis PWR and BWR fuel descriptions. The detailed three-dimensional analysis of the design basis fuels shows that these fuels meet regulatory limits. Hence, the one-dimensional dose rates evaluated for the design basis fuels are used as the basis of comparison. This approach is verified by performing detailed three-dimensional analyses of selected fuel combinations at the cool time indicated by the loading table analysis to ensure that the design basis fuel dose rates are bounding.

5.4.3.3 Cool Time Determination

The strategy used to determine limiting cooling times for each combination of fuel type, initial enrichment, and burnup is to:

1) Determine decay heat and dose rate values at each cooling time step.
2) Interpolate in the resulting collection of data to find minimum cooling time required to meet each limiting value, decay heat and transfer and storage cask dose rate, individually.
3) Select the maximum of this collection of minimum required cooling times, rounded up to the next whole year, as the minimum required cooling time for this combination of burnup, enrichment and cooling time.

The minimum cool times required to meet each limit are rounded to the nearest tenth of a year and the required cool time is determined by taking the maximum of these values and rounding up to the next whole year.

While the determination of the minimum cooling time needed to meet dose rate limits is straightforward, the determination of the limiting cooling time for decay heat is complicated by the fact that the decay heat limit varies with cooling time. Specifically, as described in Section 3.4.6, the allowable cladding temperature decreases as a function of cooling time. Therefore, the allowable decay heat also decreases. In addition, the decay heat limits are specified at relatively few cooling time values in comparison with the number of cooling time steps at which decay heats are evaluated. The following algorithm is used to determine the limiting cooling time:

a) Starting with the minimum cooling time in the decay heat limit table, find the first tabulated decay heat limit cooling time at which the actual decay heat is less than the limiting value.
b) Consider the resulting decay heat value as the decay heat limit for this case.
c) Interpolate against the assembly decay heat data to find the cooling time at which this decay heat limit is met.

This strategy is valid because: a) the allowable temperature, and therefore the allowable decay heat, never increases and; b) the assembly decay heat curve always decreases more rapidly than the decay heat limit curve. Hence, the interpolation procedure will identify a cooling time at which the actual assembly decay heat value is less than the limiting decay heat value associated with that cooling time.

5.4.3.4 Results

Results of the loading table analysis are shown in Table 5.4-21 for PWR fuels and in Table 5.4-22 for BWR fuels. Note that the loading tables provided here are intended for generic PWR

and BWR fuel. As shown in Section 5.5.1, information about the actual site specific fuel inventory may be used to develop loading tables that are less restrictive.

The results of a verification study are shown in Table 5.4-23 for normal conditions of transport and in Table 5.4-24 for accident conditions. Here, selected fuel combinations from the loading tables are analyzed in full three-dimensional detail at the required fuel cool times. For the normal conditions analysis, the results show that in each case the computed maximum dose rates are less than the design basis values. Under accident conditions, the verification results substantiate the selection of a one-dimensional dose rate limit of 600 mrem/hr.

Figure 5.4-1 PWR Total Dose Rate – Side – Normal Conditions

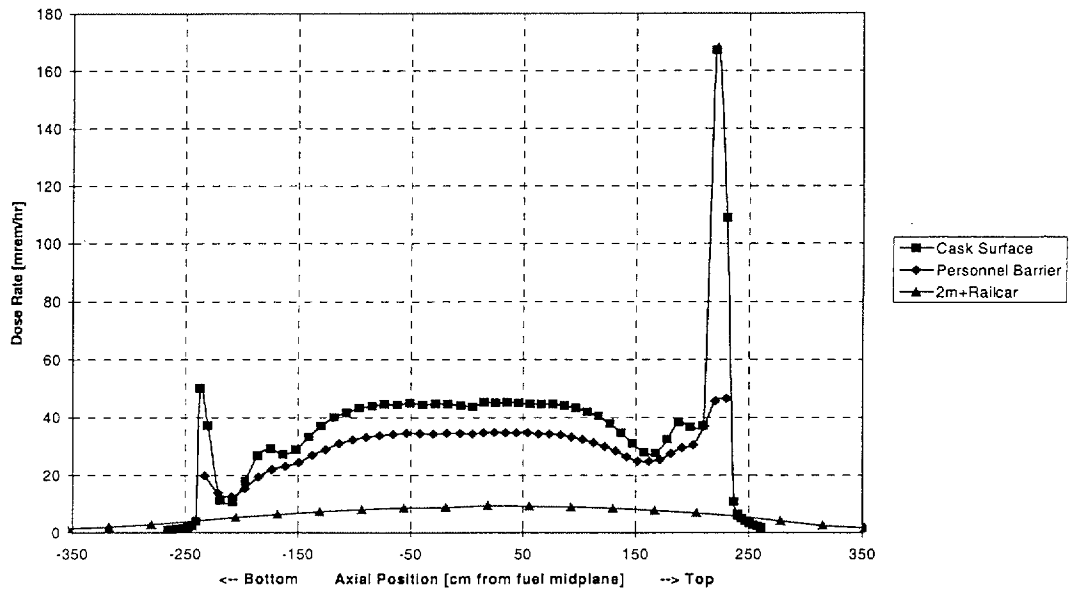


Figure 5.4-2 PWR Total Dose Rate – Top – Normal Conditions

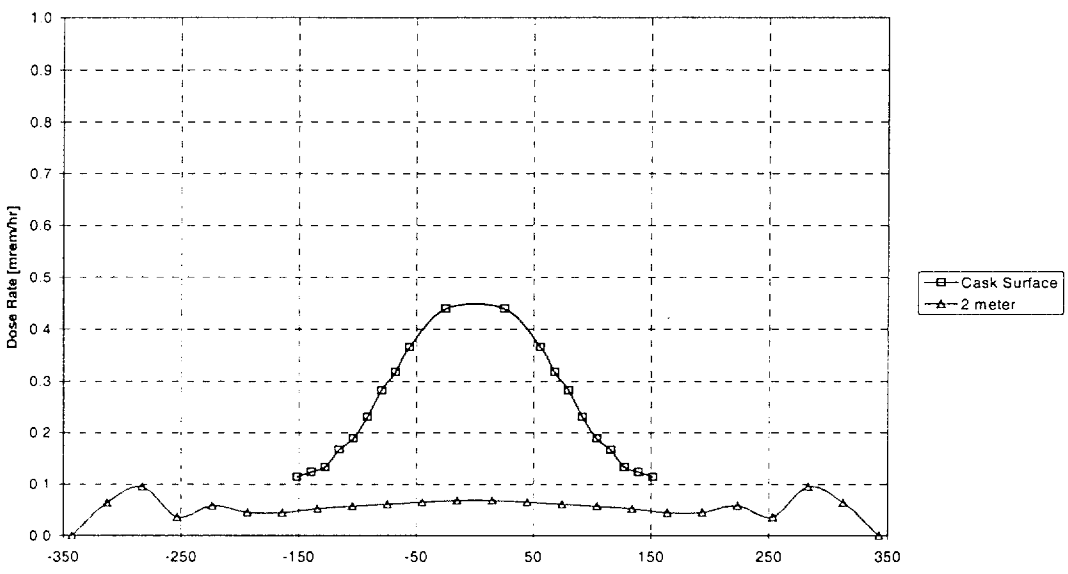


Figure 5.4-3 PWR Total Dose Rate – Bottom – Normal Conditions

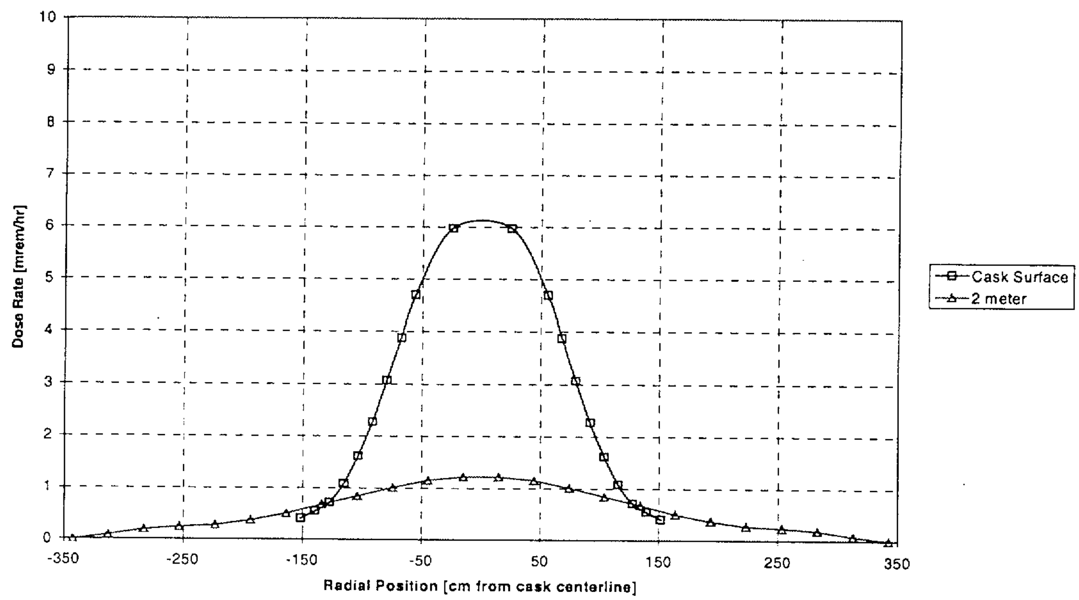


Figure 5.4-4 BWR Total Dose Rate – Side – Normal Conditions

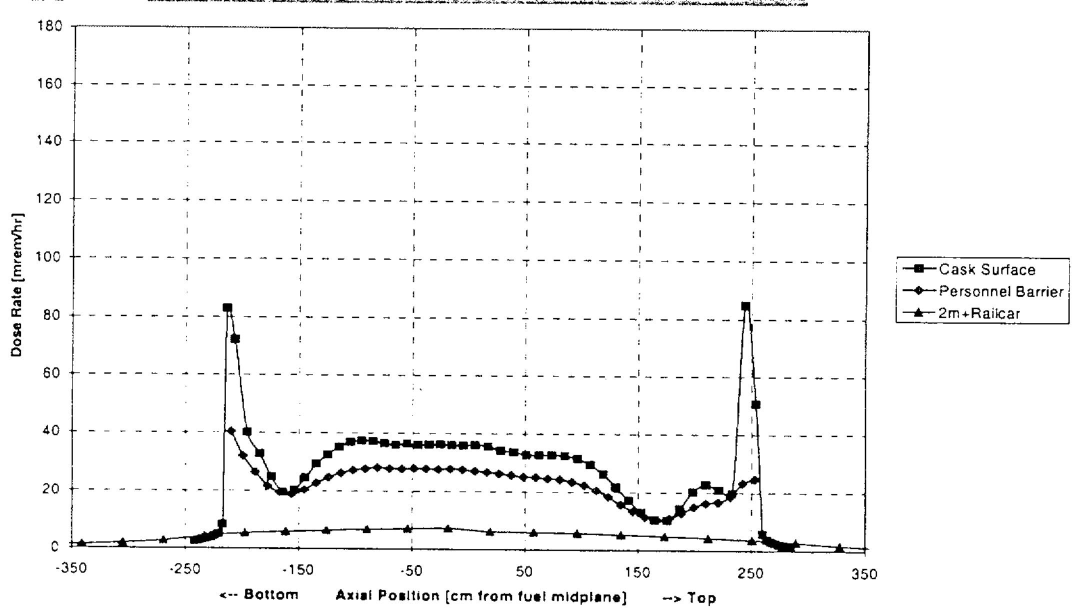


Figure 5.4-5 BWR Total Dose Rate – Azimuthal Profile at Rotation Pocket Elevation – Normal Conditions

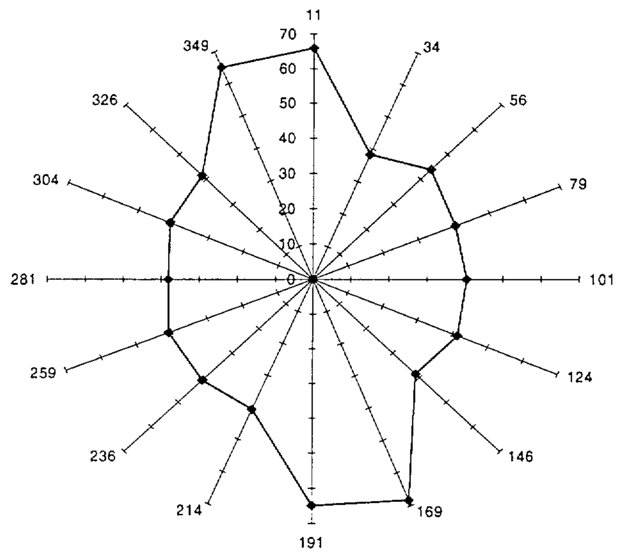


Figure 5.4-6 BWR Total Dose Rate – Top – Normal Conditions

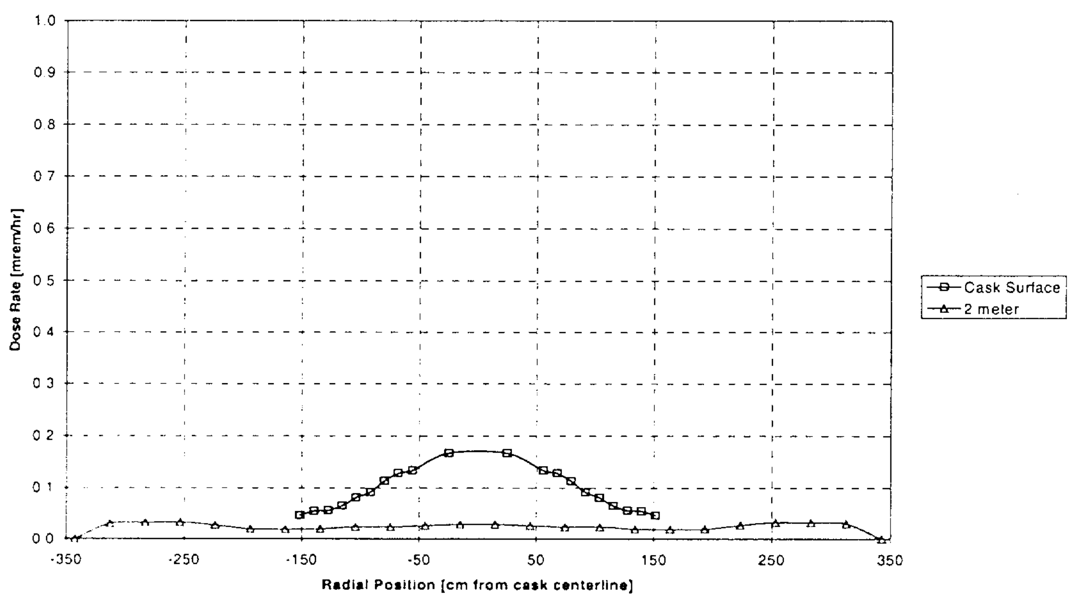


Figure 5.4-7 BWR Total Dose Rate – Bottom – Normal Conditions

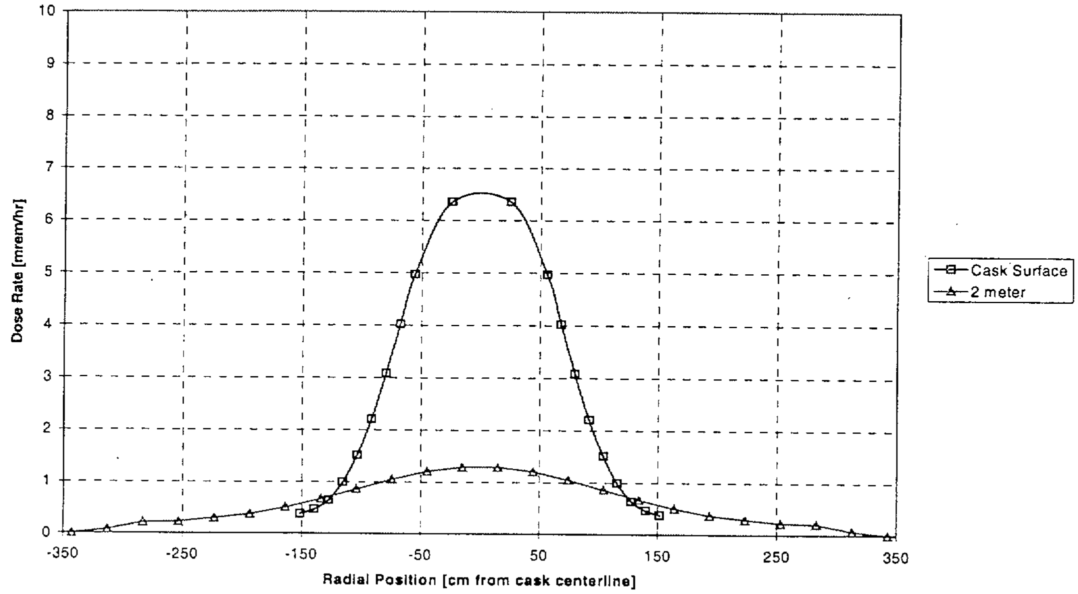


Figure 5.4-8 Effect of Heat Transfer Fins on Cask Surface Fuel Neutron and Gamma Dose Rates [mrem/hr]

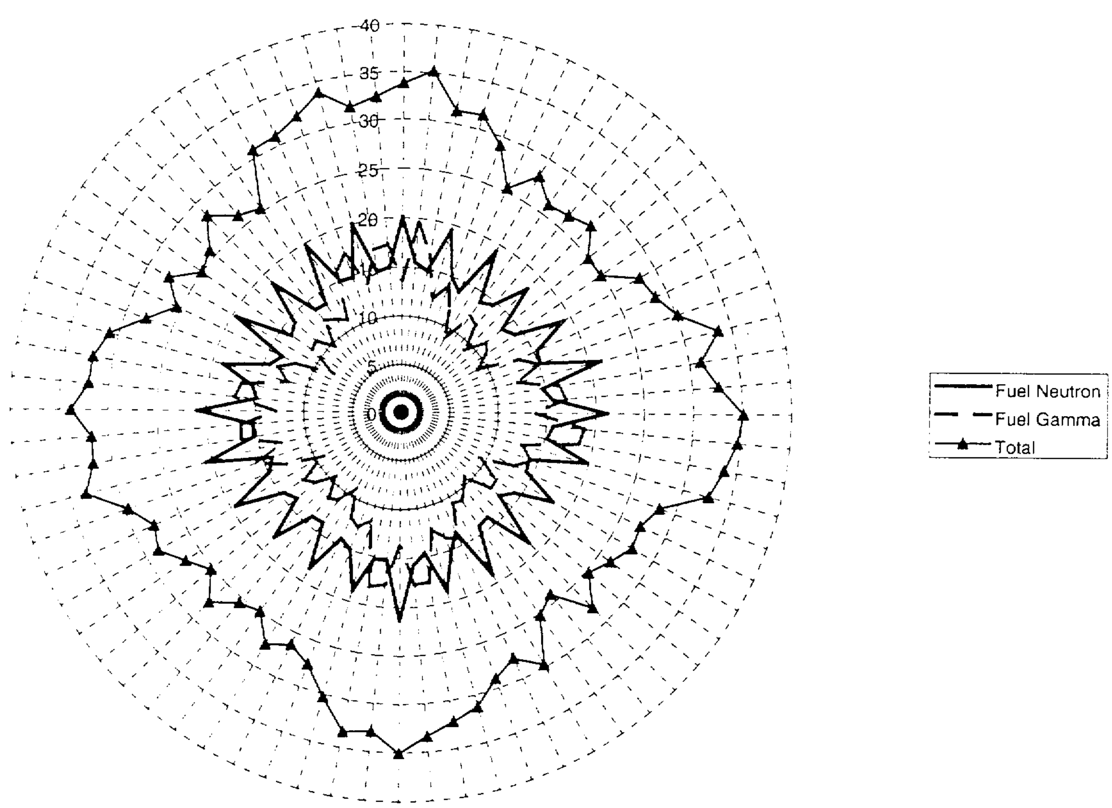


Figure 5.4-9 PWR Canister with No Spacer Assembly - Side Total Dose Rate Profile (Lower Half)

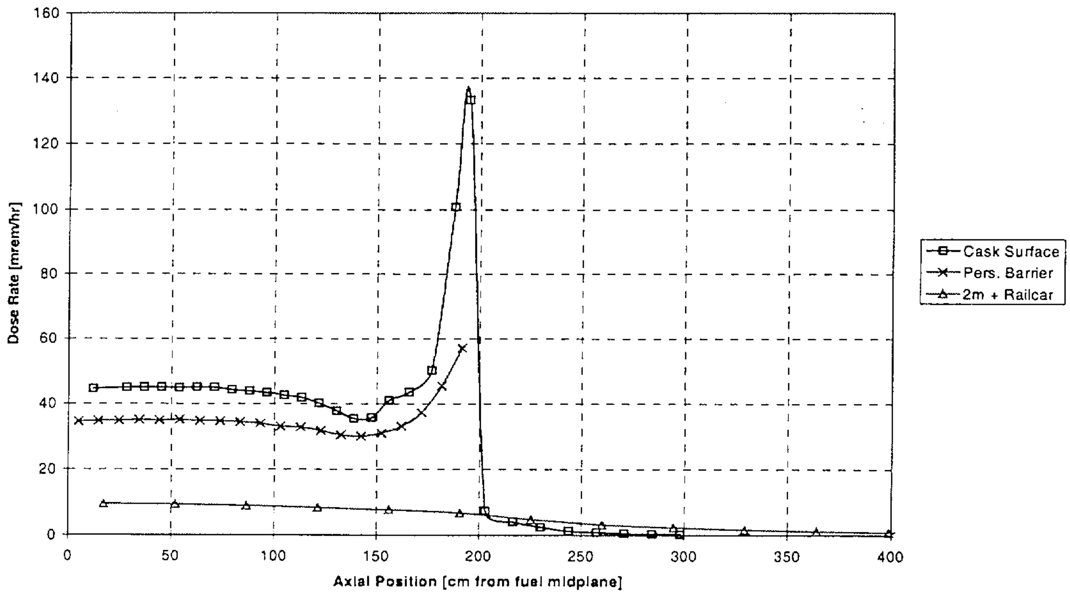


Table 5.4-1 SAS4 Runs Performed for Each Source Region

Source Region	Source Type	Histories (approx.)
Fuel	Neutron	3×10^6
Fuel	N-Gamma	6×10^5
Fuel	Gamma	12×10^6
Fuel Hardware	Gamma	12×10^6
Plenum	Gamma	12×10^6
End Fitting	Gamma	12×10^6

Table 5.4-2 PWR Surface Average Dose Rates – Top Model – Radial Detectors – Normal Conditions

Region	Source	Average Dose Rate [mrem/hr]					
		Cask Surface		Personnel Barrier		2 m Position	
Fuel	Neutron	12.77	(0.3%)	11.48	(0.5%)	1.42	(0.4%)
Fuel	N-Gamma	4.50	(0.3%)	3.17	(0.3%)	0.41	(0.3%)
Fuel	Gamma	11.16	(0.3%)	8.19	(0.2%)	1.13	(0.2%)
Hardware	Gamma	5.95	(0.3%)	4.37	(0.2%)	0.61	(0.2%)
Plenum	Gamma	3.62	(0.2%)	2.86	(0.2%)	0.39	(0.2%)
End-Fitting	Gamma	1.79	(0.3%)	2.85	(0.3%)	0.39	(0.2%)
Total		39.8	(0.1%)	32.9	(0.2%)	4.4	(0.2%)

Table 5.4-3 PWR Surface Average Dose Rates – Top Model – Axial Detectors – Normal Conditions

Region	Source	Average Dose Rate [mrem/hr]			
		Limiter Surface		2 m	
Fuel	Neutron	0.02	(0.7%)	0.03	(9.3%)
Fuel	N-Gamma	0.19	(1.1%)	0.03	(2.2%)
Fuel	Gamma	<0.01	(1.7%)	<0.01	(2.1%)
Hardware	Gamma	<0.01	(1.6%)	<0.01	(2.1%)
Plenum	Gamma	<0.01	(1.7%)	<0.01	(2.2%)
End-Fitting	Gamma	0.02	(3.7%)	<0.01	(4.7%)
Total		0.3	(1.0%)	<0.1	(4.6%)

Table 5.4-4 PWR Maximum Subdetector Dose Rates – Top Model – Radial Detectors – Normal Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Forging Surface	1	220.8	180.0	167.2	(1.8%)	101.1	(2.9%)	66.2	(0.6%)
N-Shield Shell	2,3	209.7	258.8	45.9	(4.0%)	16.6	(11.0%)	29.3	(1.1%)
Personnel Barrier	4	230.1	180.0	46.6	(1.6%)	23.2	(3.1%)	23.5	(0.5%)
Limiter Surface	5	236.7	180.0	10.9	(2.2%)	4.5	(5.3%)	6.5	(0.7%)
2m from Railcar	8	18.6	180.0	9.6	(0.4%)	3.2	(0.9%)	6.4	(0.3%)
4m from Cask	9	23.2	180.0	5.3	(0.4%)	1.7	(1.0%)	3.7	(0.3%)

Table 5.4-5 PWR Maximum Subdetector Dose Rates – Bottom Model – Radial Detectors – Normal Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Limiter Gap	1	237.3	180.0	158.8	(4.6%)	151.7	(4.8%)	7.1	(4.2%)
N-Shield Shell	2,3,4	237.3	180.0	50.0	(4.2%)	46.0	(4.5%)	4.1	(3.5%)
Personnel Barrier	5	17.9	180.0	34.7	(0.5%)	12.8	(0.8%)	21.9	(0.5%)
Limiter Surface	6	240.6	180.0	3.9	(3.7%)	2.4	(6.0%)	1.6	(1.9%)
2m from Railcar	9	18.7	180.0	9.0	(0.3%)	3.0	(0.6%)	6.0	(0.3%)
4m from Cask	10	23.4	180.0	4.8	(0.4%)	1.6	(0.8%)	3.3	(0.3%)

Table 5.4-6 PWR Maximum Subdetector Dose Rates – Top Model – Axial Detectors – Normal Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Limiter Surface	1,2	24.9	180.0	0.5	(1.9%)	<0.1	(1.4%)	0.5	(2.0%)
1m	3,4	22.2	180.0	0.2	(2.9%)	<0.1	(1.2%)	0.2	(3.0%)
2m	5	283.0	180.0	<0.1	(11.4%)	<0.1	(14.9%)	<0.1	(3.5%)
4m	6	342.6	180.0	<0.1	(13.8%)	<0.1	(17.8%)	<0.1	(2.8%)

Table 5.4-7 PWR Maximum Subdetector Dose Rates – Bottom Model – Axial Detectors – Normal Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Limiter Surface	1,2	24.9	180.0	6.0	(0.4%)	<0.1	(1.7%)	6.0	(0.4%)
1m	3,4	22.2	180.0	2.4	(0.5%)	<0.1	(1.8%)	2.4	(0.5%)
2m	5	14.9	180.0	1.3	(1.0%)	<0.1	(3.2%)	1.3	(1.0%)
4m	6	14.9	180.0	0.5	(1.6%)	<0.1	(5.1%)	0.5	(1.6%)

Table 5.4-8 BWR Maximum Subdetector Dose Rates – Top Model – Radial Detectors – Normal Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Forging Surface	1	244.2	180.0	84.8	(2.0%)	34.5	(4.6%)	50.4	(1.1%)
N-Shield Shell	2,3	5.7	180.0	36.1	(0.6%)	17.1	(0.9%)	19.0	(0.9%)
Personnel Barrier	4	5.4	180.0	27.1	(0.6%)	12.7	(0.9%)	14.4	(0.7%)
Limiter Surface	5	260.1	180.0	5.9	(2.1%)	1.4	(7.9%)	4.5	(1.2%)
2m from Railcar	8	19.3	180.0	6.3	(0.4%)	2.6	(0.7%)	3.8	(0.4%)
4m from Cask	9	24.0	180.0	3.5	(0.4%)	1.3	(0.9%)	2.2	(0.4%)

Table 5.4-9 BWR Maximum Subdetector Dose Rates – Top Model – Axial Detectors – Normal Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Limiter Surface	1,2	24.9	180.0	0.2	(2.7%)	<0.1	(1.6%)	0.2	(2.8%)
1m	3,4	22.2	180.0	<0.1	(5.3%)	<0.1	(1.6%)	<0.1	(5.6%)
2m	5	253.2	180.0	<0.1	(15.3%)	<0.1	(24.1%)	<0.1	(3.9%)
4m	6	342.6	180.0	<0.1	(17.6%)	<0.1	(24.7%)	<0.1	(4.6%)

Table 5.4-10 BWR Maximum Subdetector Dose Rates – Bottom Model – Radial Detectors – Normal Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Limiter Gap	1	214.0	180.0	225.6	(3.5%)	156.4	(5.0%)	69.2	(1.1%)
N-Shield Shell	2,3,4	214.0	180.0	82.9	(2.7%)	56.0	(4.0%)	27.0	(0.9%)
Personnel Barrier	5	210.2	180.0	40.4	(1.5%)	19.8	(3.0%)	20.7	(0.5%)
Limiter Surface	6	217.4	180.0	8.4	(2.3%)	3.8	(4.9%)	4.6	(1.1%)
2m from Railcar	9	17.9	180.0	7.3	(0.4%)	3.4	(0.8%)	4.0	(0.4%)
4m from Cask	10	22.6	180.0	4.0	(0.5%)	1.8	(1.0%)	2.3	(0.5%)

Table 5.4-11 BWR Maximum Subdetector Dose Rates – Bottom Model – Axial Detectors – Normal Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Limiter Surface	1,2	24.9	180.0	6.4	(0.4%)	<0.1	(2.0%)	6.4	(0.4%)
1m	3,4	22.2	180.0	2.6	(0.6%)	<0.1	(2.2%)	2.6	(0.6%)
2m	5	14.9	180.0	1.3	(1.0%)	<0.1	(4.3%)	1.3	(1.0%)
4m	6	14.9	180.0	0.5	(1.7%)	<0.1	(5.6%)	0.5	(1.7%)

Table 5.4-12 PWR Surface Average Dose Rates – Top Model – Radial Detectors – Accident Conditions

Region	Source	Shell Surface		1 m	
		Dose Rate [mrem/hr]	Relative Std. Dev.	Dose Rate [mrem/hr]	Relative Std. Dev.
Fuel	Neutron	968.83	(0.1%)	205.16	(0.1%)
Fuel	N-Gamma	2.23	(0.5%)	0.47	(0.3%)
Fuel	Gamma	52.35	(0.4%)	12.98	(0.2%)
Hardware	Gamma	26.10	(0.3%)	6.48	(0.2%)
Plenum	Gamma	13.48	(0.4%)	6.74	(0.6%)
EndFit	Gamma	5.25	(0.9%)	21.02	(0.4%)
Total		1068.3	(0.1%)	252.9	(0.1%)

Table 5.4-13 PWR Maximum Subdetector Dose Rates – Top Model – Radial Detectors – Accident Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Cask Surface	1	37.8	180.0	1420.1	(0.3%)	1324.1	(0.4%)	96.0	(1.1%)
Top Forging	2	213.6	180.0	1279.5	(0.6%)	137.4	(0.9%)	1142.2	(0.7%)
1m from Cask	3	7.6	180.0	461.5	(0.2%)	416.7	(0.2%)	44.8	(0.5%)

Table 5.4-14 PWR Maximum Subdetector Dose Rates – Bottom Model – Radial Detectors – Accident Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Cask Surface	1	16.7	180.0	1409.8	(0.3%)	1313.9	(0.3%)	95.9	(1.0%)
1m from Cask	2	7.6	180.0	458.5	(0.2%)	417.7	(0.2%)	40.9	(0.6%)

Table 5.4-15 PWR Maximum Subdetector Dose Rates – Top Model – Axial Detectors – Accident Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Cask Surface	1,2	12.1	180.0	62.6	(0.7%)	62.0	(0.7%)	0.7	(1.3%)
1m	3,4	197.8	180.0	20.4	(2.8%)	20.3	(2.9%)	<0.1	(8.6%)

Table 5.4-16 PWR Maximum Subdetector Dose Rates – Bottom Model – Axial Detectors – Accident Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Cask Surface	1,2	11.7	180.0	114.8	(0.6%)	39.2	(1.0%)	75.6	(0.7%)
1m	3,4	22.9	180.0	39.5	(0.4%)	10.9	(0.6%)	28.7	(0.4%)

Table 5.4-17 BWR Maximum Subdetector Dose Rates – Top Model – Radial Detectors – Accident Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Cask Surface	1	4.7	180.0	1452.8	(0.3%)	1395.9	(0.3%)	57.0	(1.5%)
Top Forging	2	6.0	180.0	1291.5	(0.3%)	1239.3	(0.3%)	52.3	(1.2%)
1m from Cask	3	8.1	180.0	432.5	(0.2%)	408.2	(0.2%)	24.3	(0.7%)

Table 5.4-18 BWR Maximum Subdetector Dose Rates – Bottom Model – Radial Detectors – Accident Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Cask Surface	1	218.2	180.0	2209.0	(1.4%)	102.9	(1.1%)	2106.2	(1.4%)
1m from Cask	2	7.2	180.0	497.9	(0.2%)	472.7	(0.2%)	25.3	(0.8%)

Table 5.4-19 BWR Maximum Subdetector Dose Rates – Top Model – Axial Detectors – Accident Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Cask Surface	1,2	12.1	180.0	21.0	(1.2%)	20.7	(1.2%)	0.4	(2.0%)
1m	3,4	197.8	180.0	12.4	(3.9%)	12.3	(3.9%)	<0.1	(8.9%)

Table 5.4-20 BWR Maximum Subdetector Dose Rates – Bottom Model – Axial Detectors – Accident Conditions

Location	Det IDs	Band [cm]	Angle [deg]	Detector Responses [mrem/hr]					
				Total		Neutron		Gamma	
Cask Surface	1,2	11.7	180.0	102.5	(0.7%)	31.9	(1.1%)	70.7	(0.8%)
1m	3,4	22.9	180.0	39.5	(0.4%)	8.7	(0.7%)	30.8	(0.4%)

Table 5.4-21 Loading Table for PWR Fuel

Enrichment	Burnup ≤ 30 GWD/MTU Minimum Cool Time [years]				30 < Burnup ≤ 35 GWD/MTU Minimum Cool Time [years]			
wt % ^{235}U (E)	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	8	8	7	8	10	11	9	10
2.1 ≤ E < 2.3	7	8	6	7	10	10	8	10
2.3 ≤ E < 2.5	7	7	6	7	9	10	8	9
2.5 ≤ E < 2.7	7	7	6	7	9	9	7	8
2.7 ≤ E < 2.9	7	7	6	7	8	9	7	8
2.9 ≤ E < 3.1	7	7	6	6	8	8	7	8
3.1 ≤ E < 3.3	6	7	6	6	8	8	7	7
3.3 ≤ E < 3.5	6	6	6	6	7	8	6	7
3.5 ≤ E < 3.7	6	6	6	6	7	7	6	7
3.7 ≤ E ≤ 4.2	6	6	6	6	7	7	6	7

Enrichment	35 < Burnup ≤ 40 GWD/MTU Minimum Cool Time [years]				40 < Burnup ≤ 45 GWD/MTU Minimum Cool Time [years]			
wt % ^{235}U (E)	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	15	15	13	15	20	21	20	20
2.1 ≤ E < 2.3	13	14	12	13	19	19	18	19
2.3 ≤ E < 2.5	12	13	11	12	17	19	17	17
2.5 ≤ E < 2.7	12	12	10	11	16	18	15	17
2.7 ≤ E < 2.9	11	11	9	11	15	18	14	17
2.9 ≤ E < 3.1	10	10	9	10	14	18	13	15
3.1 ≤ E < 3.3	10	10	9	10	13	17	13	15
3.3 ≤ E < 3.5	9	10	8	9	12	17	13	15
3.5 ≤ E < 3.7	9	10	8	9	11	17	12	15
3.7 ≤ E ≤ 4.2	8	10	8	9	11	15	12	14

Table 5.4-22 Loading Table for BWR Fuel

Enrichment	Burnup ≤ 30 GWD/MTU Minimum Cool Time [years]			30 < Burnup ≤ 35 GWD/MTU Minimum Cool Time [years]			35 < Burnup ≤ 40 GWD/MTU Minimum Cool Time [years]			40 < Burnup ≤ 45 GWD/MTU Minimum Cool Time [years]		
wt % ^{235}U (E)	9 x 9	8 x 8	7 x 7	9 x 9	8 x 8	7 x 7	9 x 9	8 x 8	7 x 7	9 x 9	8 x 8	7x 7
1.9 ≤ E ≤ 2.1	8	8	8	14	13	15	24	23	25	34	33	35
2.1 ≤ E ≤ 2.3	7	7	8	12	12	13	21	20	22	31	30	32
2.3 ≤ E ≤ 2.5	7	7	7	11	10	11	19	18	20	29	28	29
2.5 ≤ E ≤ 2.7	7	6	7	9	9	10	17	16	17	26	25	27
2.7 ≤ E ≤ 2.9	6	6	6	9	8	9	14	14	15	24	23	24
2.9 ≤ E ≤ 3.1	6	6	6	8	8	8	13	12	13	21	20	22
3.1 ≤ E ≤ 3.3	6	6	6	7	7	8	11	11	12	19	18	20
3.3 ≤ E ≤ 3.5	6	6	6	7	7	7	10	10	11	17	16	18
3.5 ≤ E ≤ 3.7	6	6	6	7	7	7	10	9	10	15	14	16
3.7 ≤ E ≤ 4.0	6	6	6	7	7	7	10	9	10	14	13	15

Table 5.4-23 Results of Verification Study for Normal Conditions

Fuel Type	Burnup [GWD/MTU]	Enrichment [wt % ^{235}U]	Decay Time [y]	Max Dose Rate [mrem/hr] Railcar+2m Rate	FSD
WE17x17	35	2.7	8	8.62	0.43%
WE17x17	30	3.7	6	7.51	0.52%
B&W15x15	50	3.7	14	8.52	0.41%
CE14x14	30	1.9	6	8.67	0.49%
CE14x14	45	3.7	8	8.95	0.51%
WE14x14	40	3.7	8	8.91	0.42%
WE15x15	45	2.3	18	8.82	0.53%
WE17x17	50	2.1	26	8.29	0.49%
WE14x14	50	1.9	28	8.61	0.50%
CE16x16	40	2.1	12	8.51	0.46%
CE16x16	50	1.9	28	8.43	0.48%
GE8x8	40	3.1	10	6.43	0.43%
GE8x8	30	1.9	8	6.40	0.40%

Table 5.4-24 Results of Verification Study for Accident Conditions

Fuel Type	Burnup [GWD/MTU]	Enrichment [wt % ^{235}U]	Decay Time [y]	Max Dose Rate [mrem/hr] 1 meter from Cask Surface Rate	FSD
WE17x17	50	2.1	26	799.55	0.14%
WE14x14	50	1.9	28	790.44	0.14%
CE16x16	40	2.1	12	597.05	0.14%
CE16x16	50	1.9	28	786.33	0.14%
GE8x8	30	1.9	8	374.63	0.15%

5.5 Appendices

THIS PAGE INTENTIONALLY LEFT BLANK

5.5.1 Site Specific Contents Shielding Evaluations

This section describes fuel assembly characteristics and configurations, or waste configurations, which are unique to specific reactor sites. These site specific content configurations result from conditions that occurred during reactor operations, participation in research and development programs, testing programs intended to improve reactor operations, and from decommissioning activities.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation of the configuration.

5.5.1.1 Maine Yankee Site Specific Spent Fuel

This analysis considers both assembly fuel sources and sources from activated non-fuel material such as control element assemblies (CEA), in-core instrument (ICI) thimbles, and fuel assemblies containing activated stainless steel replacement (SSR) rods. It also considers the consolidated fuel present in the Maine Yankee spent fuel inventory.

The Maine Yankee spent fuel inventory also contains fuel assemblies with hollow zirconium rods, removed fuel rods, axial blankets, poison rods, variable radial enrichment, and low enriched substitute rods. These components do not result in additional sources to be considered in shielding evaluations and are, therefore, enveloped by the standard fuel assembly evaluation. For shielding considerations of the variably enriched rods the planar-average enrichment should be employed in determining minimum cool times.

5.5.1.1.1 Fuel Source Term Description

Maine Yankee utilized 14x14 array size fuel based on designs provided by Combustion Engineering, Westinghouse, and Exxon Nuclear. The previously analyzed Combustion Engineering CE14x14 Standard fuel design is selected as the design basis for this analysis because its potential Uranium loading is the highest of the three vendor fuel types, based on a 0.3765-inch nominal fuel pellet diameter, a 137 inch active fuel length, and a 95% theoretical fuel density. This results in a fuel mass of 0.4307 MTU. This exceeds the maximum reported Maine Yankee fuel mass of 0.397 MTU, and therefore, produces bounding source terms. The

SAS2H model of the CE14x14 assembly at a nominal burnup of 45,000 MWD and initial enrichment of 3.7 wt %, based on data provided in Table 1.2-4, is shown in Figure 5.5.1.1-1.

Source terms for various combinations of burnup and initial enrichment are computed by adjusting the SAS2H BURN parameter to model the desired burnup and specifying the initial enrichment in the Material Information Processor input for UO_2.

5.5.1.1.1.1 Control Element Assemblies (CEA)

For the CEA evaluation, the assumptions are:

1. The irradiated portion of the CEA assembly is limited to the CEA tips, as during normal operation, the elements are retracted from the core and only the tips are subject to significant neutron flux.
2. The CEA tips are defined as that portion present in the "Gas Plenum" neutron source region in the Characteristics Database (CDB) [9].
3. Material subject to activation in the CEA tips is limited to stainless steel Inconel, and the AgInCd absorber material present in the lower eight inches of the CEA.
4. All stainless steel and Inconel material is assumed to have a concentration of 1.2 g/kg ^{59}Co. The CDB indicates that a total of 2.495 kg/CEA of this material is present in the Gas Plenum region of the core during operation.
5. The mass of AgInCd present in each CEA tip is 2.767 kg/CEA [9]. The AgInCd material is modeled as 80 wt. % Ag, 15.35 wt. % In, and 5.35 wt. % Cd. Note that the composition sums to a value greater than 100%, but this only means that conservatively more mass is represented than is actually present.
6. The irradiated CEA material is assumed to be present in the bottom eight inches of the active fuel region when inserted in the assembly.
7. The decay heat generated in the most limiting CEA at a 5-year cool time is 2.16 W/kg of stainless steel or Inconel and 3.11 W/kg of AgInCd. For a cask fully loaded with fuel assemblies containing design basis CEAs, the additional heat generation due to the CEAs amounts to: [(2.16 W/kg SS or Inconel)(2.495 kg/CEA) + (3.11 W/kg AgInCd)(2.767 kg/CEA)](24 CEA/cask) = 336 W/cask
This value is conservatively rounded up to 350 W. Although longer cool times are considered in this analysis for the fuel source terms, this decay heat generation rate is conservatively used for all longer CEA cool times.

Since the activated portion of the CEA is present in a localized region of the active fuel, and only fuel sources are used in establishing cool times in the one-dimensional dose rate analysis performed here, an adjustment to the one-dimensional dose rate limit is derived based on detailed three-dimensional results obtained for the CE14x14 fuel with and without a CEA present. The adjustment to the dose rate limit is applied only to the normal condition dose rate limit. The accident condition (loss of neutron shield) is limited by fuel neutron sources and results are not significantly affected by the additional source in the end fitting region.

Table 5.5.1.1-1 shows the activation history for CEAs employed at Maine Yankee. Based on this data, individual source term calculations are performed for each CEA group, and a single bounding CEA description is determined based on the maximum computed source rate as of January 1, 2001. The bounding CEA description is based on CEA group "A1-A8," and the resulting CEA spectra at 5, 10 15, and 20 years cool time are shown in Table 5.5.1.1-2.

Although the analysis is conducted in terms of the cool time of the design basis CEA description, the results are equally applicable in terms of CEA source rate. Since the activation spectrum of the CEA is essentially that of ^{60}Co regardless of the cool time, any potential CEA with an exposure history differing from the design basis CEA description can be accommodated by identifying the design basis CEA cool time at which the design basis CEA source rate exceeds the candidate CEA source rate. The loading table for this design basis CEA cool time can then be used for the candidate CEA.

5.5.1.1.1.2 In-Core Instrument (ICI) Thimbles

Activation of ICI thimble material is determined by accumulating the hardware activation incurred during each cycle the ICI is present in the reactor core. The ICIs are first grouped according to exposure history as shown in Table 5.5.1.1-3. The cycle exposure data for each Maine Yankee cycle is shown in Table 5.5.1.1-4. With these data, the accumulated hardware source is obtained by summing the contributions made from each cycle of exposure. It is assumed that:

1. The average cycle exposure is sufficient to represent the ICI thimble exposure during each cycle.
2. Spectral differences between hardware source terms are insignificant.

3. The ICI thimble activated hardware source rate does not decrease after January 1, 2001.
4. The ICI activated hardware spectrum is assumed to be identical to the fuel activated hardware spectrum in distribution, but not total source strength, i.e., the majority of the source is the result of ^{60}Co at a fixed spectrum.

The portion of the ICI present in the active fuel region during reactor operation is composed entirely of Zircaloy and receives no significant activation. The activated components of the ICI are present in the upper end fitting region of the core, and the material is assumed to be irradiated at a flux factor of 0.1 consistent with the activation ratio used for upper end fitting hardware. A total mass of 0.664 kg/ICI of activated material (assumed to be stainless steel with an initial ^{59}Co concentration of 1.2 g/kg) is modeled in the upper end fitting region.

The resulting total source rate as of January 1, 2001, for the activated components of each ICI group are shown in Table 5.5.1.1-3. ICI Group J has the highest source rate (1.4940E+13 γ/sec), and this value is selected as the design basis for the loading table analysis. Note that for the purposes of determining the required cool time for a fuel assembly containing an ICI, no further decay of the ICI is considered after January 1, 2001.

5.5.1.1.1.3 Stainless Steel Replacement Rods

Maine Yankee fuel assemblies containing stainless steel replacement (SSR) rods are listed in Table 5.5.1.1-5. Note that for "N" and "R" numbered fuel assemblies, the SSR rods are only subject to exposure after the first fuel assembly cycle of irradiation. For "U" numbered assemblies, the assemblies saw no additional exposure after the rods were inserted. Hence, these "U" numbered assemblies are not further considered since their SSR rods received no activation.

The SSR rod is assumed to be solid stainless steel with the same dimensions as a fuel rod and with an initial ^{59}Co concentration of 1.2 g/kg. The SSR rod mass is 2.91 kg/SSR. No explicit source term calculation is made for the SSR rods. Instead, the modeled fuel hardware mass along the active fuel region for each assembly is increased by the total mass of rods present.

5.5.1.1.1.4 Consolidated Fuel

There are two consolidated fuel lattices. The lattices house fuel rods taken from assemblies as shown in Table 5.5.1.1-6. Each lattice presents a 17x17 array, with top and bottom end fittings

connected by solid steel connector rods. No explicit source term analysis is conducted for the consolidated fuel lattices themselves, instead, an analysis is presented based on the source term computed for the fuel assemblies from which the contents are derived.

5.5.1.1.2 Model Specification

The one- and three-dimensional models described in Section 5.3 are employed in this analysis. No modifications are required to the models except for the substitution of CE14x14 homogenized source descriptions. These homogenizations are shown in Tables 5.5.1.1-7 through 5.5.1.1-9.

5.5.1.1.3 Shielding Evaluation

The shielding evaluation consists of a loading table analysis of the CE14x14 fuel following the methodology developed in Section 5.4.3. Fuel assemblies which include non-fuel hardware are addressed explicitly. The results of the analysis are loading tables which give the required cool time for a particular fuel configuration.

No restrictions are placed on the loading locations for any of the non-fuel assembly hardware components. This implies that a canister may contain up to 24 CEAs, 24 ICI thimbles, or 24 steel substitute rod assemblies or any combination therefore as long as the most limiting cool time is selected for any of the components in the canister. The only restriction is that no two specialty hardware components may be loaded in the same basket locations. Due to physical constraints, ICI thimbles and CEAs cannot be located in the same assembly. Neither CEA's or ICI thimbles may be placed into an assembly containing steel substitute rods.

5.5.1.1.4 Standard Fuel Source Term

As shown in Section 5.4.3, initial results are obtained for CE14x14 fuel with no additional non-fuel material included. The results are obtained following the loading table analysis methodology developed in Section 5.4.3. CE14x14 source terms at various combinations of initial enrichment and burnup are computed using the CE14x14 SAS2H model described in Section 5.5.1.1.1.

Following the methodology developed in Section 5.4.3, one-dimensional shielding calculations are performed for CE14x14 fuel region sources at various combinations of initial enrichment, burnup, and cool time. The resulting dose rate and source term data is interpolated to determine the cool time required for each combination of enrichment and burnup to decay below the design basis limiting values of dose and heat generation rate.

The resulting loading table for CE14x14 fuel with no additional non-fuel material is shown in Table 5.5.1.1-10. A three-dimensional verification study has been performed for selected fuel combinations obtained from the loading table, Table 5.4-23.

5.5.1.1.4.1 Control Element Assemblies (CEA)

The result of the CEA analysis is a set of loading tables for Maine Yankee fuel giving the cool time required for a fuel assembly with a specified burnup and enrichment combination either 1) to contain a design basis CEA with a cool time of 5, 10, 15,or 20 years or 2) to be present in a Class 2 canister with no CEA inserted. Note that this latter qualification arises because of the geometry difference between Class 1 and Class 2 canisters. Fuel assemblies containing CEAs will be loaded into Class 2 canisters, which are slightly longer than the Class 1 canisters used for bare fuel assemblies. The additional length is required to accommodate the CEA, which is inserted in the top of the fuel assembly.

The approach taken is to compute downward adjustments to the design basis one-dimensional dose rate limiting value (6.71 mrem/hr at 2m from railcar) which ensures that the fuel sources have decayed adequately to cover the effect of the additional source added as a result of CEA containment. The adjustment is determined on the basis of a conservative comparison of three-dimensional shielding analysis results for the original Class 1 canister containing CE14x14 fuel assemblies and the Class 2 canister containing either no CEA or CEAs cooled to 5, 10, 15, or 20 years. Results for CEA cool times longer than 20 years are bounded by the 20 year results.

5.5.1.1.4.1.1 Establishment of Limiting Values

Since the additional activated material in the CEA analysis is assumed to be present in the bottom eight inches of the active fuel region, the one-dimensional dose methodology developed here is not sensitive to the additional source term. The one-dimensional analysis is based on the response from fuel region sources alone. To account for the additional source, the one-

dimensional normal conditions dose rate limit is adjusted by an amount which ensures that the contribution from the additional activated material is bounded.

By adjusting the one-dimensional dose rate limit, we require the fuel to cool longer to a point where the decrease in fuel region dose rate matches the increased dose rate due to the additional CEA material. Hence, it is necessary to determine the amount by which the dose rate increases as a result of the added material. A one-dimensional calculation of this additional dose rate is not reasonable due to the geometry of the CEA source region. One-dimensional buckling corrections are inaccurate for a cylindrical source where the ratio of height to diameter of the source is less than unity, as is the case here.

Instead, the additional contribution to dose rate due to the activated material is computed by a detailed three-dimensional shielding model. The model is based on the three-dimensional models described in Section 5.3. However, the fuel is modeled in a Class 2 canister since that canister will be used to ship CEA-bearing assemblies. The geometric implication of the Class 2 canister is that with a shorter canister spacer employed, the end fitting source region will be located at a lower axial position than it would be for the Class 1 canister. This moves the source region closer to a point of minimal shielding.

The three-dimensional shielding evaluation is conducted for the CE14x14 fuel at the design basis burnup of 45,000 MWD/MTU and initial enrichment of 3.7 wt %. According to the cool time analysis conducted for CE14x14 fuel with no additional non-fuel material in Section 5.5.1.1.4, this fuel will require 9 years cool time before it is acceptable for shipment in the UMS transport cask. Hence, the 9-year cooled CE14x14 at 45,000 MWD/MTU and 3.7 wt % initial enrichment provides the base case for the dose rate limit adjustment calculation. This case is referred to as the Class 1 case below.

Additional three-dimensional models are defined based on the base case fuel configuration in a Class 2 canister and either containing a design basis CEA assumed to be cooled for 5, 10, 15, or 20 years or containing no CEA at all (NoCEA case below). This latter configuration is important because the slight geometry difference between Class 1 and Class 2 canisters (with respect to axial source location) is sufficient to require a modification to the cool time analysis for non-CEA containing assemblies when loaded in the Class 2 canister. Hence, the approach taken here considers the dose rate impact of both 1) the geometrical difference between Class 1 and Class 2 canisters and 2) the additional source due to activated CEA hardware.

5.5.1.1.4.1.2 CEA Source Spectra

For the Class 2 configuration with CEAs, the only difference between the cases is the combined CEA source modeled in the lower eight inches of the active fuel region. The required CEA spectra are shown in Table 5.5.1.1-2.

5.5.1.1.4.1.3 Three-Dimensional Model Results

Table 5.5.1.1-11 gives the three-dimensional UMS Transport bottom model results for each case. Only the bottom model is considered because the top model is not sensitive to changes in the CEA description. The Ratio column shown in the table gives the ratio between the base case maximum 2m+Railcar dose rate and the value computed for each remaining CEA case. This quantity is used to scale the one-dimensional design basis normal conditions dose rate limit of 6.71 mrem/hr in order to determine a modified limiting value applicable to each CEA decay case. The resulting dose rate limits are shown in the "Limit" column of the table.

5.5.1.1.4.1.4 Decay Heat Limits

As discussed in Section 5.5.1.1.1.1, the additional decay heat associated with a full cask of CEAs is conservatively taken as 0.350 kW/cask. This additional heat load is accounted for by reducing the fuel assembly decay heat limits by this amount.

5.5.1.1.4.1.5 Loading Table Analysis

With the adjusted one-dimensional dose and heat generation rate limits established above, the loading table analysis proceeds following the methodology developed in Section 5.4.3. Each combination of initial enrichment and burnup is analyzed to determine the minimum required cool time in order for an assembly to either 1) contain a design basis CEA cooled 5, 10, 15 or 20 years or 2) to be present in a Class 2 canister with no CEA inserted. The resulting cool times are shown in Table 5.5.1.1-12.

Although the analysis is presented in terms of cool time of the design basis CEA description, the results are generally applicable to any CEA assembly intended for shipment stored in CE14x14 fuel. Since spectral differences between CEAs cooled for varying times are insignificant, the key factor of interest is the total gamma source rate emanating from the exposed CEA. That is, any candidate CEA can be considered for shipment based on its total source strength alone by identifying the design basis CEA cool time with source rate most closely bounding the candidate CEA source strength. The loading table associated with this CEA cool time is then used to determine required host assembly cool times. To facilitate this process, Table 5.5.1.1-13 shows the design basis CEA gamma source rates at each analyzed cool time. For a candidate CEA with source rate lower than the 20-year value, the 20-year loading table should be used. If a candidate CEA has a source rate higher than the 5 year value from the table, it should be cooled until its source rate falls below this value. For all other candidate CEA source rates, the cool time associated with the next higher design basis CEA source strength is used to identify the correct loading table.

5.5.1.1.4.2 In-Core Instrument (ICI) Thimbles

The loading table analysis of the in-core instrument thimble follows the same methodology as that developed above for activate CEA hardware. The activated portion of the ICI is present in the upper end fitting region when loaded into a host fuel assembly. Since the source region is outside the fuel region, direct application of the one-dimensional loading table analysis is not possible. Instead, as in the CEA case, the approach is to identify a conservative adjustment to the one-dimensional dose rate limit, thereby forcing the fuel to cool a longer time in order to offset the additional dose from the ICI thimble.

5.5.1.1.4.2.1 Establishment of Limiting Values

Decay heat from the activated ICI thimble is insignificant ($\leq$ 0.05 kW/cask), so no adjustment to the decay heat limit is employed. The one-dimensional normal conditions dose rate limit is adjusted in a manner identical to that employed in the CEA analysis. Two configurations of the CE14x14 Class 1 canister are analyzed in full three-dimensional detail. The first configuration is the base case CE14x14 fuel at 45,000 MWD/MTU, 3.7 wt % enrichment, and 8 years cool time. The second configuration is identical to the base case with the addition of the source term for 24 ICI thimbles to the upper end fitting source region. No credit is taken for the self-shielding effectiveness of the added material. The base case upper end fitting total source strength is

7.6397E+13 γ/sec. The design basis ICI source strength is determined in Section 5.5.1.1.1.2 to be 1.4940E+13 γ/sec. The ICI activated hardware spectrum is assumed to be identical to the fuel activated hardware spectrum.

The results of the two three-dimensional cases are shown in Table 5.5.1.1-14. The ICI thimble source effects only a slight change in the maximum 2 m dose rate. The adjusted one-dimensional normal conditions dose rate limit is 6.63 mrem/hr.

5.5.1.1.4.2.2 Loading Table Analysis

With the one-dimensional dose rate limit established above, the loading table analysis is performed for CE14x14 fuel assemblies at various combinations of enrichment and burnup. Note that unlike the CEA analysis, the ICI thimbles are assumed not to decay after January 1, 2001. The resulting loading table is shown in Table 5.5.1.1-15.

5.5.1.1.4.3 Stainless Steel Replacement Rods

Maine Yankee fuel assemblies containing stainless steel replacement (SSR) rods are listed in Table 5.5.1.1-5. Note that for "N" and "R" numbered fuel assemblies, the SSR rods are only subject to exposure after the first cycle of irradiation of the fuel assembly. For "U" numbered assemblies, the assemblies saw no additional exposure after the rods were inserted. Hence, these "U" numbered assemblies are not further considered since the SSR rods received no activation.

The SSR rod is assumed to be solid stainless steel with the same dimensions as a fuel rod and a mass of 2.91 kg/SSR.

Based on the exposure data provided, SAS2H source calculations are performed explicitly for each SSR-bearing fuel assembly, which received additional exposure. Each fuel assembly is modeled at its initial enrichment (rounded down to the nearest enrichment level equal to a modeled enrichment value) and cycle length parameters are computed to achieve the required burnups as indicated in Table 5.5.1.1-5. The resulting SSR source strengths as of January 1, 2001, are shown in Table 5.5.1.1-16.

A cool time analysis is conducted for each assembly containing irradiated SSR rods. The activated SSR material is treated explicitly by adding the source directly to the fuel hardware

source term. Hence, no adjustment to the one-dimensional dose rate limits is required as in previous analyses involving added non-fuel sources. The results of the cool time analysis for each assembly are shown in Table 5.5.1.1-16.

5.5.1.1.4.4 Consolidated Fuel

There are two consolidated fuel lattices intended for shipment in the Universal Transport Cask. The lattices house fuel rods taken from assemblies as shown in Table 5.5.1.1-6. The consolidated fuel rods have decayed for over twenty years and do not represent a significant shielding issue.

A limiting cool time analysis is conducted by identifying a fuel assembly description analyzed in the loading table analysis which bounds the parameters of the fuel rods in the consolidated fuel lattices. The parameters of those fuel rods are shown in Table 5.5.1.1-17. The CE14x14 fuel at 30,000 MWD/MTU and 1.9 wt % enrichment represents a bounding assembly type since it has a significantly higher burnup and a lower enrichment than the original assemblies. This fuel requires six years cool time before it can be loaded in the transport cask as shown in Table 5.5.1.1-10. The consolidated fuel has been cooled for at least 24 years. For container CN-1 lattice, one can immediately conclude that dose rates are bounded by the limiting fuel.

However, the CN-10 lattice contains significantly more fuel rods than an intact assembly. Neglecting the mitigating effects of additional self-shielding, this situation is addressed by comparing the radiation source strength of the limiting fuel at six and 24 years cool time. Conservatively assuming that all fuel rods present in CN-10 are at the limiting conditions of 30,000 MWD/MTU and 1.9 wt %, the ratio of the source rate in the CN-10 to the source rate in the limiting fuel assembly is shown to be less than one for each source type in Table 5.5.1.1-18. For each source type, the ratio is computed as:

Ratio = (Num Rods in CN-10)(Source Rate at 24 Yr) / (Num Rods in F/A)(Source Rate at 6 Yr)

Hence, CN-10 is also bounded by the limiting case and the consolidated fuel is eligible for shipment in the transport cask as of January 1, 2001.

Figure 5.5.1.1-1 SAS2H Model Input File – CE 14x14

```
=SAS2H          PARM=(HALT03,SKIPSHIPDATA)
CE 14X14 3.7 W/O U235, 45000 MWD/MTU 12.0-22.0 YEAR COOLING
27GROUPNDF4 LATTICECELL
UO2          1 0.950 900 92235 3.7 92238 96.3 END
ZIRCALLOY  2 1.0 620 END
H2O          3 DEN=0.725 1.0 580 END
ARBM-BORMOD 0.725 1 1 0 0 5000 100 3 550.0E-6 580 END
END COMP
SQUAREPITCH 1.4732 0.9563 1 3 1.1176 2 0.9754 0 END
NPIN=176 FUEL=347.98 NCYC=3 NLIB=1 PRIN=6 LIGH=5
INPL=1 NUMH=20 NUMI=0 ORTU=0.5588 SRTU=0.49285 END
POWER=13.065 BURN=463.5350 DOWN=60.0 END
POWER=13.065 BURN=463.5350 DOWN=60.0 END
POWER=13.065 BURN=463.5350 DOWN=1461.00 END
FE 0.672 CR 0.190 NI 0.115 MN 0.020 CO 0.0012
END
```

Table 5.5.1.1-1 CEA Exposure History by Group – Maine Yankee

CEA Group	First Cycle	Last Cycle	Maximum Exposure (MWD/MTU)	Number of Cycles	Exposure Per Cycle (MWD/MTU)	Cool Time as of 1/1/2001 (y)
A1-A8	7	15	60239	9	6693	4
B1-B5	9	15	48909	7	6987	4
C1-C11, C13-C15	10	15	44315	6	7386	4
D1-D15	11	15	35283	5	7057	4
E1-E17, GN, *78, 101, 102, 138-153	12	15	29367	4	7342	4
F1,F2	13	15	18663	3	6221	4
4A	12	12	9786	1	9786	8
C12	10	12	24309	3	8103	8
NA	1	11	75444	11	6859	10
1-69	1	8	53258	8	6657	15

The asterisk is added to CEA 78* to distinguish it from the original CEA 78.

Table 5.5.1.1-2 CE14x14 CEA Hardware Spectra – 5, 10, 15 and 20 Years Cool Time – Maine Yankee

Group	5 year (γ / sec)	10 year (γ / sec)	15 year (γ / sec)	20 year (γ / sec)
1	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
3	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
4	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
5	1.3479E-04	4.4697E-06	1.4822E-07	4.9154E-09
6	7.1467E+06	2.6384E+06	1.3598E+06	7.0431E+05
7	4.0337E+09	1.6979E+09	8.7691E+08	4.5422E+08
8	3.7246E+10	2.3434E+08	1.4804E+06	1.5188E+04
9	1.8642E+14	7.1649E+13	3.6955E+13	1.9142E+13
10	4.8840E+14	2.5265E+14	1.3086E+14	6.7790E+13
11	1.3804E+14	9.4554E+11	4.7779E+10	3.7897E+10
12	1.1469E+15	9.3808E+14	9.1172E+14	8.8714E+14
13	4.3885E+14	4.2316E+14	4.1174E+14	4.0065E+14
14	9.1526E+11	5.5505E+11	5.2913E+11	5.0949E+11
15	1.2039E+12	8.4093E+11	8.0140E+11	7.6939E+11
16	3.8479E+12	2.9855E+12	2.7489E+12	2.5803E+12
17	5.1828E+13	4.4134E+13	4.2118E+13	4.0659E+13
18	3.4899E+14	2.7741E+14	2.6393E+14	2.5520E+14
SS Source Rate	6.3886E+14	3.2951E+14	1.7066E+14	8.8413E+13
AgInCd Source Rate	2.1666E+15	1.6829E+15	1.6308E+15	1.5861E+15
Total Source Rate	2.8055E+15	2.0124E+15	1.8014E+15	1.6745E+15
SFA[1]	5.6110E+15	4.0249E+15	3.6029E+15	3.3490E+15

1. SAS4 file input value.

Table 5.5.1.1-3 Maine Yankee ICI Thimble Exposure History and Total Source Rate by Group

Group	Quantity	Cycles Exposed	Number of Cycles	Total Source [γ/sec]
A	41	1, 1A, 2	3	9.1881E+11
B	1	1	1	2.3775E+11
C	2	1, 1A	2	3.6244E+11
D	1	1A, 2	2	6.8106E+11
E	3	2	1	5.5637E+11
F	15	3 thru 11, 13	10	1.1695E+13
G	12	3 thru 11, 14	10	1.2126E+13
H	12	3 thru 11, 15	10	1.1454E+13
I	3	3 thru 9,14,15	9	1.1309E+13
J	2	10 thru 15	6	1.4940E+13
K	1	10 thru 12	3	6.1296E+12
L	25	12 thru 15	4	1.1491E+13
M	17	12	1	2.6801E+12
N	3	13 thru 15	3	8.8105E+12

Table 5.5.1.1-4 Maine Yankee Core Exposure History by Cycle of Operation

Cycle	Discharge Date	Cycle Burnup [MWD/MTU]	Core Average Enrichment [wt %]
1	6/29/74	10367	2.44
1A	5/2/75	4492	2.30
2	4/9/77	17365	2.45
3	7/14/78	11105	2.59
4	1/11/80	10500	2.84
5	5/8/81	10799	2.98
6	9/24/82	11585	3.01
7	3/31/84	12483	3.10
8	8/17/85	12504	3.20
9	3/28/87	14424	3.29
10	10/15/88	12675	3.36
11	4/7/90	13786	3.50
12	2/14/92	15364	3.62
13	7/30/93	13668	3.68
14	1/14/95	13075	3.75
15	12/6/96	7859	3.76

Table 5.5.1.1-5 Maine Yankee Fuel Assemblies with Stainless Steel Replacement Rods (SSR) Showing Cycles of Operation and Burnup Received

Assembly Number	1st Cycle	2nd Cycle	3rd Cycle	1st Cycle Burnup[1]	2nd Cycle Burnup[1]	3rd Cycle Burnup[1]	Number SSR
N420	9	10	11	16,428	13,467	11,893	3
N842	9	10	-	18,420	13,885	0	1
N868	9	10	11	18,622	13,386	4,919	1
R032	12	13	14	16,464	15,386	12,168	1
R439	12	13	14	20,371	14,779	11,685	1
R444	12	13	14	20,371	14,779	11,685	4
U01	15	-	-	7,339	0	0	1
U05	15	-	-	7,339	0	0	1
U16	15	-	-	10,598	0	0	1
U37	15	-	-	9,005	0	0	1
U51	15	-	-	8,288	0	0	1
U60	15	-	-	8,288	0	0	5

1. MWD/MTU

Table 5.5.1.1-6 Contents of Maine Yankee Consolidated Fuel Lattices CN-1 and CN-10

C.F. Lattice	Original Fuel Assembly	Number Rods	Actual Burnup [MWD/MTU]	Initial Enrichment [wt %]
CN-1	EF0039	172	5150	1.929
CN-10	EF0045	176	17150	1.953
	EF0046	107	17150	1.953

Table 5.5.1.1-7 Maine Yankee CE14x14 Homogenized Fuel Region Isotopic Composition

Isotope	CE14x14 [atom/b-cm]
ALUMINUM	2.05114E-03
BORON-10	1.90898E-04
BORON-11	7.68387E-04
CARBON-12	2.39821E-04
CHROMIUM(SS304)	7.19369E-04
IRON(SS304)	2.4501E-03
MANGANESE	7.16674E-05
NICKEL(SS304)	3.18674E-04
OXYGEN-16	8.72597E-03
URANIUM-234	2.39964E-07
URANIUM-235	3.14135E-05
URANIUM-238	4.33133E-03
ZIRCALLOY	3.06324E-03

Table 5.5.1.1-8 Isotopic Compositions of Maine Yankee CE14x14 Fuel Assembly Non-Fuel Source Regions

Isotope	Upper Plenum [atom/b-cm]	Upper End Fit [atom/b-cm]	Lower End Fit [atom/b-cm]
CHROMIUM(SS304)	1.59190E-03	1.89910E-03	3.08125E-03
MANGANESE	1.58594E-04	1.89199E-04	3.06971E-04
IRON(SS304)	5.42166E-03	6.46791E-03	1.04941E-02
NICKEL(SS304)	7.05196E-04	8.41284E-04	1.36497E-03
ZIRCALLOY	3.22036E-03	-	-

Table 5.5.1.1-9 Isotopic Compositions of Maine Yankee CE14x14 Canister Annular Region Materials (One-Dimensional Analysis Only)

Isotope	Fuel Annulus [atom/b-cm]	Upper Plenum Annulus [atom/b-cm]	Upper End Fit Annulus [atom/b-cm]	Lower End Fit Annulus [atom/b-cm]
ALUMINUM	5.96817E-03	-	-	-
CHROMIUM(SS304)	1.77895E-03	9.31065E-04	2.53529E-03	4.13797E-03
MANGANESE	1.77228E-04	9.27577E-05	2.52579E-04	4.12247E-04
IRON(SS304)	6.05870E-03	3.1710E-03	8.63463E-03	1.40930E-02
NICKEL(SS304)	7.88057E-04	4.12453E-04	1.12311E-03	1.83308E-03

Table 5.5.1.1-10 Loading Table for Maine Yankee CE14x14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable

Loading Table for CE14x14 Fuel with No Non-Standard Fuel Material					
Enrichment	**Burnup (B) [GWD/MTU]**				
[wt %]	**$B \le 30$**	**$30 \le B < 35$**	**$35 \le B < 40$**	**$40 \le B < 45$**	**$45 \le B < 50$**
$1.9 \le E < 2.1$	6 years	8 years	11 years	18 years	27 years
$2.1 \le E < 2.3$	6 years	7 years	10 years	15 years	24 years
$2.3 \le E < 2.5$	6 years	7 years	9 years	14 years	22 years
$2.5 \le E < 2.7$	6 years	7 years	9 years	12 years	19 years
$2.7 \le E < 2.9$	6 years	6 years	8 years	11 years	17 years
$2.9 \le E < 3.1$	5 years	6 years	8 years	10 years	15 years
$3.1 \le E < 3.3$	5 years	6 years	7 years	10 years	15 years
$3.3 \le E < 3.5$	5 years	6 years	7 years	9 years	15 years
$3.5 \le E < 3.7$	5 years	6 years	7 years	9 years	14 years
$3.7 \le E \le 4.2$	5 years	6 years	7 years	9 years	14 years

Table 5.5.1.1-11 Three-Dimensional Shielding Analysis Results for Various Maine Yankee CEA Configurations Establishing One-Dimensional Dose Rate Limits for Loading Table Analysis

CEA Cool Time [y]	Dose Rate [mrem/hr]	FSD	Ratio [%]	Limit [mrem/hr]
Class 1 Result	7.70	0.72%		6.71
NoCea	7.92	0.63%	97.2%	6.53
5y	9.41	0.55%	81.9%	5.49
10y	8.53	0.59%	90.3%	6.06
15y	8.22	0.60%	93.6%	6.28
20y	8.08	0.61%	95.3%	6.39

Table 5.5.1.1-12 Loading Table for Maine Yankee CE14x14 Fuel Containing CEA Cooled to Indicated Time

Loading Table for ce14x14 Fuel - Minimum Required Cool Time in Years						
Burnup	30 GWD/MTU	Minimum Cool Time [y] for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr CEA	20 Yr CEA
1.9	6	6	7	6	6	6
2.1	6	6	7	6	6	6
2.3	6	6	6	6	6	6
2.5	6	6	6	6	6	6
2.7	6	6	6	6	6	6
2.9	5	6	6	6	6	6
3.1	5	5	6	6	6	5
3.3	5	5	6	6	5	5
3.5	5	5	6	5	5	5
3.7	5	5	6	5	5	5
Burnup	35 GWD/MTU	Minimum Cool Time [y] for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr CEA	20 Yr CEA
1.9	8	8	9	8	8	8
2.1	7	7	9	8	8	8
2.3	7	7	8	7	7	7
2.5	7	7	8	7	7	7
2.7	6	7	7	7	7	7
2.9	6	6	7	7	6	6
3.1	6	6	7	6	6	6
3.3	6	6	7	6	6	6
3.5	6	6	6	6	6	6
3.7	6	6	6	6	6	6
Burnup	40 GWD/MTU	Minimum Cool Time [y] for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr CEA	20 Yr CEA
1.9	11	12	14	13	12	12
2.1	10	10	13	11	11	11
2.3	9	9	12	10	10	10
2.5	9	9	10	9	9	9
2.7	8	8	10	9	8	8
2.9	8	8	9	8	8	8
3.1	7	7	8	8	8	8
3.3	7	7	8	7	7	7
3.5	7	7	8	7	7	7
3.7	7	7	7	7	7	7
Burnup	45 GWD/MTU	Minimum Cool Time [y] for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr CEA	20 Yr CEA
1.9	18	18	21	19	18	18
2.1	15	16	19	17	17	16
2.3	14	14	18	16	15	15
2.5	12	13	16	14	14	13
2.7	11	12	14	13	12	12
2.9	10	11	13	12	11	11
3.1	10	10	12	11	10	10
3.3	9	9	11	10	10	10
3.5	9	9	10	10	10	10
3.7	9	9	10	10	10	10
Burnup	50 GWD/MTU	Minimum Cool Time [y] for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr CEA	20 Yr CEA
1.9	27	27	29	27	27	27
2.1	24	24	27	25	24	24
2.3	22	22	25	23	22	22
2.5	19	19	23	21	20	20
2.7	17	17	21	19	18	18
2.9	15	16	19	18	18	18
3.1	15	15	18	17	17	17
3.3	15	15	17	17	17	17
3.5	14	14	15	15	15	15
3.7	14	14	15	15	15	15

Table 5.5.1.1-13 Design Basis Maine Yankee CEA Source Rate at Each Cool Time Analyzed

CEA Cool Time [y]	Source Strength [γ/sec/CEA]
5	1.1690E+14
10	8.3851E+13
15	7.5060E+13
20	6.9771E+13

Table 5.5.1.1-14 Establishment of Dose Rate Limit for Maine Yankee ICI Thimble Analysis

	Bottom		Top	
Case	Rate (mrem/hr)	FSD	Rate (mrem/hr)	FSD
No ICI	8.50	0.7%	9.78	0.8%
4 Yr Cooled ICI	8.50	0.7%	9.87	0.8%
Delta			-0.08	
Original Limit			6.71	
Adjusted Limit			6.63	

Table 5.5.1.1-15 Loading Table for Maine Yankee CE14x14 Fuel Containing ICI Thimble

Enrichment	Burnup [GWD/MTU]				
[wt %]	30	35	40	45	50
1.9	6 years	8 years	11 years	18 years	27 years
2.1	6 years	7 years	10 years	16 years	24 years
2.3	6 years	7 years	9 years	14 years	22 years
2.5	6 years	7 years	9 years	13 years	19 years
2.7	6 years	6 years	8 years	11 years	17 years
2.9	5 years	6 years	8 years	10 years	15 years
3.1	5 years	6 years	7 years	10 years	15 years
3.3	5 years	6 years	7 years	9 years	15 years
3.5	5 years	6 years	7 years	9 years	14 years
3.7	5 years	6 years	7 years	9 years	14 years

Table 5.5.1.1-16 Required Cool Time for Maine Yankee Fuel Assemblies with Activated Stainless Steel Replacement Rods

Assy Number	Burnup [GWD/MTU]	Enrichment [wt %]	SSR Source [g/s/assy]	Cool Time [y]	Earliest Loadable
N420	45	3.3	2.1602E+13	10	Jan 2001
N842	35	3.3	3.1396E+12	6	Jan 2001
N868	40	3.3	5.2444E+12	7	Jan 2001
R032	45	3.5	1.4550E+13	9	Jan 2005
R439	50	3.5	1.3998E+13	14	Jan 2010
R444	50	3.5	5.5993E+13	19	Jan 2015

Table 5.5.1.1-17 Maine Yankee Consolidated Fuel Model Parameters

			Actual		Modeled		Required	Cool Time
Lattice	Assy	Num Rods	Burnup [MWD/MTU]	Enrichment [wt %]	Burnup [MWD/MTU]	Enrichment [wt %]	Cool Time [y]	1/1/01 [y]
CN-1	EF0039	172	5150	1.929	30000	1.9	6	26
CN-10	EF0045	176	17150	1.953	30000	1.9	6	24
	EF0046	107	17150	1.953	30000	1.9	6	24

Table 5.5.1.1-18 Maine Yankee Source Rate Analysis for CN-10 Consolidated Fuel Lattice

Cool Time [y]	Num Rods Present	Decay Heat [kW/cask]	Fuel Neutron [n/s/assy]	Fuel Gamma [g/sec/assy]	Fuel Hardware [g/sec/assy]
6	176	13.9	1.63E+08	3.16E+15	9.28E+12
24	283	7.42	8.41E+07	1.28E+15	8.67E+11
Src Ratio 24/6		0.86	0.83	0.65	0.15

5.5.1.2 Maine Yankee Site Specific GTCC Waste

Source terms, decay heat, and dose rates are calculated for the Maine Yankee Greater Than Class C (GTCC) waste to be transported in the Universal Transport Cask. The calculations are performed by first determining the gamma source spectra and decay heat from the GTCC isotopics, then using the gamma source information to calculate radial and axial dose rates. The results of these calculations show that the dose rates produced by the GTCC waste are bounded by the dose rates previously calculated for the design basis spent fuel, as presented in Section 5.1.3. Detailed results for the Maine Yankee GTCC waste are presented in Section 5.5.1.2.4.

5.5.1.2.1 GTCC Waste Transport Configuration

The GTCC waste basket is described in Section 1.3.1.1.2. The GTCC waste material is loaded into a cylindrical shell that is 3 inches thick. The cavity is divided into two loading sections using an insert that is placed into the GTCC basket after loading the bottom section. Each of the two sections may contain up to 10,000 pounds of waste, for a total of 20,000 pounds per canister.

5.5.1.2.2 GTCC Waste Source Term

The radionuclide inventories presented in Table 5.5.1.2-1 are utilized to develop the bounding source spectra and decay heat after 5 years cooling from March 1, 1998. These bounding radionuclide inventories are developed from information provided by the utility of all significant isotopes in the activated metal to be transported. The decay heat resulting from this source term is 4,490 watts.

The gamma source spectra presented in Table 5.5.1.2-2 is produced from the radionuclide inventory presented in Table 5.5.1.2-1. This information is used to derive the total gamma source strength which is utilized along with the homogenized source volume to calculate the uniformly distributed source strength per unit volume, as shown in Table 5.5.1.2-3.

The source volume information presented in Table 5.5.1.2-3 is based on the cylindrical region of the basket and the 77-inch (195.58 cm) length of each of the two GTCC canister loading cavities. These dimensions are used to determine the volumetric source strength and homogenized material densities that are used as input in the SAS1 sequence of the SCALE 4.3 computer code package to calculate external dose rates.

5.5.1.2.3 GTCC Waste Shielding Models

The radial and top axial models for the transport cask are described in the following sections. The relevant dimensions of the GTCC basket/canister within the transport cask are taken from the drawings presented in Section 1.3.4.

5.5.1.2.3.1 Radial Models

The 1-D radial models consist of a series of infinitely long concentric cylinders representing a cross section taken at a point along the length of the GTCC source cavity region of the GTCC basket/canister. The nested cylindrical regions consist of the following regions and cumulative thicknesses:

1.	Homogenized waste source region	61.900 cm
2.	GTCC basket stainless steel shield	7.620 cm
3.	Basket support disks	13.690 cm
4.	Canister stainless steel shell	1.588 cm
5.	Gap	0.698 cm
6.	Transport cask stainless steel inner shell	5.080 cm
7.	Transport cask lead gamma shielding	6.871 cm
8.	Transport cask lead gap	0.114 cm
9.	Transport cask stainless steel outer shell	6.985 cm
10.	Transport cask NS-4-FR neutron shielding	11.430 cm
11.	Transport cask stainless steel neutron shield shell	0.635 cm

The SAS1 input is generated using these regions, and specifying a buckling height of 391.16 cm, which is equal to the two 77-inch (195.58 cm) loading cavity heights within the GTCC basket.

5.5.1.2.3.2 Top Axial Models

The 1-D SAS1 top axial models consist of a series of infinite slabs representing the GTCC waste canister and cask materials. Both the normal and accident conditions models are symmetric about the bottom of the top source region, assuming that the bottom region source does not penetrate the 3-inch separator plate and top source region sufficiently to contribute to the dose rate at the top of the cask. The slabs modeled for the top axial normal condition models are:

1.	Homogenized waste source region	195.580 cm
2.	Basket stainless steel top lid	3.810 cm
3.	Canister stainless steel shield lid	17.780 cm
4.	Canister stainless steel structural lid	7.620 cm
5.	Transport Cask stainless steel lid	16.510 cm
6.	Impact Limiter stainless steel inner shell	0.635 cm
7.	Impact Limiter redwood	75.565 cm
8.	Impact Limiter balsa wood	3.810 cm
9.	Impact Limiter stainless steel outer shell	0.635 cm

The SAS1 input is generated using these regions, and specifying a buckling diameter of 170.4 cm, which is equal to the diameter of the canister. This buckling is conservative, as the value is larger than the actual source diameter of approximately 124 cm.

5.5.1.2.3.3 Bottom Axial Models

Like the 1-D top axial models, the 1-D bottom axial models represent the canister and cask materials as a series of infinite slabs. Both the normal and accident conditions models are symmetric about the top of the bottom source region, assuming that the top region source does not penetrate the 3-inch separator plate and bottom source region sufficiently to contribute to the dose rate at the bottom of the cask. The slabs modeled for the bottom axial normal condition models are:

1.	Homogenized waste source region	195.580 cm
2.	Basket stainless steel bottom plate	7.620 cm
3.	Canister stainless steel bottom	4.445 cm
4.	Transport cask spacer (modeled as void)	42.545 cm
5.	Transport cask stainless steel inner bottom plate	10.795 cm
6.	Transport cask NS-4-FR neutron shielding	2.540 cm
7.	Transport cask stainless steel outer bottom plate	12.700 cm
8.	Impact Limiter stainless steel inner shell	0.635 cm
9.	Impact Limiter redwood	75.565 cm
10.	Impact Limiter balsa wood	3.810 cm
11.	Impact Limiter stainless steel outer shell	0.635 cm

The SAS1 input is generated using these regions, and specifying a buckling diameter of 170.4 cm, which is equal to the diameter of the canister. This buckling is conservative, as the value is larger than the actual source diameter of approximately 124 cm.

5.5.1.2.4 GTCC Waste Dose Rate Results

The models described in the previous sections were analyzed using the SAS1 1-D code. The resulting dose rates were multiplied by a 20% peaking factor to provide additional conservatism to the results. This conservative factor accounts for potential non-uniformity in the radionuclide distribution within the GTCC waste material, and for potential shifting of the waste material during transport.

The Transport Cask containing Maine Yankee GTCC waste produces external dose rates below those calculated for the design basis PWR fuel loading. This ensures that the GTCC waste will meet all 10CFR71 normal and accident shielding requirements. During normal operation 10CFR71.47 requires the package surface dose rate be less than 200 mrem/hr and the dose rate two meters from the transportation vehicle not exceed 10 mrem/hr, including the conservative peaking factor.

As presented in Table 5.5.1.2-4, the highest calculated normal condition package surface dose rate is 33 mrem/hr, which occurs on the radial surface. Likewise, the highest package dose rate at 2 meters from the surface is 9.24 mrem/hr.

During accident conditions, 10CFR71.51 limits the dose rate to 1000 mrem/hr at 1 meter from the package surface. The maximum dose rate at 1 meter from the radial surface, as shown in Table 5.5.1.2-5, considering the combined accident of a loss of neutron shielding and lead slump, is 204 mrem/hr.

Table 5.5.1.2-1 Design Basis GTCC Source Term

Radionuclide	Curie Inventory
H-3	3.00E+02 Ci
C-14	1.50E+02 Ci
MN-54	3.50E+02 Ci
FE-55	2.00E+05 Ci
CO-58	1.00E+01 Ci
CO-60	2.90E+05 Ci
NI-59	8.20E+02 Ci
NI-63	9.00E+04 Ci
NB-94	1.00E+01 Ci
TC-99	1.00E+01 Ci
Total	5.82E+05 Ci

Table 5.5.1.2-2 Design Basis GTCC Gamma Source Spectra

Energy Range			gamma/sec	MeV/sec	Normalized gamma/sec
8.00E+00	to	1.00E+01	0.00E+00	0.00E+00	0.00E+00
6.50E+00	to	8.00E+00	0.00E+00	0.00E+00	0.00E+00
5.00E+00	to	6.50E+00	0.00E+00	0.00E+00	0.00E+00
4.00E+00	to	5.00E+00	0.00E+00	0.00E+00	0.00E+00
3.00E+00	to	4.00E+00	0.00E+00	0.00E+00	0.00E+00
2.50E+00	to	3.00E+00	1.76E+08	4.84E+08	7.96E-09
2.00E+00	to	2.50E+00	1.13E+11	2.55E+11	5.14E-06
1.66E+00	to	2.00E+00	1.75E+09	3.20E+09	7.92E-08
1.33E+00	to	1.66E+00	4.78E+15	7.15E+15	2.16E-01
1.00E+00	to	1.33E+00	1.69E+16	1.97E+16	7.66E-01
8.00E-01	to	1.00E+00	1.35E+13	1.21E+13	6.09E-04
6.00E-01	to	8.00E-01	3.92E+11	2.74E+11	1.77E-05
4.00E-01	to	6.00E-01	1.71E+11	8.57E+10	7.76E-06
3.00E-01	to	4.00E-01	9.12E+11	3.19E+11	4.13E-05
2.00E-01	to	3.00E-01	6.96E+11	1.74E+11	3.15E-05
1.00E-01	to	2.00E-01	1.40E+13	2.10E+12	6.33E-04
5.00E-02	to	1.00E-01	5.80E+13	4.35E+12	2.63E-03
1.00E-02	to	5.00E-02	2.95E+14	8.84E+12	1.33E-02
TOTALS			2.21E+16	2.69E+16	1.00E+00

Table 5.5.1.2-3 Design Basis GTCC Source Strength

Parameter	Value
Total Source Strength	2.21E+16 gamma/sec
Equivalent Source Radius	61.9 cm
Equivalent Source Length	391.16 cm
Equivalent Source Volume	4.709E+06 cc
Weight Volume Fraction	2.438E-01
Volumetric Source	4.69E+09 gamma/sec/cc

Table 5.5.1.2-4 GTCC Waste Dose Rate Results – Normal Conditions of Transport

Location	Surface	2 Meters from Surface
Radial	32.66 mr/hr	9.24 mr/hr
Top Axial	0.21 mr/hr	0.06 mr/hr
Bottom Axial	3.11 mr/hr	0.91 mr/hr

Table 5.5.1.2-5 GTCC Waste Dose Rate Results – Accident Conditions

Location	1 Meter from Surface
Radial	203.94 mr/hr
Top Axial	1.18 mr/hr
Bottom Axial	23.51 mr/hr

5.5.2 Dose Response Factors

Neutron and photon dose response factors are shown in Table 5.5.2-1 and Table 5.5.2-2, respectively. These values are obtained from the standard SCALE values, which are based on ANSI/ANS 6.1-1977 values.

Table 5.5.2-1 Neutron Dose Response Factors

Group	Elow [MeV]	Ehi [MeV]	Response [(Rem/hr)/(/cm^2/sec)]
1	6.43E+00	2.00E+01	1.4916E-04
2	3.00E+00	6.43E+00	1.4464E-04
3	1.85E+00	3.00E+00	1.2701E-04
4	1.40E+00	1.85E+00	1.2811E-04
5	9.00E-01	1.40E+00	1.2977E-04
6	4.00E-01	9.00E-01	1.0281E-04
7	1.00E-01	4.00E-01	5.1183E-05
8	1.70E-02	1.00E-01	1.2319E-05
9	3.00E-03	1.70E-02	3.8365E-06
10	5.50E-04	3.00E-03	3.7247E-06
11	1.00E-04	5.50E-04	4.0150E-06
12	3.00E-05	1.00E-04	4.2926E-06
13	1.00E-05	3.00E-05	4.4744E-06
14	3.05E-06	1.00E-05	4.5676E-06
15	1.77E-06	3.05E-06	4.5581E-06
16	1.30E-06	1.77E-06	4.5185E-06
17	1.13E-06	1.30E-06	4.4879E-06
18	1.00E-06	1.13E-06	4.4665E-06
19	8.00E-07	1.00E-06	4.4345E-06
20	4.00E-07	8.00E-07	4.3271E-06
21	3.25E-07	4.00E-07	4.1975E-06
22	2.25E-07	3.25E-07	4.0976E-06
23	1.00E-07	2.25E-07	3.8390E-06
24	5.00E-08	1.00E-07	3.6748E-06
25	3.00E-08	5.00E-08	3.6748E-06
26	1.00E-08	3.00E-08	3.6748E-06
27	1.00E-11	1.00E-08	3.6748E-06

Table 5.5.2-2 Photon Dose Response Factors

Group	Elow [MeV]	Ehi [MeV]	Response [(Rem/hr)/(/cm^2/sec)]
1	8.00E+00	1.00E+01	8.7716E-06
2	6.50E+00	8.00E+00	7.4785E-06
3	5.00E+00	6.50E+00	6.3748E-06
4	4.00E+00	5.00E+00	5.4136E-06
5	3.00E+00	4.00E+00	4.6221E-06
6	2.50E+00	3.00E+00	3.9596E-06
7	2.00E+00	2.50E+00	3.4686E-06
8	1.66E+00	2.00E+00	3.0192E-06
9	1.33E+00	1.66E+00	2.6276E-06
10	1.00E+00	1.33E+00	2.2051E-06
11	8.00E-01	1.00E+00	1.8326E-06
12	6.00E-01	8.00E-01	1.5228E-06
13	4.00E-01	6.00E-01	1.1725E-06
14	3.00E-01	4.00E-01	8.7594E-07
15	2.00E-01	3.00E-01	6.3061E-07
16	1.00E-01	2.00E-01	3.8338E-07
17	5.00E-02	1.00E-01	2.6693E-07
18	1.00E-02	5.00E-02	9.3472E-07

THIS PAGE INTENTIONALLY LEFT BLANK

5.5.3 Sample Input Files

5.5.3.1 SAS2H Input

5.5.3.1.1 PWR Design Basis

```
=SAS2H      PARM=(HALT03,SKIPSHIPDATA)
WEST 17X17 3.7 W/O U235, 45000 MWD/MTU 5-10 YEAR COOLING
27GROUPNDF4 LATTICECELL
UO2       1 0.95 900 92235 3.7 92238 96.3 END
ZIRCALLOY  2 1.0 620 END
H2O        3 DEN=0.725 1.0 580 END
ARBM-BORMOD 0.725 1 1 0 0 5000 100 3 550.0E-6 580 END
END COMP
SQUAREPITCH 1.2598 0.8192 1 3 0.9500 2 0.8357 0 END
NPIN=264 FUEL=365.76 NCYC=3 NLIB=1 PRIN=6 LIGH=5
INPL=1 NUMH=24 NUMI=1 ORTU=0.6122 SRTU=0.5715 END
POWER=19.36 BURN=361.92 DOWN=60.0 END
POWER=19.36 BURN=361.92 DOWN=60.0 END
POWER=19.36 BURN=361.92 DOWN=1461.00 END
FE 0.672 CR 0.190 NI 0.115 MN 0.020 CO 0.0012
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING TO 10 YEARS AND FISSION PRODUCT GAMMA REBIN
3$$  21 0 1 A33 -86  E
54$$  A8 1 E T
35$$  0 T
56$$  0 6 A13 -2 5 3 E
57**  4.0 E T
COOLING TO 10 YEARS AND FISSION PRODUCT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60**  5.0 6.0 7.0 8.0 9.0 10.0
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F.01
81$$  2 51 26 1 E
82$$  F6 T
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
56$$ F0 T
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING TO 10 YEARS AND ACTINIDE GAMMA REBIN
3$$  21 0 1 A33 -86  E
54$$  A8 1 E T
35$$  0 T
56$$  0 6 A13 -2 5 3 E
57**  4.0 E T
COOLING TO 10 YEARS AND ACTINIDE GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60**  5.0 6.0 7.0 8.0 9.0 10.0
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F.01
81$$  2 51 26 1 E
82$$  F5 T
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
```

5.5.3.1.1 PWR Design Basis (Continued)

```
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
56$$ F0 T
END
=ORIGENS
0$$ A4 21 A8 26 A10 51 71 E
1$$ 1 1T
COOLING TO 10 YEARS AND LIGHT ELEMENT GAMMA REBIN
3$$ 21 0 1 A33 -86 E
54$$ A8 1 E T
35$$ 0 T
56$$ 0 6 A13 -2 5 3 E
57** 4.0 E T
COOLING TO 10 YEARS AND LIGHT ELEMENT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0
65$$ A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61** F.01
81$$ 2 51 26 1 E
82$$ F4 T
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
56$$ F0 T
END
```

5.5.3.1.2 BWR Design Basis

```
=SAS2H     PARM=(HALT03,SKIPSHIPDATA)
GE/4-6 9x9-7 3.25 W/O U235 40,000 MWD/MTU, 40% VOID, 5-15 YEARS COOLING
27GROUPNDF4 LATTICECELL
UO2      1 0.95 840 92235 3.25 92238 96.75 END
ZIRCALLOY 2 1.0 620. END
H2O 3 DEN=0.446 1.0 562. END
H2O 4 DEN=0.743 1.0 553. END
ZIRCALLOY 5 1.0 553 END
H2O 6 DEN=0.446 1.0 562. END
END COMP
SQUAREPITCH 1.44 0.955 1 3 1.120 2 0.978 0 END
NPIN/ASSM=74 FUELENGTH=381.00 NCYCLES=3 NLIB/CYC=1 PRINTLEVEL=6
LIGHTEL=5 INPLEVEL=2 NUMZONES=7 END
 4 1.669 5 1.768 6 2.150 500 7.313 6 7.564 5 7.793 4 8.598
POWER=4.95 BURN=499.40 DOWN=60 END
POWER=4.95 BURN=499.40 DOWN=60 END
POWER=4.95 BURN=499.40 DOWN=1461 END
FE 0.672 CR 0.190 NI 0.115 MN 0.020 CO 0.0012
END
=ORIGENS
0$$ A4 21 A8 26 A10 51 71 E
1$$ 1 1T
COOLING TO 10 YEARS AND FISSION PRODUCT GAMMA REBIN
3$$ 21 0 1 A33 -86 E
54$$ A8 1 E T
35$$ 0 T
56$$ 0 6 A13 -2 5 3 E
57** 4.0 E T
COOLING TO 10 YEARS AND FISSION PRODUCT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0
65$$ A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61** F.01
81$$ 2 51 26 1 E
82$$ F6 T
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
```

5.5.3.1.2 BWR Design Basis (Continued)

```
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
56$$ F0 T
END
=ORIGENS
0$$ A4 21 A8 26 A10 51 71 E
1$$ 1 1T
COOLING TO 10 YEARS AND ACTINIDE GAMMA REBIN
3$$ 21 0 1 A33 -86 E
54$$ A8 1 E T
35$$ 0 T
56$$ 0 6 A13 -2 5 3 E
57** 4.0 E T
COOLING TO 10 YEARS AND ACTINIDE GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0
65$$ A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61** F.01
81$$ 2 51 26 1 E
82$$ F5 T
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
56$$ F0 T
END
=ORIGENS
0$$ A4 21 A8 26 A10 51 71 E
1$$ 1 1T
COOLING TO 10 YEARS AND LIGHT ELEMENT GAMMA REBIN
3$$ 21 0 1 A33 -86 E
54$$ A8 1 E T
35$$ 0 T
56$$ 0 6 A13 -2 5 3 E
57** 4.0 E T
COOLING TO 10 YEARS AND LIGHT ELEMENT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0
65$$ A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61** F.01
81$$ 2 51 26 1 E
82$$ F4 T
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
LIGHT ELEMENT SCALE GROUP STRUCTURE
56$$ F0 T
END
```

5.5.3.2 Sample Shielding Model Files

Sample input files are provided for the modified version of SAS4 employed in this analysis. The following modifications are required for the files to work with SAS4:

- Change "SAS4A" to "SAS4" in the first line of each file.
- Remove the "ICS=" parameter from the problem parameter card.
- Remove the CSF, CSL, and CSD cards.
- Provide new surface and/or point detector descriptions in accordance with SAS4 input requirements.

5.5.3.2.1 PWR Top Radial Fuel Neutron – Normal Conditions

```
=SAS4A       PARM='SIZE=1000000'
UMS Transport-we17x17_10Y_45B-Top-Normal-Dry-Radial-Fuel-Neutron-UMS Profile
27N-18COUPLE  INFHOMMEDIUM
' WE17x17 Material File
' $Id: we17x17.matl,v 1.4 1997/07/22 22:55:52 csh Exp $
'
' Reference EA790-4009 Rev. 0 Spreadsheet pwr-mat-h2o.xls
'
' Mat'l 1 is fuel, 2 is basket outside of fuel radius, 3 is TSC
'
UO2          1  DEN=2.1530 1.0 END
ZIRCALLOY    1  DEN=0.4494 1.0 END
SS304        1  DEN=0.3807 1.0 END
AL           1  DEN=0.0894 1.0 END
B4C          1  DEN=0.0220 1.0 END
SS304        2  DEN=0.7691 1.0 END
AL           2  DEN=0.2544 1.0 END
'
' Begin Upper Plenum
' 4 is in fuel radius, 5 is outside of fuel radius
'
ZIRCALLOY    3  DEN=0.4494 1.0 END
SS304        3  DEN=1.0318 1.0 END
SS304        4  DEN=0.6101 1.0 END
'
'BEGIN U. EF
'
SS304        5  DEN=1.2537 1.0 END
SS304        6  DEN=1.1366 1.0 END
'
'BEGIN LOWER END FITTING
'
SS304        9  DEN=1.4554 1.0 END
SS304       10  DEN=1.7805 1.0 END
' MASTER MATERIALS LIST
' $Id: master.matl,v 1.3 1997/07/09 19:56:48 csh Exp $
'
CARBONSTEEL     11  1.0        END
SS304           12  1.0        END
PB              13  1.0        END
B-10            14  0.0  8.553-5 END
B-11            14  0.0  3.422-4 END
AL              14  0.0  7.763-3 END
H               14  0.0  5.854-2 END
O               14  0.0  2.609-2 END
C               14  0.0  2.264-2 END
N               14  0.0  1.394-3 END
```

5.5.3.2.1 PWR Top Radial Fuel Neutron – Normal Conditions (Continued)

```
BALSA           15  1.0      END
REDWOOD         16  1.0      END
AL              17  1.0      END
REG-CONCRETE    18  den=2.2426  1.0   END
N               19  1.E-6   END
CU              20  0.4286  END
SS304           20  0.5714  END
END COMP
IDR=0 ITY=1 IZM=10 ISN=8 FRD=71.695  END
' WE17x17 Top Radial Biasing Rev. 2.8 - Normal Conditions
' IZM=10
   71.695  83.579  85.166  85.865  90.945  97.815
   97.930 104.915 116.345 116.980 END
   1 2 12  0 12 13  0 12 14 12 END
XEND
TIM=3000.0 NST=2000 NIT=250 ISO=0 IGO=4 RAN=CAFE001 SFA=7.5140e+09
ICS=9 NOD=0 IPF=8 END
SOE
' W17_45B 10 year Neutron Spectrum
5.1380E+06 5.8410E+07 6.4710E+07 3.6450E+07 4.9320E+07 5.3780E+07
1.0530E+07 20R0.0 END
SXY 1 -83.579 83.579 -83.579 83.579   0.000 182.880 83.579
    182.880  1.0 1.0 END
' WE17x17 Top Model Radial Detector Desc. Rev. 2.8 - Normal Conditions
' ICS=9
CSF 108.280 116.980 116.980 135.763 157.480 216.980
     316.980 357.480 516.980 END
CSL 216.078 234.975 203.378 216.078   0.000 203.378   0.000 234.975
     234.975 262.915   0.000 443.560   0.000 543.560   0.000 593.560
       0.000 743.560 END
CSD  2  1  1 16 20  1 24  1  8  1 16  1
     16  1 16  1 16  1 END
' PWR Top Model Axial Profiles - Neutron Source
' IPF=8
BUB  0.0000 128.0160 137.1600 146.3040 155.4480 164.5920
     173.7360 182.8800 END
BUF  1.37800e+00  1.38400e+00  9.63300e-01  6.48100e-01  4.18500e-01
     2.56900e-01  1.47900e-01  7.84000e-02 END
GEND
 UMS Transport Cask - WE17x17 - Top - v2.8
 0 0 1 50
RCC    1      0.000    0.000 -216.078    0.000    0.000  432.156
            116.980
RCC    2      0.000    0.000 -214.808    0.000    0.000  429.616
            116.345
RCC    3      0.000    0.000 -216.088    0.000    0.000  432.176
            116.980
RCC    4      0.000    0.000 -214.818    0.000    0.000  429.636
            116.345
TRC    5      0.000    0.000 -216.078    0.000    0.000   14.402
            108.280  104.915
TRC    6      0.000    0.000  216.078    0.000    0.000  -14.402
            108.280  104.915
RCC    7      0.000    0.000 -262.915    0.000    0.000  525.831
            108.280
RCC    8      0.000    0.000 -216.078    0.000    0.000  432.156
            108.280
RCC    9      0.000    0.000 -216.088    0.000    0.000  432.176
            104.915
RCC   10      0.000    0.000 -216.078    0.000    0.000  432.156
             97.940
RCC   11      0.000    0.000 -216.088    0.000    0.000  432.176
             97.815
RCC   12      0.000    0.000 -208.324    0.000    0.000  416.648
             97.940
BOX   13     95.503  -26.747 -221.158   12.473    0.000    0.000
              0.000    0.000  442.316    0.000   53.494    0.000
RCC   14      0.000    0.000 -216.088    0.000    0.000  432.176
             97.930
```

5.5.3.2.1 PWR Top Radial Fuel Neutron – Normal Conditions (Continued)

```
RCC  15     0.000    0.000 -216.088    0.000    0.000  432.176
           90.945
RCC  16     0.000    0.000 -246.415    0.000    0.000  492.831
           85.865
RCC  17     0.000    0.000 -246.405    0.000    0.000  492.811
           90.955
RCC  18     0.000    0.000 -257.835    0.000    0.000   11.430
           99.517
RCC  19     0.000    0.000  257.835    0.000    0.000  -11.430
           99.517
RCC  20     0.000    0.000 -262.915    0.000    0.000    5.090
           88.722
RCC  21     0.000    0.000  262.915    0.000    0.000   -5.090
           88.722
RCC  22     0.000    0.000 -262.915    0.000    0.000   16.510
           99.517
RCC  23     0.000    0.000  262.915    0.000    0.000  -16.510
           99.517
RCC  24     0.000    0.000 -216.078    0.000    0.000  432.156
          113.106
RCC  25     0.000    0.000 -207.823    0.000    0.000  415.646
          113.106
BOX  26   112.090  -40.640 -203.378   15.240    0.000    0.000
            0.000   81.280    0.000    0.000    0.000  -12.710
BOX  27   111.455  -40.640 -202.108   15.240    0.000    0.000
            0.000   81.280    0.000    0.000    0.000  -13.980
BOX  28  -112.090  -40.640 -203.378  -15.240    0.000    0.000
            0.000   81.280    0.000    0.000    0.000  -12.710
BOX  29  -111.455  -40.640 -202.108  -15.240    0.000    0.000
            0.000   81.280    0.000    0.000    0.000  -13.980
BOX  30   -40.640  112.090 -203.378   81.280    0.000    0.000
            0.000   15.240    0.000    0.000    0.000  -12.710
BOX  31   -40.640  111.455 -202.108   81.280    0.000    0.000
            0.000   15.240    0.000    0.000    0.000  -13.980
BOX  32   -40.640 -112.090 -203.378   81.280    0.000    0.000
            0.000  -15.240    0.000    0.000    0.000  -12.710
BOX  33   -40.640 -111.455 -202.108   81.280    0.000    0.000
            0.000  -15.240    0.000    0.000    0.000  -13.980
BOX  34   112.090  -40.640  203.378   15.240    0.000    0.000
            0.000   81.280    0.000    0.000    0.000   12.710
BOX  35   111.455  -40.640  202.108   15.240    0.000    0.000
            0.000   81.280    0.000    0.000    0.000   13.980
BOX  36  -112.090  -40.640  203.378  -15.240    0.000    0.000
            0.000   81.280    0.000    0.000    0.000   12.710
BOX  37  -111.455  -40.640  202.108  -15.240    0.000    0.000
            0.000   81.280    0.000    0.000    0.000   13.980
BOX  38   -40.640  112.090  203.378   81.280    0.000    0.000
            0.000   15.240    0.000    0.000    0.000   12.710
BOX  39   -40.640  111.455  202.108   81.280    0.000    0.000
            0.000   15.240    0.000    0.000    0.000   13.980
BOX  40   -40.640 -112.090  203.378   81.280    0.000    0.000
            0.000  -15.240    0.000    0.000    0.000   12.710
BOX  41   -40.640 -111.455  202.108   81.280    0.000    0.000
            0.000  -15.240    0.000    0.000    0.000   13.980
RCC  42     0.000  -74.879 -262.925    0.000    0.000    3.820
            6.706
RCC  43     0.000  -74.879  262.925    0.000    0.000   -3.820
            6.706
RCC  44     0.000    0.000 -343.560    0.000    0.000  687.121
          157.480
RCC  45     0.000    0.000 -342.925    0.000    0.000  685.851
          156.845
RCC  46     0.000    0.000 -339.115    0.000    0.000  678.231
          125.984
RCC  47     0.000    0.000 -284.251    0.000    0.000  568.503
          156.845
RCC  48     0.000    0.000 -263.550    0.000    0.000  527.101
          108.915
RCC  49     0.000    0.000 -235.610    0.000    0.000  471.221
          156.845
```

5.5.3.2.1 PWR Top Radial Fuel Neutron – Normal Conditions (Continued)

```
RCC  50      0.000    0.000 -234.975    0.000    0.000  469.951
           157.480
RCC  51      0.000    0.000 -262.915    0.000    0.000  525.831
           116.980
RCC  52      0.000    0.000 -234.985    0.000    0.000  469.971
           135.763
RCC  53      0.000    0.000 -443.560    0.000    0.000  887.121
           216.980
RCC  54      0.000    0.000 -543.560    0.000    0.000 1087.121
           316.980
RCC  55      0.000    0.000 -593.560    0.000    0.000 1187.121
           357.480
RCC  56      0.000    0.000 -643.560    0.000    0.000 1287.121
           416.980
RCC  57      0.000    0.000 -743.560    0.000    0.000 1487.121
           516.980
RCC  58      0.000    0.000 -262.915    0.000    0.000  525.831
           108.280
RCC  59      0.000    0.000 -362.915    0.000    0.000  725.831
           204.915
RCC  60      0.000    0.000 -843.560    0.000    0.000 1687.121
           616.980
RCC  61      0.000    0.000 -943.560    0.000    0.000 1887.121
           716.980
RCC  62      0.000    0.000 -246.405    0.000    0.000  492.811
            85.166
RCC  63      0.000    0.000 -246.405    0.000    0.000  492.811
            83.579
RCC  64      0.000    0.000 -246.405    0.000    0.000  492.811
            83.589
RCC  65      0.000    0.000 -238.785    0.000    0.000  477.571
            83.589
RCC  66      0.000    0.000 -237.363    0.000    0.000  474.726
            83.589
RCC  67      0.000    0.000 -229.438    0.000    0.000  458.876
            83.589
RCC  68      0.000    0.000 -229.997    0.000    0.000  459.994
            83.589
RCC  69     41.910  -59.690 -238.795    0.000    0.000  477.591
             7.620
RCC  70    -41.910   59.690 -238.795    0.000    0.000  477.591
             7.620
RCC  71     41.910  -59.690 -237.373    0.000    0.000  474.746
             5.080
RCC  72    -41.910   59.690 -237.373    0.000    0.000  474.746
             5.080
RCC  73     41.910  -59.690 -229.448    0.000    0.000  458.896
             1.384
RCC  74    -41.910   59.690 -229.448    0.000    0.000  458.896
             1.384
RCC  75     41.910  -59.690 -237.373    0.000    0.000  474.746
             2.832
RCC  76    -41.910   59.690 -237.373    0.000    0.000  474.746
             2.832
RCC  77      0.000    0.000 -221.005    0.000    0.000  442.011
            83.589
RCC  78      0.000    0.000 -215.849    0.000    0.000  431.698
            71.695
RCC  79      0.000    0.000 -198.780    0.000    0.000  397.561
            71.695
RCC  80      0.000    0.000 -182.880    0.000    0.000  365.760
            71.695
RCC  81      0.000    0.000 -221.015    0.000    0.000  442.031
            71.695
RCC  82      0.000    0.000 -215.849    0.000    0.000  431.698
            83.589
RCC  83      0.000    0.000 -198.780    0.000    0.000  397.561
            83.589
RCC  84      0.000    0.000 -182.880    0.000    0.000  365.760
            83.589
RPP  85    -25.456   25.456  -77.320   77.320 -215.849  215.849
```

5.5.3.2.1 PWR Top Radial Fuel Neutron – Normal Conditions (Continued)

```
RPP  86  -52.817  52.817 -52.817  52.817 -215.849 215.849
RPP  87  -77.320  77.320 -25.456  25.456 -215.849 215.849
RPP  88  -25.456  25.456 -77.320  77.320 -198.780 198.780
RPP  89  -52.817  52.817 -52.817  52.817 -198.780 198.780
RPP  90  -77.320  77.320 -25.456  25.456 -198.780 198.780
RPP  91  -25.456  25.456 -77.320  77.320 -182.880 182.880
RPP  92  -52.817  52.817 -52.817  52.817 -182.880 182.880
RPP  93  -77.320  77.320 -25.456  25.456 -182.880 182.880
RPP  94  -25.456  25.456 -77.320  77.320 -221.015 221.015
RPP  95  -52.817  52.817 -52.817  52.817 -221.015 221.015
RPP  96  -77.320  77.320 -25.456  25.456 -221.015 221.015
RCC  97    0.000   0.000 -204.368   0.000    0.000   3.175
          83.185
RCC  98    0.000   0.000  204.368   0.000    0.000  -3.175
          83.185
RCC  99    0.000   0.000 -204.358   0.000    0.000 -14.107
          83.185
RCC 100    0.000   0.000  204.358   0.000    0.000  14.107
          83.185
RCC 101    0.000   0.000 -204.358   0.000    0.000 -14.107
          82.233
RCC 102    0.000   0.000  204.358   0.000    0.000  14.107
          82.233
RCC 103    0.000   0.000 -189.967   0.000    0.000   1.270
          83.185
RCC 104    0.000   0.000 -177.470   0.000    0.000   1.270
          83.185
RCC 105    0.000   0.000 -164.973   0.000    0.000   1.270
          83.185
RCC 106    0.000   0.000 -152.476   0.000    0.000   1.270
          83.185
RCC 107    0.000   0.000 -139.979   0.000    0.000   1.270
          83.185
RCC 108    0.000   0.000 -127.483   0.000    0.000   1.270
          83.185
RCC 109    0.000   0.000 -114.986   0.000    0.000   1.270
          83.185
RCC 110    0.000   0.000 -102.489   0.000    0.000   1.270
          83.185
RCC 111    0.000   0.000  -89.992   0.000    0.000   1.270
          83.185
RCC 112    0.000   0.000  -77.495   0.000    0.000   1.270
          83.185
RCC 113    0.000   0.000  -64.999   0.000    0.000   1.270
          83.185
RCC 114    0.000   0.000  -52.502   0.000    0.000   1.270
          83.185
RCC 115    0.000   0.000  -40.005   0.000    0.000   1.270
          83.185
RCC 116    0.000   0.000  -27.508   0.000    0.000   1.270
          83.185
RCC 117    0.000   0.000  -15.011   0.000    0.000   1.270
          83.185
RCC 118    0.000   0.000   -2.515   0.000    0.000   1.270
          83.185
RCC 119    0.000   0.000  189.967   0.000    0.000  -1.270
          83.185
RCC 120    0.000   0.000  177.470   0.000    0.000  -1.270
          83.185
RCC 121    0.000   0.000  164.973   0.000    0.000  -1.270
          83.185
RCC 122    0.000   0.000  152.476   0.000    0.000  -1.270
          83.185
RCC 123    0.000   0.000  139.979   0.000    0.000  -1.270
          83.185
RCC 124    0.000   0.000  127.483   0.000    0.000  -1.270
          83.185
RCC 125    0.000   0.000  114.986   0.000    0.000  -1.270
          83.185
RCC 126    0.000   0.000  102.489   0.000    0.000  -1.270
          83.185
```

5.5.3.2.1 PWR Top Radial Fuel Neutron – Normal Conditions (Continued)

```
RCC 127    0.000    0.000   89.992    0.000    0.000   -1.270
          83.185
RCC 128    0.000    0.000   77.495    0.000    0.000   -1.270
          83.185
RCC 129    0.000    0.000   64.999    0.000    0.000   -1.270
          83.185
RCC 130    0.000    0.000   52.502    0.000    0.000   -1.270
          83.185
RCC 131    0.000    0.000   40.005    0.000    0.000   -1.270
          83.185
RCC 132    0.000    0.000   27.508    0.000    0.000   -1.270
          83.185
RCC 133    0.000    0.000   15.011    0.000    0.000   -1.270
          83.185
RCC 134    0.000    0.000    2.515    0.000    0.000   -1.270
          83.185
RCC 135    0.000    0.000 -183.718    0.000    0.000    1.270
          82.868
RCC 136    0.000    0.000 -171.221    0.000    0.000    1.270
          82.868
RCC 137    0.000    0.000 -158.725    0.000    0.000    1.270
          82.868
RCC 138    0.000    0.000 -146.228    0.000    0.000    1.270
          82.868
RCC 139    0.000    0.000 -133.731    0.000    0.000    1.270
          82.868
RCC 140    0.000    0.000 -121.234    0.000    0.000    1.270
          82.868
RCC 141    0.000    0.000 -108.737    0.000    0.000    1.270
          82.868
RCC 142    0.000    0.000  -96.241    0.000    0.000    1.270
          82.868
RCC 143    0.000    0.000  -83.744    0.000    0.000    1.270
          82.868
RCC 144    0.000    0.000  -71.247    0.000    0.000    1.270
          82.868
RCC 145    0.000    0.000  -58.750    0.000    0.000    1.270
          82.868
RCC 146    0.000    0.000  -46.253    0.000    0.000    1.270
          82.868
RCC 147    0.000    0.000  -33.757    0.000    0.000    1.270
          82.868
RCC 148    0.000    0.000  -21.260    0.000    0.000    1.270
          82.868
RCC 149    0.000    0.000   -8.763    0.000    0.000    1.270
          82.868
RCC 150    0.000    0.000  183.718    0.000    0.000   -1.270
          82.868
RCC 151    0.000    0.000  171.221    0.000    0.000   -1.270
          82.868
RCC 152    0.000    0.000  158.725    0.000    0.000   -1.270
          82.868
RCC 153    0.000    0.000  146.228    0.000    0.000   -1.270
          82.868
RCC 154    0.000    0.000  133.731    0.000    0.000   -1.270
          82.868
RCC 155    0.000    0.000  121.234    0.000    0.000   -1.270
          82.868
RCC 156    0.000    0.000  108.737    0.000    0.000   -1.270
          82.868
RCC 157    0.000    0.000   96.241    0.000    0.000   -1.270
          82.868
RCC 158    0.000    0.000   83.744    0.000    0.000   -1.270
          82.868
RCC 159    0.000    0.000   71.247    0.000    0.000   -1.270
          82.868
RCC 160    0.000    0.000   58.750    0.000    0.000   -1.270
          82.868
RCC 161    0.000    0.000   46.253    0.000    0.000   -1.270
          82.868
```

5.5.3.2.1 PWR Top Radial Fuel Neutron – Normal Conditions (Continued)

```
RCC 162     0.000     0.000    33.757     0.000     0.000    -1.270
           82.868
RCC 163     0.000     0.000    21.260     0.000     0.000    -1.270
           82.868
RCC 164     0.000     0.000     8.763     0.000     0.000    -1.270
           82.868
BOX 165   116.909     4.404  -202.108    -0.061     1.399     0.000
         -233.757   -10.206     0.000     0.000     0.000   404.216
BOX 166   111.786    34.512  -202.108    -0.421     1.335     0.000
         -223.150   -70.359     0.000     0.000     0.000   404.216
BOX 167    99.044    62.268  -202.108    -0.752     1.181     0.000
         -197.336  -125.717     0.000     0.000     0.000   404.216
BOX 168    79.553    85.781  -202.108    -1.032     0.946     0.000
         -158.074  -172.508     0.000     0.000     0.000   404.216
BOX 169    54.641   103.448  -202.108    -1.242     0.646     0.000
         -108.040  -207.542     0.000     0.000     0.000   404.216
BOX 170    26.005   114.065  -202.108    -1.367     0.303     0.000
          -50.642  -228.433     0.000     0.000     0.000   404.216
BOX 171    -4.404   116.909  -202.108    -1.399    -0.061     0.000
           10.206  -233.757     0.000     0.000     0.000   404.216
BOX 172   -34.512   111.786  -202.108    -1.335    -0.421     0.000
           70.359  -223.150     0.000     0.000     0.000   404.216
BOX 173   -62.268    99.044  -202.108    -1.181    -0.752     0.000
          125.717  -197.336     0.000     0.000     0.000   404.216
BOX 174   -85.781    79.553  -202.108    -0.946    -1.032     0.000
          172.508  -158.074     0.000     0.000     0.000   404.216
BOX 175  -103.448    54.641  -202.108    -0.646    -1.242     0.000
          207.542  -108.040     0.000     0.000     0.000   404.216
BOX 176  -114.065    26.005  -202.108    -0.303    -1.367     0.000
          228.433   -50.642     0.000     0.000     0.000   404.216
RCC 177     0.000     0.000  -202.098     0.000     0.000   404.196
          117.000
 END
' UMS Transport Cask - WE17x17 - Top - Normal - v2.8
FUE   OR  +91   OR  +92   OR  +93
UPL   OR  +88  -91   OR  +89  -92   OR  +90  -93
UEF   OR  +85  -88   OR  +86  -89   OR  +87  -90
SHL   OR  +65  +63  -66  -69  -70   OR  +65  +63  +66
       -67  -71  -72   OR  +65  +63  +67  -77  -73  -74
STL  +64  +63  -65
CSS  +62  -63
PTC   OR  +69  +65  -66   OR  +70  +65  -66
PCM   OR  +71  +66  -75  -68   OR  +72  +66  -76  -68
PMC   OR  +75  +66  -68   OR  +71  +68  -67   OR  +76
      +66  -68   OR  +72  +68  -67
PBC   OR  +73  +67  -77   OR  +74  +67  -77
TWT  +97  -94  -95  -96
TWB  +98  -94  -95  -96
ANU   OR +101  -97  -85  -86  -87   OR +102  -98  -85
      -86  -87   OR  +77  +63  -85  -86  -87  -83  -97
      -98  -99 -100 -103 -104 -105 -106 -107 -108 -109
     -110 -111 -112 -113 -114 -115 -116 -117 -118 -119
     -120 -121 -122 -123 -124 -125 -126 -127 -128 -129
     -130 -131 -132 -133 -134 -135 -136 -137 -138 -139
     -140 -141 -142 -143 -144 -145 -146 -147 -148 -149
     -150 -151 -152 -153 -154 -155 -156 -157 -158 -159
     -160 -161 -162 -163 -164
ANM  +77  +63  +83  -85  -86  -87  -97  -98  -99 -100
     -103 -104 -105 -106 -107 -108 -109 -110 -111 -112
     -113 -114 -115 -116 -117 -118 -119 -120 -121 -122
     -123 -124 -125 -126 -127 -128 -129 -130 -131 -132
     -133 -134 -135 -136 -137 -138 -139 -140 -141 -142
     -143 -144 -145 -146 -147 -148 -149 -150 -151 -152
     -153 -154 -155 -156 -157 -158 -159 -160 -161 -162
     -163 -164
FLT  +99 -101  -97
FLB +100 -102  -98
S01 +103  -94  -95  -96
S02 +104  -94  -95  -96
S03 +105  -94  -95  -96
```

5.5.3.2.1 PWR Top Radial Fuel Neutron – Normal Conditions (Continued)

```
S04 +106  -94  -95  -96
S05 +107  -94  -95  -96
S06 +108  -94  -95  -96
S07 +109  -94  -95  -96
S08 +110  -94  -95  -96
S09 +111  -94  -95  -96
S10 +112  -94  -95  -96
S11 +113  -94  -95  -96
S12 +114  -94  -95  -96
S13 +115  -94  -95  -96
S14 +116  -94  -95  -96
S15 +117  -94  -95  -96
S16 +118  -94  -95  -96
T01 +119  -94  -95  -96
T02 +120  -94  -95  -96
T03 +121  -94  -95  -96
T04 +122  -94  -95  -96
T05 +123  -94  -95  -96
T06 +124  -94  -95  -96
T07 +125  -94  -95  -96
T08 +126  -94  -95  -96
T09 +127  -94  -95  -96
T10 +128  -94  -95  -96
T11 +129  -94  -95  -96
T12 +130  -94  -95  -96
T13 +131  -94  -95  -96
T14 +132  -94  -95  -96
T15 +133  -94  -95  -96
T16 +134  -94  -95  -96
F01 +135  -94  -95  -96
F02 +136  -94  -95  -96
F03 +137  -94  -95  -96
F04 +138  -94  -95  -96
F05 +139  -94  -95  -96
F06 +140  -94  -95  -96
F07 +141  -94  -95  -96
F08 +142  -94  -95  -96
F09 +143  -94  -95  -96
F10 +144  -94  -95  -96
F11 +145  -94  -95  -96
F12 +146  -94  -95  -96
F13 +147  -94  -95  -96
F14 +148  -94  -95  -96
F15 +149  -94  -95  -96
G01 +150  -94  -95  -96
G02 +151  -94  -95  -96
G03 +152  -94  -95  -96
G04 +153  -94  -95  -96
G05 +154  -94  -95  -96
G06 +155  -94  -95  -96
G07 +156  -94  -95  -96
G08 +157  -94  -95  -96
G09 +158  -94  -95  -96
G10 +159  -94  -95  -96
G11 +160  -94  -95  -96
G12 +161  -94  -95  -96
G13 +162  -94  -95  -96
G14 +163  -94  -95  -96
G15 +164  -94  -95  -96
H01 +165  +2  -9
H02 +166  +2  -9
H03 +167  +2  -9
H04 +168  +2  -9
H05 +169  +2  -9
H06 +170  +2  -9
H07 +171  +2  -9
H08 +172  +2  -9
H09 +173  +2  -9
H10 +174  +2  -9
H11 +175  +2  -9
```

5.5.3.2.1 PWR Top Radial Fuel Neutron – Normal Conditions (Continued)

```
H12 +176   +2   -9
NSS   OR   +1   -2   -5   -6  -26  -28  -30  -32  -34
     -36  -38  -40   OR  +27   +1  -26   OR  +29   +1
     -28   OR  +31   +1  -30   OR  +33   +1  -32   OR
     +35   +1  -34   OR  +37   +1  -36   OR  +39   +1
     -38   OR  +41   +1  -40
NS4   +2   -9   -5   -6  -27  -29  -31  -33  -35  -37
     -39  -41 -165 -166 -167 -168 -169 -170 -171 -172
    -173 -174 -175 -176
OSS   OR   +9   -5   -6  -14   OR   +5  -14   OR   +6
     -14
PBG  +14  +10  -11
PBS  +11  +10  -15
ISS  +15  -16
TFG   +7   -8  -16  -22  -23
LID   OR  +18   OR  +20  -42   OR  +19   OR  +21  -43
LVN   OR  +42  +22   OR  +43  +23
LBR   OR  +22  -20  -18   OR  +23  -21  -19
CAV  +16  +17  -62
LSS   OR  +44  -45  -50   OR  +49  -50  -48   OR  +48
      -7  -50
LFW   OR  +46  -48  -49   OR  +47  -48  -49
LSW  +45  -46  -47
TRV   OR  +26   +1   OR  +28   +1   OR  +30   +1   OR
     +32   +1   OR  +34   +1   OR  +36   +1   OR  +38
      +1   OR  +40   +1
FRV  +51  +50   -1   -7
DEP  +52  -51
DEB   OR  +53  -44   OR  +53  +50  -52
DEC  +54  -53
DER  +55  -54
DED  +56  -55
DEE  +57  -56
INV  +60  -57
EXV  +61  -60
 END
'114 Regions
    1    1    1    1    4    1    1    1    3    3
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    3    3
    1    1    1    1    3    1    2    1    2    1
    2    1    2    1
'Universes
114R0
'114 Materials
    1    3    5   12   12   12   12   12 1000 1000
   12   12 1000   19   12   12   12   12   12   12
   12   12   12   12   12   12   12   12   12   12
   12   12   12   12   12   12   12   12   12   12
   12   12   12   12   12   12   12   12   17   17
   17   17   17   17   17   17   17   17   17   17
   17   17   17   17   17   17   17   17   17   17
   17   17   17   17   17   17   17   17   20   20
   20   20   20   20   20   20   20   20   20   20
   12   14   12 1000   13   12   12   12 1000 1000
 1000   12   16   15 1000 1000 1000 1000 1000 1000
 1000 1000 1000    0
0 0
0
END
```

5.5.3.2.2 BWR Bottom Axial End Fitting Gamma – Accident Conditions

```
=SAS4A       PARM='SIZE=1000000'
UMS Transport-ge9x9-21_10Y_40B-Bottom-All Accident-Dry-Axial-End Fitting-Gamma
27N-18COUPLE  INFHOMMEDIUM
' GE9x9-2L Material File
' $Id: ge9x9-21.matl,v 1.3 1997/05/29 22:44:11 csh Exp $
'
' Reference EA790-4202 Rev. 1 Spreadsheet bwr-mat-h2o.xls
'
' Mat'l 1 is fuel, 2 is basket outside of fuel radius, 3 is TSC
'
UO2          1  DEN=1.9583 1.0 END
ZIRCALLOY    1  DEN=0.6769 1.0 END
SS304        1  DEN=0.2228 1.0 END
CARBONSTEEL  1  DEN=0.1932 1.0 END
AL           1  DEN=0.0874 1.0 END
B4C          1  DEN=0.0059 1.0 END
CARBONSTEEL  2  DEN=1.2195 1.0 END
AL           2  DEN=0.1404 1.0 END
'
' Begin Upper Plenum
' 4 is in fuel radius, 5 is outside of fuel radius
'
ZIRCALLOY    3  DEN=0.6551 1.0 END
SS304        3  DEN=0.2198 1.0 END
CARBONSTEEL  3  DEN=0.1486 1.0 END
CARBONSTEEL  4  DEN=0.9381 1.0 END
'
'BEGIN U. EF
'
SS304        5  DEN=0.5708 1.0 END
SS304        6  DEN=0.8665 1.0 END
'
'BEGIN LOWER END FITTING
'
SS304        9  DEN=1.4132 1.0 END
SS304       10  DEN=1.0283 1.0 END
' MASTER MATERIALS LIST
' $Id: master.matl,v 1.3 1997/07/09 19:56:48 csh Exp $
'
CARBONSTEEL    11  1.0       END
SS304          12  1.0       END
PB             13  1.0       END
B-10           14  0.0   8.553-5 END
B-11           14  0.0   3.422-4 END
AL             14  0.0   7.763-3 END
H              14  0.0   5.854-2 END
O              14  0.0   2.609-2 END
C              14  0.0   2.264-2 END
N              14  0.0   1.394-3 END
BALSA          15  1.0       END
REDWOOD        16  1.0       END
AL             17  1.0       END
REG-CONCRETE   18  den=2.2426   1.0    END
N              19  1.E-6  END
CU             20  0.4286 END
SS304          20  0.5714 END
END COMP
IDR=1 ITY=2 IZM=7 ISN=8 FRD=73.235  END
' GE9x9-2L Bottom Axial Biasing Rev. 2.8 - All Accident
' IZM=7
  190.500 209.271 213.716 217.526 228.321 230.861
  243.561 END
   1  9 12  0 12 14 12 END
XEND
TIM=3000.0 NST=2000 NIT=1000 ISO=1 IGO=4 RAN=F05D728AF501 SFA=3.0580e+14
ICS=4 NOD=0 IPF=0 END
SOE
27R0.0
' 9X9_001 10 year Hardware Gamma Spectrum
```

5.5.3.2.2 BWR Bottom Axial End Fitting Gamma – Accident Conditions (Continued)

```
0.0000E-01 0.0000E-01 0.0000E-01 0.0000E-01 1.1331E-15 3.0602E+04
1.9736E+07 5.6617E-06 8.3162E+11 2.9448E+12 7.2217E+08 3.4793E+06
1.0019E+07 1.5852E+08 1.2082E+08 2.4332E+09 1.0086E+10 5.0986E+10 END
SXY 9 -83.579 83.579 -83.579 83.579 190.500 209.271 83.579
    209.271  1.0 1.0 END
' GE9x9-2L Bottom Model Axial Detector Desc. Rev. 2.8 - All Accident Conditions
' ICS=4
CSF 243.561 243.561 343.561 343.561 END
CSL   0.000  46.919  46.919 104.915   0.000  91.641  91.641 204.915 END
CSD  2  1  8  1  2  1  8  1 END
'
GEND
 UMS Transport Cask - GE9x9-2L - Bot - v2.8
 0 0 1 50
RCC   1     0.000     0.000 -212.446     0.000     0.000   424.891
          116.980
RCC   2     0.000     0.000 -211.176     0.000     0.000   422.351
          116.345
RCC   3     0.000     0.000 -206.096     0.000     0.000   412.191
          116.345
RCC   4     0.000     0.000 -243.561     0.000     0.000   487.121
          104.915
RCC   5     0.000     0.000 -230.861     0.000     0.000   461.721
           97.930
RCC   6     0.000     0.000 -215.621     0.000     0.000   431.241
           97.930
RCC   7     0.000     0.000 -215.621     0.000     0.000   431.241
           97.815
RCC   8     0.000     0.000 -207.867     0.000     0.000   415.734
           97.940
BOX   9    95.503   -26.747 -220.701    12.473     0.000     0.000
            0.000     0.000  441.401     0.000    53.494     0.000
RCC  10     0.000     0.000 -230.861     0.000     0.000   461.721
           90.945
RCC  11     0.000     0.000 -228.321     0.000     0.000   456.641
           90.945
RCC  12     0.000     0.000 -217.526     0.000     0.000   435.051
           85.865
RCC  13     0.000     0.000 -213.716     0.000     0.000   427.431
           85.865
WED  14   102.375   -17.780 -179.426    16.154     0.000     0.000
            0.000     0.000   36.059     0.000    35.560     0.000
WED  15   102.375   -17.780 -160.376     7.620     0.000     0.000
            0.000     0.000   17.009     0.000    35.560     0.000
WED  16  -102.375   -17.780 -179.426   -16.154     0.000     0.000
            0.000     0.000   36.059     0.000    35.560     0.000
WED  17  -102.375   -17.780 -160.376    -7.620     0.000     0.000
            0.000     0.000   17.009     0.000    35.560     0.000
BOX  18   102.375   -17.780 -212.471    18.694     0.000     0.000
            0.000     0.000   33.071     0.000    35.560     0.000
BOX  19  -102.375   -17.780 -212.471   -18.694     0.000     0.000
            0.000     0.000   33.071     0.000    35.560     0.000
BOX  20   108.369   -10.160 -212.471    12.700     0.000     0.000
            0.000     0.000   16.231     0.000    20.320     0.000
BOX  21  -108.369   -10.160 -212.471   -12.700     0.000     0.000
            0.000     0.000   16.231     0.000    20.320     0.000
RCC  22   108.369     0.000 -196.240    12.700     0.000     0.000
           12.700
RCC  23  -108.369     0.000 -196.240   -12.700     0.000     0.000
           12.700
WED  24   102.375   -17.780  179.426    16.154     0.000     0.000
            0.000     0.000  -36.059     0.000    35.560     0.000
WED  25   102.375   -17.780  160.376     7.620     0.000     0.000
            0.000     0.000  -17.009     0.000    35.560     0.000
WED  26  -102.375   -17.780  179.426   -16.154     0.000     0.000
            0.000     0.000  -36.059     0.000    35.560     0.000
WED  27  -102.375   -17.780  160.376    -7.620     0.000     0.000
            0.000     0.000  -17.009     0.000    35.560     0.000
BOX  28   102.375   -17.780  212.471    18.694     0.000     0.000
            0.000     0.000  -33.071     0.000    35.560     0.000
```

5.5.3.2.2 BWR Bottom Axial End Fitting Gamma – Accident Conditions (Continued)

```
BOX  29  -102.375  -17.780  212.471  -18.694    0.000    0.000
            0.000    0.000  -33.071    0.000   35.560    0.000
BOX  30   108.369  -10.160  212.471   12.700    0.000    0.000
            0.000    0.000  -16.231    0.000   20.320    0.000
BOX  31  -108.369  -10.160  212.471  -12.700    0.000    0.000
            0.000    0.000  -16.231    0.000   20.320    0.000
RCC  32   108.369    0.000  196.240   12.700    0.000    0.000
           12.700
RCC  33  -108.369    0.000  196.240  -12.700    0.000    0.000
           12.700
RCC  34     0.000    0.000 -324.206    0.000    0.000  648.411
          157.480
RCC  35     0.000    0.000 -323.571    0.000    0.000  647.141
          156.845
RCC  36     0.000    0.000 -319.761    0.000    0.000  639.521
          125.984
RCC  37     0.000    0.000 -264.897    0.000    0.000  529.793
          156.845
RCC  38     0.000    0.000 -244.196    0.000    0.000  488.391
          105.550
RCC  39     0.000    0.000 -216.256    0.000    0.000  432.511
          156.845
RCC  40     0.000    0.000 -215.621    0.000    0.000  431.241
          157.480
RCC  41     0.000  104.925 -220.701    0.000   -3.820    0.000
            6.706
RCC  42     0.000  104.925  220.701    0.000   -3.820    0.000
            6.706
RCC  43     0.000  101.115 -220.701    0.000   -4.988    0.000
            2.540
RCC  44     0.000  101.115  220.701    0.000   -4.988    0.000
            2.540
RCC  45     0.000   96.136 -220.701    0.000   -5.222    0.000
            0.495
RCC  46     0.000   96.136  220.701    0.000   -5.222    0.000
            0.495
RCC  47     0.000   93.455 -220.701    0.000   -9.933    4.155
            0.495
RCC  48     0.000   93.455  220.701    0.000   -9.933   -4.155
            0.495
RCC  49     0.000    0.000 -243.561    0.000    0.000  487.121
          116.980
RCC  50     0.000    0.000 -215.631    0.000    0.000  431.261
          135.763
RCC  51     0.000    0.000 -424.206    0.000    0.000  848.411
          216.980
RCC  52     0.000    0.000 -524.206    0.000    0.000 1048.411
          316.980
RCC  53     0.000    0.000 -574.206    0.000    0.000 1148.411
          357.480
RCC  54     0.000    0.000 -624.206    0.000    0.000 1248.411
          416.980
RCC  55     0.000    0.000 -724.206    0.000    0.000 1448.411
          516.980
RCC  56     0.000    0.000 -243.561    0.000    0.000  487.121
          104.915
RCC  57     0.000    0.000 -343.561    0.000    0.000  687.121
          204.915
RCC  58     0.000    0.000 -824.206    0.000    0.000 1648.411
          616.980
RCC  59     0.000    0.000 -924.206    0.000    0.000 1848.411
          716.980
RCC  60     0.000    0.000 -213.716    0.000    0.000  427.431
           85.166
RCC  61     0.000    0.000 -209.271    0.000    0.000  418.541
           83.579
RCC  62     0.000    0.000 -209.281    0.000    0.000  418.561
           73.235
RCC  63     0.000    0.000 -190.500    0.000    0.000  381.000
           73.235
```

5.5.3.2.2 BWR Bottom Axial End Fitting Gamma – Accident Conditions (Continued)

```
RCC  64     0.000     0.000 -190.500    0.000    0.000  381.000
           73.235
RCC  65     0.000     0.000 -209.281    0.000    0.000  418.561
           73.235
RCC  66     0.000     0.000 -209.271    0.000    0.000  418.541
           83.589
RCC  67     0.000     0.000 -190.500    0.000    0.000  381.000
           83.589
RCC  68     0.000     0.000 -190.500    0.000    0.000  381.000
           83.589
RPP  69   -43.167    43.167  -69.563   69.563 -209.281  209.281
RPP  70   -60.764    60.764  -51.966   51.966 -209.281  209.281
RPP  71   -78.362    78.362  -16.772   16.772 -209.281  209.281
RPP  72   -43.167    43.167  -69.563   69.563 -190.500  190.500
RPP  73   -60.764    60.764  -51.966   51.966 -190.500  190.500
RPP  74   -78.362    78.362  -16.772   16.772 -190.500  190.500
RPP  75   -43.167    43.167  -69.563   69.563 -190.500  190.500
RPP  76   -60.764    60.764  -51.966   51.966 -190.500  190.500
RPP  77   -78.362    78.362  -16.772   16.772 -190.500  190.500
RPP  78   -43.167    43.167  -69.563   69.563 -209.281  209.281
RPP  79   -60.764    60.764  -51.966   51.966 -209.281  209.281
RPP  80   -78.362    78.362  -16.772   16.772 -209.281  209.281
RCC  81     0.000     0.000 -199.111    0.000    0.000    2.540
           83.185
RCC  82     0.000     0.000  199.111    0.000    0.000   -2.540
           83.185
RCC  83     0.000     0.000 -186.665    0.000    0.000    1.588
           83.185
RCC  84     0.000     0.000 -176.936    0.000    0.000    1.588
           83.185
RCC  85     0.000     0.000 -167.208    0.000    0.000    1.588
           83.185
RCC  86     0.000     0.000 -157.480    0.000    0.000    1.588
           83.185
RCC  87     0.000     0.000 -147.752    0.000    0.000    1.588
           83.185
RCC  88     0.000     0.000 -138.024    0.000    0.000    1.588
           83.185
RCC  89     0.000     0.000 -128.295    0.000    0.000    1.588
           83.185
RCC  90     0.000     0.000 -118.567    0.000    0.000    1.588
           83.185
RCC  91     0.000     0.000 -108.839    0.000    0.000    1.588
           83.185
RCC  92     0.000     0.000  -99.111    0.000    0.000    1.588
           83.185
RCC  93     0.000     0.000  -89.383    0.000    0.000    1.588
           83.185
RCC  94     0.000     0.000  -79.654    0.000    0.000    1.588
           83.185
RCC  95     0.000     0.000  -69.926    0.000    0.000    1.588
           83.185
RCC  96     0.000     0.000  -60.198    0.000    0.000    1.588
           83.185
RCC  97     0.000     0.000  -50.470    0.000    0.000    1.588
           83.185
RCC  98     0.000     0.000  -40.742    0.000    0.000    1.588
           83.185
RCC  99     0.000     0.000  -31.013    0.000    0.000    1.588
           83.185
RCC 100     0.000     0.000  -21.285    0.000    0.000    1.588
           83.185
RCC 101     0.000     0.000  -11.557    0.000    0.000    1.588
           83.185
RCC 102     0.000     0.000   -1.829    0.000    0.000    1.588
           83.185
RCC 103     0.000     0.000  186.665    0.000    0.000   -1.588
           83.185
RCC 104     0.000     0.000  176.936    0.000    0.000   -1.588
           83.185
```

5.5.3.2.2 BWR Bottom Axial End Fitting Gamma – Accident Conditions (Continued)

```
RCC 105     0.000    0.000  167.208    0.000    0.000   -1.588
           83.185
RCC 106     0.000    0.000  157.480    0.000    0.000   -1.588
           83.185
RCC 107     0.000    0.000  147.752    0.000    0.000   -1.588
           83.185
RCC 108     0.000    0.000  138.024    0.000    0.000   -1.588
           83.185
RCC 109     0.000    0.000  128.295    0.000    0.000   -1.588
           83.185
RCC 110     0.000    0.000  118.567    0.000    0.000   -1.588
           83.185
RCC 111     0.000    0.000  108.839    0.000    0.000   -1.588
           83.185
RCC 112     0.000    0.000   99.111    0.000    0.000   -1.588
           83.185
RCC 113     0.000    0.000   89.383    0.000    0.000   -1.588
           83.185
RCC 114     0.000    0.000   79.654    0.000    0.000   -1.588
           83.185
RCC 115     0.000    0.000   69.926    0.000    0.000   -1.588
           83.185
RCC 116     0.000    0.000   60.198    0.000    0.000   -1.588
           83.185
RCC 117     0.000    0.000   50.470    0.000    0.000   -1.588
           83.185
RCC 118     0.000    0.000   40.742    0.000    0.000   -1.588
           83.185
RCC 119     0.000    0.000   31.013    0.000    0.000   -1.588
           83.185
RCC 120     0.000    0.000   21.285    0.000    0.000   -1.588
           83.185
RCC 121     0.000    0.000   11.557    0.000    0.000   -1.588
           83.185
RCC 122     0.000    0.000    1.829    0.000    0.000   -1.588
           83.185
RCC 123     0.000    0.000  -74.632    0.000    0.000    1.270
           82.779
RCC 124     0.000    0.000  -64.903    0.000    0.000    1.270
           82.779
RCC 125     0.000    0.000  -55.175    0.000    0.000    1.270
           82.779
RCC 126     0.000    0.000  -45.447    0.000    0.000    1.270
           82.779
RCC 127     0.000    0.000  -35.719    0.000    0.000    1.270
           82.779
RCC 128     0.000    0.000  -25.991    0.000    0.000    1.270
           82.779
RCC 129     0.000    0.000  -16.262    0.000    0.000    1.270
           82.779
RCC 130     0.000    0.000   -6.534    0.000    0.000    1.270
           82.779
RCC 131     0.000    0.000   74.632    0.000    0.000   -1.270
           82.779
RCC 132     0.000    0.000   64.903    0.000    0.000   -1.270
           82.779
RCC 133     0.000    0.000   55.175    0.000    0.000   -1.270
           82.779
RCC 134     0.000    0.000   45.447    0.000    0.000   -1.270
           82.779
RCC 135     0.000    0.000   35.719    0.000    0.000   -1.270
           82.779
RCC 136     0.000    0.000   25.991    0.000    0.000   -1.270
           82.779
RCC 137     0.000    0.000   16.262    0.000    0.000   -1.270
           82.779
RCC 138     0.000    0.000    6.534    0.000    0.000   -1.270
           82.779
BOX 139   116.909    4.404 -212.446   -0.061    1.399    0.000
         -233.757  -10.206    0.000    0.000    0.000  424.891
```

5.5.3.2.2 BWR Bottom Axial End Fitting Gamma – Accident Conditions (Continued)

```
BOX 140   111.786   34.512 -212.446   -0.421    1.335    0.000
         -223.150  -70.359    0.000    0.000    0.000  424.891
BOX 141    99.044   62.268 -212.446   -0.752    1.181    0.000
         -197.336 -125.717    0.000    0.000    0.000  424.891
BOX 142    79.553   85.781 -212.446   -1.032    0.946    0.000
         -158.074 -172.508    0.000    0.000    0.000  424.891
BOX 143    54.641  103.448 -212.446   -1.242    0.646    0.000
         -108.040 -207.542    0.000    0.000    0.000  424.891
BOX 144    26.005  114.065 -212.446   -1.367    0.303    0.000
          -50.642 -228.433    0.000    0.000    0.000  424.891
BOX 145    -4.404  116.909 -212.446   -1.399   -0.061    0.000
           10.206 -233.757    0.000    0.000    0.000  424.891
BOX 146   -34.512  111.786 -212.446   -1.335   -0.421    0.000
           70.359 -223.150    0.000    0.000    0.000  424.891
BOX 147   -62.268   99.044 -212.446   -1.181   -0.752    0.000
          125.717 -197.336    0.000    0.000    0.000  424.891
BOX 148   -85.781   79.553 -212.446   -0.946   -1.032    0.000
          172.508 -158.074    0.000    0.000    0.000  424.891
BOX 149  -103.448   54.641 -212.446   -0.646   -1.242    0.000
          207.542 -108.040    0.000    0.000    0.000  424.891
BOX 150  -114.065   26.005 -212.446   -0.303   -1.367    0.000
          228.433  -50.642    0.000    0.000    0.000  424.891
END
' UMS Transport Cask - GE9x9-2L - Bot - All Accident - v2.8
FUE   OR  +75   OR  +76   OR  +77
LPL   OR  +72  -75   OR  +73  -76   OR  +74  -77
LEF   OR  +69  +61  -72   OR  +70  +61  -73   OR  +71
     +61  -74
CSS  +60  -61
BWT  +82  -78  -79  -80
BWB  +81  -78  -79  -80
ANV  +61  -78  -79  -80  -82  -81  -83  -84  -85  -86
      -87  -88  -89  -90  -91  -92  -93  -94  -95  -96
      -97  -98  -99 -100 -101 -102 -103 -104 -105 -106
     -107 -108 -109 -110 -111 -112 -113 -114 -115 -116
     -117 -118 -119 -120 -121 -122 -123 -124 -125 -126
     -127 -128 -129 -130 -131 -132 -133 -134 -135 -136
     -137 -138
S01  +83  -78  -79  -80
S02  +84  -78  -79  -80
S03  +85  -78  -79  -80
S04  +86  -78  -79  -80
S05  +87  -78  -79  -80
S06  +88  -78  -79  -80
S07  +89  -78  -79  -80
S08  +90  -78  -79  -80
S09  +91  -78  -79  -80
S10  +92  -78  -79  -80
S11  +93  -78  -79  -80
S12  +94  -78  -79  -80
S13  +95  -78  -79  -80
S14  +96  -78  -79  -80
S15  +97  -78  -79  -80
S16  +98  -78  -79  -80
S17  +99  -78  -79  -80
S18 +100  -78  -79  -80
S19 +101  -78  -79  -80
S20 +102  -78  -79  -80
S01 +103  -78  -79  -80
S02 +104  -78  -79  -80
S03 +105  -78  -79  -80
S04 +106  -78  -79  -80
S05 +107  -78  -79  -80
S06 +108  -78  -79  -80
S07 +109  -78  -79  -80
S08 +110  -78  -79  -80
S09 +111  -78  -79  -80
S10 +112  -78  -79  -80
S11 +113  -78  -79  -80
S12 +114  -78  -79  -80
S13 +115  -78  -79  -80
```

5.5.3.2.2 BWR Bottom Axial End Fitting Gamma – Accident Conditions (Continued)

```
S14 +116  -78  -79  -80
S15 +117  -78  -79  -80
S16 +118  -78  -79  -80
S17 +119  -78  -79  -80
S18 +120  -78  -79  -80
S19 +121  -78  -79  -80
S20 +122  -78  -79  -80
F12 +123  -78  -79  -80
F13 +124  -78  -79  -80
F14 +125  -78  -79  -80
F15 +126  -78  -79  -80
F16 +127  -78  -79  -80
F17 +128  -78  -79  -80
F18 +129  -78  -79  -80
F19 +130  -78  -79  -80
F12 +131  -78  -79  -80
F13 +132  -78  -79  -80
F14 +133  -78  -79  -80
F15 +134  -78  -79  -80
F16 +135  -78  -79  -80
F17 +136  -78  -79  -80
F18 +137  -78  -79  -80
F19 +138  -78  -79  -80
OSS   +4   -5  -41  -42  -43  -44  -45  -46  -47  -48
PBS   +8   +6   -9  -11
PBV   +6   -8  -11   OR   +9   +6  -11
BRG   +5   -6  -10  -41  -42  -43  -44  -45  -46  -47
     -48
ISS  +11  -12  -41  -42  -43  -44  -45  -46  -47  -48
AXN  +10  -11
SPC  +12  -13
CAV  +13  -60
VET   OR  +43  -41   OR  +45  -43   OR  +47  -45  -12
VTC  +41   +4
VEB   OR  +44  -42   OR  +46  -44   OR  +48  -46  -12
VBC  +42   +4
DEB  +57   -4
INV  +58  -57
EXV  +59  -58
 END
'78 Regions
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    3    3    3    3    2    1    2
'Universes
78R0
'78 Materials
    1    7    9   12   12   12   19   11   11   11
   11   11   11   11   11   11   11   11   11   11
   11   11   11   11   11   11   11   11   11   11
   11   11   11   11   11   11   11   11   11   11
   11   11   11   11   11   11   11   17   17   17
   17   17   17   17   17   17   17   17   17   17
   17   17   17   12   13 1000   12   12   14 1000
 1000 1000 1000 1000 1000 1000 1000    0
0 0
0
END
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions

```
=SAS4A    PARM='SIZE=1000000'
UMS Transport-we17x17_10Y_45B-Bottom-Normal-Dry-Radial-Fuel-Gamma-UMS Profile
27N-18COUPLE  INFHOMMEDIUM
' WE17x17 Material File
' $Id: we17x17.matl,v 1.4 1997/07/22 22:55:52 csh Exp $
'
' Reference EA790-4009 Rev. 0 Spreadsheet pwr-mat-h2o.xls
'
' Mat'l 1 is fuel, 2 is basket outside of fuel radius, 3 is TSC
'
UO2          1  DEN=2.1530 1.0 END
ZIRCALLOY    1  DEN=0.4494 1.0 END
SS304        1  DEN=0.3807 1.0 END
AL           1  DEN=0.0894 1.0 END
B4C          1  DEN=0.0220 1.0 END
SS304        2  DEN=0.7691 1.0 END
AL           2  DEN=0.2544 1.0 END
'
' Begin Upper Plenum
' 4 is in fuel radius, 5 is outside of fuel radius
'
ZIRCALLOY    3  DEN=0.4494 1.0 END
SS304        3  DEN=1.0318 1.0 END
SS304        4  DEN=0.6101 1.0 END
'
'BEGIN U. EF
'
SS304        5  DEN=1.2537 1.0 END
SS304        6  DEN=1.1366 1.0 END
'
'BEGIN LOWER END FITTING
'
SS304        9  DEN=1.4554 1.0 END
SS304       10  DEN=1.7805 1.0 END
' MASTER MATERIALS LIST
' $Id: master.matl,v 1.3 1997/07/09 19:56:48 csh Exp $
'
CARBONSTEEL   11  1.0     END
SS304         12  1.0     END
PB            13  1.0     END
B-10          14  0.0   8.553-5 END
B-11          14  0.0   3.422-4 END
AL            14  0.0   7.763-3 END
H             14  0.0   5.854-2 END
O             14  0.0   2.609-2 END
C             14  0.0   2.264-2 END
N             14  0.0   1.394-3 END
BALSA         15  1.0     END
REDWOOD       16  1.0     END
AL            17  1.0     END
REG-CONCRETE  18  den=2.2426   1.0     END
N             19  1.E-6  END
CU            20  0.4286 END
SS304         20  0.5714 END
END COMP
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions (Continued)

```
IDR=0 ITY=2 IZM=10 ISN=8 FRD=71.695  END
' WE17x17 Bottom Radial Biasing Rev. 2.8 - Normal Conditions
' IZM=10
    71.695   83.579   85.166   85.865   90.945   97.815
    97.930  104.915  116.345  116.980 END
    1  2  12   0  12  13   0  12  14  12  END
XEND
TIM=3000.0 NST=2000 NIT=1000 ISO=0 IGO=4 RAN=F05D728AF501 SFA=9.0120e+16
ICS=10 NOD=0 IPF=8 END
SOE
27R0.0
' W17_45B 10 year Fuel Gamma Spectrum
1.5743E+05 7.4152E+05 3.7803E+06 9.4200E+06 4.2178E+08 3.7019E+09
5.3258E+10 1.3025E+11 8.1351E+12 7.6492E+13 1.0567E+14 1.8094E+15
1.7568E+14 3.7162E+13 6.0583E+13 2.1673E+14 2.7731E+14 9.8776E+14 END
SXY 1 -83.579 83.579 -83.579 83.579     0.000 182.880 83.579
     182.880   1.0 1.0 END
' WE17x17 Bottom Model Radial Detector Desc. Rev. 2.8 - Normal Conditions
' ICS=10
CSF 104.915 116.980 116.980 116.980 135.763 157.480
    216.980 316.980 357.480 516.980 END
CSL 235.687 238.862 235.687 238.862 202.667 235.687    0.000 202.667
      0.000 238.862 238.862 266.802    0.000 447.447    0.000 547.447
      0.000 597.447    0.000 747.447 END
CSD  1  1  1  1  3 16 18  1 20  1  8  1
    16  1 16  1 16  1 16  1 END
' PWR Bottom Model Axial Profiles - Photon Source
' IPF=8
BUB  0.0000 128.0160 137.1600 146.3040 155.4480 164.5920
    173.7360 182.8800 END
BUF  1.07900e+00  1.08000e+00  9.91200e-01  9.02300e-01  8.13500e-01
     7.24700e-01  6.35800e-01  5.47000e-01 END
GEND
 UMS Transport Cask - WE17x17 - Bot - v2.8
 0 0 1 50
RCC    1       0.000     0.000 -235.687     0.000     0.000   471.373
             116.980
RCC    2       0.000     0.000 -234.417     0.000     0.000   468.833
             116.345
RCC    3       0.000     0.000 -229.337     0.000     0.000   458.673
             116.345
RCC    4       0.000     0.000 -266.802     0.000     0.000   533.603
             104.915
RCC    5       0.000     0.000 -254.102     0.000     0.000   508.203
              97.930
RCC    6       0.000     0.000 -238.862     0.000     0.000   477.723
              97.930
RCC    7       0.000     0.000 -238.862     0.000     0.000   477.723
              97.815
RCC    8       0.000     0.000 -231.108     0.000     0.000   462.216
              97.940
BOX    9      95.503   -26.747 -243.942    12.473     0.000     0.000
               0.000     0.000  487.883     0.000    53.494     0.000
RCC   10       0.000     0.000 -254.102     0.000     0.000   508.203
              90.945
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions (Continued)

```
RCC  11    0.000     0.000 -251.562    0.000    0.000  503.123
          90.945
RCC  12    0.000     0.000 -240.767    0.000    0.000  481.533
          85.865
RCC  13    0.000     0.000 -198.222    0.000    0.000  396.443
          85.865
WED  14  102.375   -17.780 -202.667   16.154    0.000    0.000
           0.000     0.000   36.059    0.000   35.560    0.000
WED  15  102.375   -17.780 -183.617    7.620    0.000    0.000
           0.000     0.000   17.009    0.000   35.560    0.000
WED  16 -102.375   -17.780 -202.667  -16.154    0.000    0.000
           0.000     0.000   36.059    0.000   35.560    0.000
WED  17 -102.375   -17.780 -183.617   -7.620    0.000    0.000
           0.000     0.000   17.009    0.000   35.560    0.000
BOX  18  102.375   -17.780 -235.712   18.694    0.000    0.000
           0.000     0.000   33.071    0.000   35.560    0.000
BOX  19 -102.375   -17.780 -235.712  -18.694    0.000    0.000
           0.000     0.000   33.071    0.000   35.560    0.000
BOX  20  108.369   -10.160 -235.712   12.700    0.000    0.000
           0.000     0.000   16.231    0.000   20.320    0.000
BOX  21 -108.369   -10.160 -235.712  -12.700    0.000    0.000
           0.000     0.000   16.231    0.000   20.320    0.000
RCC  22  108.369     0.000 -219.481   12.700    0.000    0.000
          12.700
RCC  23 -108.369     0.000 -219.481  -12.700    0.000    0.000
          12.700
WED  24  102.375   -17.780  202.667   16.154    0.000    0.000
           0.000     0.000  -36.059    0.000   35.560    0.000
WED  25  102.375   -17.780  183.617    7.620    0.000    0.000
           0.000     0.000  -17.009    0.000   35.560    0.000
WED  26 -102.375   -17.780  202.667  -16.154    0.000    0.000
           0.000     0.000  -36.059    0.000   35.560    0.000
WED  27 -102.375   -17.780  183.617   -7.620    0.000    0.000
           0.000     0.000  -17.009    0.000   35.560    0.000
BOX  28  102.375   -17.780  235.712   18.694    0.000    0.000
           0.000     0.000  -33.071    0.000   35.560    0.000
BOX  29 -102.375   -17.780  235.712  -18.694    0.000    0.000
           0.000     0.000  -33.071    0.000   35.560    0.000
BOX  30  108.369   -10.160  235.712   12.700    0.000    0.000
           0.000     0.000  -16.231    0.000   20.320    0.000
BOX  31 -108.369   -10.160  235.712  -12.700    0.000    0.000
           0.000     0.000  -16.231    0.000   20.320    0.000
RCC  32  108.369     0.000  219.481   12.700    0.000    0.000
          12.700
RCC  33 -108.369     0.000  219.481  -12.700    0.000    0.000
          12.700
RCC  34    0.000     0.000 -347.447    0.000    0.000  694.893
         157.480
RCC  35    0.000     0.000 -346.812    0.000    0.000  693.623
         156.845
RCC  36    0.000     0.000 -343.002    0.000    0.000  686.003
         125.984
RCC  37    0.000     0.000 -288.138    0.000    0.000  576.275
         156.845
RCC  38    0.000     0.000 -267.437    0.000    0.000  534.873
         105.550
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions (Continued)

```
RCC  39     0.000     0.000 -239.497     0.000     0.000   478.993
          156.845
RCC  40     0.000     0.000 -238.862     0.000     0.000   477.723
          157.480
RCC  41     0.000   104.925 -243.942     0.000    -3.820     0.000
            6.706
RCC  42     0.000   104.925  243.942     0.000    -3.820     0.000
            6.706
RCC  43     0.000   101.115 -243.942     0.000    -4.988     0.000
            2.540
RCC  44     0.000   101.115  243.942     0.000    -4.988     0.000
            2.540
RCC  45     0.000    96.136 -243.942     0.000    -5.222     0.000
            0.495
RCC  46     0.000    96.136  243.942     0.000    -5.222     0.000
            0.495
RCC  47     0.000    93.455 -243.942     0.000    -9.933     4.155
            0.495
RCC  48     0.000    93.455  243.942     0.000    -9.933    -4.155
            0.495
RCC  49     0.000     0.000 -266.802     0.000     0.000   533.603
          116.980
RCC  50     0.000     0.000 -238.872     0.000     0.000   477.743
          135.763
RCC  51     0.000     0.000 -447.447     0.000     0.000   894.893
          216.980
RCC  52     0.000     0.000 -547.447     0.000     0.000  1094.893
          316.980
RCC  53     0.000     0.000 -597.447     0.000     0.000  1194.893
          357.480
RCC  54     0.000     0.000 -647.447     0.000     0.000  1294.893
          416.980
RCC  55     0.000     0.000 -747.447     0.000     0.000  1494.893
          516.980
RCC  56     0.000     0.000 -266.802     0.000     0.000   533.603
          104.915
RCC  57     0.000     0.000 -366.802     0.000     0.000   733.603
          204.915
RCC  58     0.000     0.000 -847.447     0.000     0.000  1694.893
          616.980
RCC  59     0.000     0.000 -947.447     0.000     0.000  1894.893
          716.980
RCC  60     0.000     0.000 -198.222     0.000     0.000   396.443
           85.166
RCC  61     0.000     0.000 -193.777     0.000     0.000   387.553
           83.579
RCC  62     0.000     0.000 -193.787     0.000     0.000   387.573
           71.695
RCC  63     0.000     0.000 -182.880     0.000     0.000   365.760
           71.695
RCC  64     0.000     0.000 -182.880     0.000     0.000   365.760
           71.695
RCC  65     0.000     0.000 -193.787     0.000     0.000   387.573
           71.695
RCC  66     0.000     0.000 -193.777     0.000     0.000   387.553
           83.589
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions (Continued)

```
RCC  67    0.000    0.000 -182.880    0.000    0.000  365.760
              83.589
RCC  68    0.000    0.000 -182.880    0.000    0.000  365.760
              83.589
RPP  69  -25.456   25.456  -77.320   77.320 -193.787  193.787
RPP  70  -52.817   52.817  -52.817   52.817 -193.787  193.787
RPP  71  -77.320   77.320  -25.456   25.456 -193.787  193.787
RPP  72  -25.456   25.456  -77.320   77.320 -182.880  182.880
RPP  73  -52.817   52.817  -52.817   52.817 -182.880  182.880
RPP  74  -77.320   77.320  -25.456   25.456 -182.880  182.880
RPP  75  -25.456   25.456  -77.320   77.320 -182.880  182.880
RPP  76  -52.817   52.817  -52.817   52.817 -182.880  182.880
RPP  77  -77.320   77.320  -25.456   25.456 -182.880  182.880
RPP  78  -25.456   25.456  -77.320   77.320 -193.787  193.787
RPP  79  -52.817   52.817  -52.817   52.817 -193.787  193.787
RPP  80  -77.320   77.320  -25.456   25.456 -193.787  193.787
RCC  81    0.000    0.000 -191.237    0.000    0.000    2.540
              83.185
RCC  82    0.000    0.000  191.237    0.000    0.000   -2.540
              83.185
RCC  83    0.000    0.000 -177.470    0.000    0.000    1.270
              83.185
RCC  84    0.000    0.000 -164.973    0.000    0.000    1.270
              83.185
RCC  85    0.000    0.000 -152.476    0.000    0.000    1.270
              83.185
RCC  86    0.000    0.000 -139.979    0.000    0.000    1.270
              83.185
RCC  87    0.000    0.000 -127.483    0.000    0.000    1.270
              83.185
RCC  88    0.000    0.000 -114.986    0.000    0.000    1.270
              83.185
RCC  89    0.000    0.000 -102.489    0.000    0.000    1.270
              83.185
RCC  90    0.000    0.000  -89.992    0.000    0.000    1.270
              83.185
RCC  91    0.000    0.000  -77.495    0.000    0.000    1.270
              83.185
RCC  92    0.000    0.000  -64.999    0.000    0.000    1.270
              83.185
RCC  93    0.000    0.000  -52.502    0.000    0.000    1.270
              83.185
RCC  94    0.000    0.000  -40.005    0.000    0.000    1.270
              83.185
RCC  95    0.000    0.000  -27.508    0.000    0.000    1.270
              83.185
RCC  96    0.000    0.000  -15.011    0.000    0.000    1.270
              83.185
RCC  97    0.000    0.000   -2.515    0.000    0.000    1.270
              83.185
RCC  98    0.000    0.000  177.470    0.000    0.000   -1.270
              83.185
RCC  99    0.000    0.000  164.973    0.000    0.000   -1.270
              83.185
RCC 100    0.000    0.000  152.476    0.000    0.000   -1.270
              83.185
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions (Continued)

```
RCC 101      0.000      0.000  139.979      0.000      0.000     -1.270
            83.185
RCC 102      0.000      0.000  127.483      0.000      0.000     -1.270
            83.185
RCC 103      0.000      0.000  114.986      0.000      0.000     -1.270
            83.185
RCC 104      0.000      0.000  102.489      0.000      0.000     -1.270
            83.185
RCC 105      0.000      0.000   89.992      0.000      0.000     -1.270
            83.185
RCC 106      0.000      0.000   77.495      0.000      0.000     -1.270
            83.185
RCC 107      0.000      0.000   64.999      0.000      0.000     -1.270
            83.185
RCC 108      0.000      0.000   52.502      0.000      0.000     -1.270
            83.185
RCC 109      0.000      0.000   40.005      0.000      0.000     -1.270
            83.185
RCC 110      0.000      0.000   27.508      0.000      0.000     -1.270
            83.185
RCC 111      0.000      0.000   15.011      0.000      0.000     -1.270
            83.185
RCC 112      0.000      0.000    2.515      0.000      0.000     -1.270
            83.185
RCC 113      0.000      0.000 -171.221      0.000      0.000      1.270
            82.868
RCC 114      0.000      0.000 -158.725      0.000      0.000      1.270
            82.868
RCC 115      0.000      0.000 -146.228      0.000      0.000      1.270
            82.868
RCC 116      0.000      0.000 -133.731      0.000      0.000      1.270
            82.868
RCC 117      0.000      0.000 -121.234      0.000      0.000      1.270
            82.868
RCC 118      0.000      0.000 -108.737      0.000      0.000      1.270
            82.868
RCC 119      0.000      0.000  -96.241      0.000      0.000      1.270
            82.868
RCC 120      0.000      0.000  -83.744      0.000      0.000      1.270
            82.868
RCC 121      0.000      0.000  -71.247      0.000      0.000      1.270
            82.868
RCC 122      0.000      0.000  -58.750      0.000      0.000      1.270
            82.868
RCC 123      0.000      0.000  -46.253      0.000      0.000      1.270
            82.868
RCC 124      0.000      0.000  -33.757      0.000      0.000      1.270
            82.868
RCC 125      0.000      0.000  -21.260      0.000      0.000      1.270
            82.868
RCC 126      0.000      0.000   -8.763      0.000      0.000      1.270
            82.868
RCC 127      0.000      0.000  171.221      0.000      0.000     -1.270
            82.868
RCC 128      0.000      0.000  158.725      0.000      0.000     -1.270
            82.868
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions (Continued)

```
RCC 129       0.000      0.000  146.228     0.000     0.000    -1.270
            82.868
RCC 130       0.000      0.000  133.731     0.000     0.000    -1.270
            82.868
RCC 131       0.000      0.000  121.234     0.000     0.000    -1.270
            82.868
RCC 132       0.000      0.000  108.737     0.000     0.000    -1.270
            82.868
RCC 133       0.000      0.000   96.241     0.000     0.000    -1.270
            82.868
RCC 134       0.000      0.000   83.744     0.000     0.000    -1.270
            82.868
RCC 135       0.000      0.000   71.247     0.000     0.000    -1.270
            82.868
RCC 136       0.000      0.000   58.750     0.000     0.000    -1.270
            82.868
RCC 137       0.000      0.000   46.253     0.000     0.000    -1.270
            82.868
RCC 138       0.000      0.000   33.757     0.000     0.000    -1.270
            82.868
RCC 139       0.000      0.000   21.260     0.000     0.000    -1.270
            82.868
RCC 140       0.000      0.000    8.763     0.000     0.000    -1.270
            82.868
BOX 141     116.909      4.404 -235.687    -0.061     1.399     0.000
           -233.757    -10.206    0.000     0.000     0.000   471.373
BOX 142     111.786     34.512 -235.687    -0.421     1.335     0.000
           -223.150    -70.359    0.000     0.000     0.000   471.373
BOX 143      99.044     62.268 -235.687    -0.752     1.181     0.000
           -197.336   -125.717    0.000     0.000     0.000   471.373
BOX 144      79.553     85.781 -235.687    -1.032     0.946     0.000
           -158.074   -172.508    0.000     0.000     0.000   471.373
BOX 145      54.641    103.448 -235.687    -1.242     0.646     0.000
           -108.040   -207.542    0.000     0.000     0.000   471.373
BOX 146      26.005    114.065 -235.687    -1.367     0.303     0.000
            -50.642   -228.433    0.000     0.000     0.000   471.373
BOX 147      -4.404    116.909 -235.687    -1.399    -0.061     0.000
             10.206   -233.757    0.000     0.000     0.000   471.373
BOX 148     -34.512    111.786 -235.687    -1.335    -0.421     0.000
             70.359   -223.150    0.000     0.000     0.000   471.373
BOX 149     -62.268     99.044 -235.687    -1.181    -0.752     0.000
            125.717   -197.336    0.000     0.000     0.000   471.373
BOX 150     -85.781     79.553 -235.687    -0.946    -1.032     0.000
            172.508   -158.074    0.000     0.000     0.000   471.373
BOX 151    -103.448     54.641 -235.687    -0.646    -1.242     0.000
            207.542   -108.040    0.000     0.000     0.000   471.373
BOX 152    -114.065     26.005 -235.687    -0.303    -1.367     0.000
            228.433    -50.642    0.000     0.000     0.000   471.373
END
 UMS Transport Cask - WE17x17 - Bot - Normal - v2.8
FUE   OR  +75  OR  +76  OR  +77
LPL   OR  +72 -75  OR  +73 -76  OR  +74 -77
LEF   OR  +69  +61  -72  OR  +70  +61  -73  OR  +71
      +61  -74
CSS   +60  -61
BWT   +82  -78  -79  -80
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions (Continued)

```
BWB  +81  -78  -79  -80
ANV  +61  -78  -79  -80  -82  -81  -83  -84  -85  -86
          -87  -88  -89  -90  -91  -92  -93  -94  -95  -96
          -97  -98  -99 -100 -101 -102 -103 -104 -105 -106
         -107 -108 -109 -110 -111 -112 -113 -114 -115 -116
         -117 -118 -119 -120 -121 -122 -123 -124 -125 -126
         -127 -128 -129 -130 -131 -132 -133 -134 -135 -136
         -137 -138 -139 -140
S01  +83  -78  -79  -80
S02  +84  -78  -79  -80
S03  +85  -78  -79  -80
S04  +86  -78  -79  -80
S05  +87  -78  -79  -80
S06  +88  -78  -79  -80
S07  +89  -78  -79  -80
S08  +90  -78  -79  -80
S09  +91  -78  -79  -80
S10  +92  -78  -79  -80
S11  +93  -78  -79  -80
S12  +94  -78  -79  -80
S13  +95  -78  -79  -80
S14  +96  -78  -79  -80
S15  +97  -78  -79  -80
T01  +98  -78  -79  -80
T02  +99  -78  -79  -80
T03 +100  -78  -79  -80
T04 +101  -78  -79  -80
T05 +102  -78  -79  -80
T06 +103  -78  -79  -80
T07 +104  -78  -79  -80
T08 +105  -78  -79  -80
T09 +106  -78  -79  -80
T10 +107  -78  -79  -80
T11 +108  -78  -79  -80
T12 +109  -78  -79  -80
T13 +110  -78  -79  -80
T14 +111  -78  -79  -80
T15 +112  -78  -79  -80
F01 +113  -78  -79  -80
F02 +114  -78  -79  -80
F03 +115  -78  -79  -80
F04 +116  -78  -79  -80
F05 +117  -78  -79  -80
F06 +118  -78  -79  -80
F07 +119  -78  -79  -80
F08 +120  -78  -79  -80
F09 +121  -78  -79  -80
F10 +122  -78  -79  -80
F11 +123  -78  -79  -80
F12 +124  -78  -79  -80
F13 +125  -78  -79  -80
F14 +126  -78  -79  -80
G01 +127  -78  -79  -80
G02 +128  -78  -79  -80
G03 +129  -78  -79  -80
G04 +130  -78  -79  -80
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions (Continued)

```
G05 +131  -78  -79  -80
G06 +132  -78  -79  -80
G07 +133  -78  -79  -80
G08 +134  -78  -79  -80
G09 +135  -78  -79  -80
G10 +136  -78  -79  -80
G11 +137  -78  -79  -80
G12 +138  -78  -79  -80
G13 +139  -78  -79  -80
G14 +140  -78  -79  -80
H01 +141   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H02 +142   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H03 +143   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H04 +144   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H05 +145   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H06 +146   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H07 +147   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H08 +148   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H09 +149   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H10 +150   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H11 +151   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H12 +152   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
NSS   +1   -2   -4  -18  -19  -28  -29
NS4   OR   +3   -4  -14  -16  -18  -19  -24  -26  -28
           -29 -141 -142 -143 -144 -145 -146 -147 -148 -149
          -150 -151 -152  OR  +15   -4   OR  +17   -4   OR
           +25   -4   OR  +27   -4
NSE   +2   -3   -4  -18  -19  -28  -29
OSS   +4   -5  -41  -42  -43  -44  -45  -46  -47  -48
PBG   +6   -7
PBS   +7  -11
BRG   +5   -6  -10  -41  -42  -43  -44  -45  -46  -47
           -48
ISS  +11  -12  -41  -42  -43  -44  -45  -46  -47  -48
AXN  +10  -11
SPC  +12  -13
CAV  +13  -60
PL1   OR  +14   +1   -4  -15   OR  +18   +1   -4  -20
           -22
PL2   OR  +16   +1   -4  -17   OR  +19   +1   -4  -21
           -23
PU1   OR  +24   +1   -4  -25   OR  +28   +1   -4  -30
           -32
```

5.5.3.2.3 PWR Bottom Radial Fuel Gamma – Normal Conditions (Continued)

```
PU2   OR  +26   +1   -4  -27   OR  +29   +1   -4  -31
          -33
LSS   OR  +34  -35  -40   OR  +39  -40  -38   OR  +38
          -4  -40
LFW   OR  +36  -38  -39   OR  +37  -38  -39
LSW  +35  -36  -37
VL1   OR  +22   +1   OR  +20   +1
VL2   OR  +23   +1   OR  +21   +1
VU1   OR  +32   +1   OR  +30   +1
VU2   OR  +33   +1   OR  +31   +1
LGP  +49  +40   -1   -4
VET   OR  +43  -41   OR  +45  -43   OR  +47  -45  -12
VTC  +41   +4
VEB   OR  +44  -42   OR  +46  -44   OR  +48  -46  -12
VBC  +42   +4
DEP  +50  +40  -49
DEB   OR  +51  -34   OR  +51  +40  -50
DEC  +52  -51
DER  +53  -52
DED  +54  -53
DEE  +55  -54
INV  +58  -55
EXV  +59  -58
END
'112 Regions
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    3    3    3    3    4
    3    3    3    3    2    1    2    1    2    1
    2    1
'Universes
112R0
'112 Materials
    1    7    9   12   12   12   19   12   12   12
   12   12   12   12   12   12   12   12   12   12
   12   12   12   12   12   12   12   12   12   12
   12   12   12   12   12   12   12   17   17   17
   17   17   17   17   17   17   17   17   17   17
   17   17   17   17   17   17   17   17   17   17
   17   17   17   17   17   20   20   20   20   20
   20   20   20   20   20   20   20   12   14 1000
   12 1000   13   12   12   14 1000 1000   12   12
   12   12   12   16   15 1000 1000 1000 1000 1000
 1000   12 1000   12 1000 1000 1000 1000 1000 1000
 1000    0
0 0
0
END
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions

```
=SAS4A    PARM='SIZE=1000000'
UMS Transport-we17x17 10Y 45B-Top-Normal-Dry-Radial-End Fitting-Gamma
27N-18COUPLE  INFHOMMEDIUM
' WE17x17 Material File
' $Id: we17x17.matl,v 1.4 1997/07/22 22:55:52 csh Exp $
'
' Reference EA790-4009 Rev. 0 Spreadsheet pwr-mat-h2o.xls
'
' Mat'l 1 is fuel, 2 is basket outside of fuel radius, 3 is TSC
'
UO2          1  DEN=2.1530 1.0 END
ZIRCALLOY  1  DEN=0.4494 1.0 END
SS304        1  DEN=0.3807 1.0 END
AL           1  DEN=0.0894 1.0 END
B4C          1  DEN=0.0220 1.0 END
SS304        2  DEN=0.7691 1.0 END
AL           2  DEN=0.2544 1.0 END
'
' Begin Upper Plenum
' 4 is in fuel radius, 5 is outside of fuel radius
'
ZIRCALLOY  3  DEN=0.4494 1.0 END
SS304        3  DEN=1.0318 1.0 END
SS304        4  DEN=0.6101 1.0 END
'
'BEGIN U. EF
'
SS304        5  DEN=1.2537 1.0 END
SS304        6  DEN=1.1366 1.0 END
'
'BEGIN LOWER END FITTING
'
SS304        9  DEN=1.4554 1.0 END
SS304       10  DEN=1.7805 1.0 END
' MASTER MATERIALS LIST
' $Id: master.matl,v 1.3 1997/07/09 19:56:48 csh Exp $
'
CARBONSTEEL   11  1.0    END
SS304         12  1.0    END
PB            13  1.0    END
B-10          14  0.0    8.553-5 END
B-11          14  0.0    3.422-4 END
AL            14  0.0    7.763-3 END
H             14  0.0    5.854-2 END
O             14  0.0    2.609-2 END
C             14  0.0    2.264-2 END
N             14  0.0    1.394-3 END
BALSA         15  1.0    END
REDWOOD       16  1.0    END
AL            17  1.0    END
REG-CONCRETE  18  den=2.2426  1.0    END
N             19  1.E-6  END
CU            20  0.4286 END
SS304         20  0.5714 END
END COMP
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
IDR=0 ITY=2 IZM=10 ISN=8 FRD=71.695  END
' WE17x17 Top Radial Biasing Rev. 2.8 - Normal Conditions
' IZM=10
   71.695  83.579  85.166  85.865  90.945  97.815
   97.930 104.915 116.345 116.980 END
   5 6 12  0 12 13  0 12 14 12 END
XEND
TIM=3000.0 NST=2000 NIT=1000 ISO=1 IGO=4 RAN=F05D728AF501 SFA=2.1140e+14
ICS=9 NOD=0 IPF=0 END
SOE
27R0.0
' W17_45B 10 year Hardware Gamma Spectrum
0.0000E-01 0.0000E-01 0.0000E-01 0.0000E-01 2.2491E-15 3.2497E+04
2.0958E+07 7.5979E-06 8.8313E+11 3.1272E+12 1.0399E+09 3.6948E+06
1.0639E+07 1.6834E+08 1.2830E+08 2.5839E+09 1.0711E+10 5.4096E+10 END
SXY 5 -83.579 83.579 -83.579 83.579 198.780 215.849 83.579
     215.849  1.0 1.0 END
' WE17x17 Top Model Radial Detector Desc. Rev. 2.8 - Normal Conditions
' ICS=9
CSF 108.280 116.980 116.980 135.763 157.480 216.980
     316.980 357.480 516.980 END
CSL 216.078 234.975 203.378 216.078    0.000 203.378    0.000 234.975
     234.975 262.915    0.000 443.560    0.000 543.560    0.000 593.560
       0.000 743.560 END
CSD  2   1   1 16 20  1 24  1   8   1 16   1
     16   1 16   1 16   1 END
'
GEND
 UMS Transport Cask - WE17x17 - Top - v2.8
 0 0 1 50
RCC    1      0.000      0.000 -216.078      0.000      0.000   432.156
              116.980
RCC    2      0.000      0.000 -214.808      0.000      0.000   429.616
              116.345
RCC    3      0.000      0.000 -216.088      0.000      0.000   432.176
              116.980
RCC    4      0.000      0.000 -214.818      0.000      0.000   429.636
              116.345
TRC    5      0.000      0.000 -216.078      0.000      0.000    14.402
              108.280   104.915
TRC    6      0.000      0.000  216.078      0.000      0.000   -14.402
              108.280   104.915
RCC    7      0.000      0.000 -262.915      0.000      0.000   525.831
              108.280
RCC    8      0.000      0.000 -216.078      0.000      0.000   432.156
              108.280
RCC    9      0.000      0.000 -216.088      0.000      0.000   432.176
              104.915
RCC   10      0.000      0.000 -216.078      0.000      0.000   432.156
               97.940
RCC   11      0.000      0.000 -216.088      0.000      0.000   432.176
               97.815
RCC   12      0.000      0.000 -208.324      0.000      0.000   416.648
               97.940
BOX   13     95.503  -26.747 -221.158     12.473      0.000     0.000
              0.000      0.000  442.316      0.000     53.494     0.000
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
RCC 14     0.000    0.000 -216.088    0.000    0.000  432.176
          97.930
RCC 15     0.000    0.000 -216.088    0.000    0.000  432.176
          90.945
RCC 16     0.000    0.000 -246.415    0.000    0.000  492.831
          85.865
RCC 17     0.000    0.000 -246.405    0.000    0.000  492.811
          90.955
RCC 18     0.000    0.000 -257.835    0.000    0.000   11.430
          99.517
RCC 19     0.000    0.000  257.835    0.000    0.000  -11.430
          99.517
RCC 20     0.000    0.000 -262.915    0.000    0.000    5.090
          88.722
RCC 21     0.000    0.000  262.915    0.000    0.000   -5.090
          88.722
RCC 22     0.000    0.000 -262.915    0.000    0.000   16.510
          99.517
RCC 23     0.000    0.000  262.915    0.000    0.000  -16.510
          99.517
RCC 24     0.000    0.000 -216.078    0.000    0.000  432.156
         113.106
RCC 25     0.000    0.000 -207.823    0.000    0.000  415.646
         113.106
BOX 26   112.090  -40.640 -203.378   15.240    0.000    0.000
           0.000   81.280    0.000    0.000    0.000  -12.710
BOX 27   111.455  -40.640 -202.108   15.240    0.000    0.000
           0.000   81.280    0.000    0.000    0.000  -13.980
BOX 28  -112.090  -40.640 -203.378  -15.240    0.000    0.000
           0.000   81.280    0.000    0.000    0.000  -12.710
BOX 29  -111.455  -40.640 -202.108  -15.240    0.000    0.000
           0.000   81.280    0.000    0.000    0.000  -13.980
BOX 30   -40.640  112.090 -203.378   81.280    0.000    0.000
           0.000   15.240    0.000    0.000    0.000  -12.710
BOX 31   -40.640  111.455 -202.108   81.280    0.000    0.000
           0.000   15.240    0.000    0.000    0.000  -13.980
BOX 32   -40.640 -112.090 -203.378   81.280    0.000    0.000
           0.000  -15.240    0.000    0.000    0.000  -12.710
BOX 33   -40.640 -111.455 -202.108   81.280    0.000    0.000
           0.000  -15.240    0.000    0.000    0.000  -13.980
BOX 34   112.090  -40.640  203.378   15.240    0.000    0.000
           0.000   81.280    0.000    0.000    0.000   12.710
BOX 35   111.455  -40.640  202.108   15.240    0.000    0.000
           0.000   81.280    0.000    0.000    0.000   13.980
BOX 36  -112.090  -40.640  203.378  -15.240    0.000    0.000
           0.000   81.280    0.000    0.000    0.000   12.710
BOX 37  -111.455  -40.640  202.108  -15.240    0.000    0.000
           0.000   81.280    0.000    0.000    0.000   13.980
BOX 38   -40.640  112.090  203.378   81.280    0.000    0.000
           0.000   15.240    0.000    0.000    0.000   12.710
BOX 39   -40.640  111.455  202.108   81.280    0.000    0.000
           0.000   15.240    0.000    0.000    0.000   13.980
BOX 40   -40.640 -112.090  203.378   81.280    0.000    0.000
           0.000  -15.240    0.000    0.000    0.000   12.710
BOX 41   -40.640 -111.455  202.108   81.280    0.000    0.000
           0.000  -15.240    0.000    0.000    0.000   13.980
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
RCC  42      0.000 -74.879 -262.925     0.000     0.000     3.820
             6.706
RCC  43      0.000 -74.879  262.925     0.000     0.000    -3.820
             6.706
RCC  44      0.000   0.000 -343.560     0.000     0.000   687.121
           157.480
RCC  45      0.000   0.000 -342.925     0.000     0.000   685.851
           156.845
RCC  46      0.000   0.000 -339.115     0.000     0.000   678.231
           125.984
RCC  47      0.000   0.000 -284.251     0.000     0.000   568.503
           156.845
RCC  48      0.000   0.000 -263.550     0.000     0.000   527.101
           108.915
RCC  49      0.000   0.000 -235.610     0.000     0.000   471.221
           156.845
RCC  50      0.000   0.000 -234.975     0.000     0.000   469.951
           157.480
RCC  51      0.000   0.000 -262.915     0.000     0.000   525.831
           116.980
RCC  52      0.000   0.000 -234.985     0.000     0.000   469.971
           135.763
RCC  53      0.000   0.000 -443.560     0.000     0.000   887.121
           216.980
RCC  54      0.000   0.000 -543.560     0.000     0.000  1087.121
           316.980
RCC  55      0.000   0.000 -593.560     0.000     0.000  1187.121
           357.480
RCC  56      0.000   0.000 -643.560     0.000     0.000  1287.121
           416.980
RCC  57      0.000   0.000 -743.560     0.000     0.000  1487.121
           516.980
RCC  58      0.000   0.000 -262.915     0.000     0.000   525.831
           108.280
RCC  59      0.000   0.000 -362.915     0.000     0.000   725.831
           204.915
RCC  60      0.000   0.000 -843.560     0.000     0.000  1687.121
           616.980
RCC  61      0.000   0.000 -943.560     0.000     0.000  1887.121
           716.980
RCC  62      0.000   0.000 -246.405     0.000     0.000   492.811
            85.166
RCC  63      0.000   0.000 -246.405     0.000     0.000   492.811
            83.579
RCC  64      0.000   0.000 -246.405     0.000     0.000   492.811
            83.589
RCC  65      0.000   0.000 -238.785     0.000     0.000   477.571
            83.589
RCC  66      0.000   0.000 -237.363     0.000     0.000   474.726
            83.589
RCC  67      0.000   0.000 -229.438     0.000     0.000   458.876
            83.589
RCC  68      0.000   0.000 -229.997     0.000     0.000   459.994
            83.589
RCC  69     41.910 -59.690 -238.795     0.000     0.000   477.591
             7.620
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
RCC  70   -41.910   59.690 -238.795    0.000    0.000  477.591
            7.620
RCC  71    41.910  -59.690 -237.373    0.000    0.000  474.746
            5.080
RCC  72   -41.910   59.690 -237.373    0.000    0.000  474.746
            5.080
RCC  73    41.910  -59.690 -229.448    0.000    0.000  458.896
            1.384
RCC  74   -41.910   59.690 -229.448    0.000    0.000  458.896
            1.384
RCC  75    41.910  -59.690 -237.373    0.000    0.000  474.746
            2.832
RCC  76   -41.910   59.690 -237.373    0.000    0.000  474.746
            2.832
RCC  77     0.000    0.000 -221.005    0.000    0.000  442.011
           83.589
RCC  78     0.000    0.000 -215.849    0.000    0.000  431.698
           71.695
RCC  79     0.000    0.000 -198.780    0.000    0.000  397.561
           71.695
RCC  80     0.000    0.000 -182.880    0.000    0.000  365.760
           71.695
RCC  81     0.000    0.000 -221.015    0.000    0.000  442.031
           71.695
RCC  82     0.000    0.000 -215.849    0.000    0.000  431.698
           83.589
RCC  83     0.000    0.000 -198.780    0.000    0.000  397.561
           83.589
RCC  84     0.000    0.000 -182.880    0.000    0.000  365.760
           83.589
RPP  85   -25.456   25.456  -77.320   77.320 -215.849  215.849
RPP  86   -52.817   52.817  -52.817   52.817 -215.849  215.849
RPP  87   -77.320   77.320  -25.456   25.456 -215.849  215.849
RPP  88   -25.456   25.456  -77.320   77.320 -198.780  198.780
RPP  89   -52.817   52.817  -52.817   52.817 -198.780  198.780
RPP  90   -77.320   77.320  -25.456   25.456 -198.780  198.780
RPP  91   -25.456   25.456  -77.320   77.320 -182.880  182.880
RPP  92   -52.817   52.817  -52.817   52.817 -182.880  182.880
RPP  93   -77.320   77.320  -25.456   25.456 -182.880  182.880
RPP  94   -25.456   25.456  -77.320   77.320 -221.015  221.015
RPP  95   -52.817   52.817  -52.817   52.817 -221.015  221.015
RPP  96   -77.320   77.320  -25.456   25.456 -221.015  221.015
RCC  97     0.000    0.000 -204.368    0.000    0.000    3.175
           83.185
RCC  98     0.000    0.000  204.368    0.000    0.000   -3.175
           83.185
RCC  99     0.000    0.000 -204.358    0.000    0.000  -14.107
           83.185
RCC 100     0.000    0.000  204.358    0.000    0.000   14.107
           83.185
RCC 101     0.000    0.000 -204.358    0.000    0.000  -14.107
           82.233
RCC 102     0.000    0.000  204.358    0.000    0.000   14.107
           82.233
RCC 103     0.000    0.000 -189.967    0.000    0.000    1.270
           83.185
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
RCC 104    0.000    0.000 -177.470    0.000    0.000    1.270
          83.185
RCC 105    0.000    0.000 -164.973    0.000    0.000    1.270
          83.185
RCC 106    0.000    0.000 -152.476    0.000    0.000    1.270
          83.185
RCC 107    0.000    0.000 -139.979    0.000    0.000    1.270
          83.185
RCC 108    0.000    0.000 -127.483    0.000    0.000    1.270
          83.185
RCC 109    0.000    0.000 -114.986    0.000    0.000    1.270
          83.185
RCC 110    0.000    0.000 -102.489    0.000    0.000    1.270
          83.185
RCC 111    0.000    0.000  -89.992    0.000    0.000    1.270
          83.185
RCC 112    0.000    0.000  -77.495    0.000    0.000    1.270
          83.185
RCC 113    0.000    0.000  -64.999    0.000    0.000    1.270
          83.185
RCC 114    0.000    0.000  -52.502    0.000    0.000    1.270
          83.185
RCC 115    0.000    0.000  -40.005    0.000    0.000    1.270
          83.185
RCC 116    0.000    0.000  -27.508    0.000    0.000    1.270
          83.185
RCC 117    0.000    0.000  -15.011    0.000    0.000    1.270
          83.185
RCC 118    0.000    0.000   -2.515    0.000    0.000    1.270
          83.185
RCC 119    0.000    0.000  189.967    0.000    0.000   -1.270
          83.185
RCC 120    0.000    0.000  177.470    0.000    0.000   -1.270
          83.185
RCC 121    0.000    0.000  164.973    0.000    0.000   -1.270
          83.185
RCC 122    0.000    0.000  152.476    0.000    0.000   -1.270
          83.185
RCC 123    0.000    0.000  139.979    0.000    0.000   -1.270
          83.185
RCC 124    0.000    0.000  127.483    0.000    0.000   -1.270
          83.185
RCC 125    0.000    0.000  114.986    0.000    0.000   -1.270
          83.185
RCC 126    0.000    0.000  102.489    0.000    0.000   -1.270
          83.185
RCC 127    0.000    0.000   89.992    0.000    0.000   -1.270
          83.185
RCC 128    0.000    0.000   77.495    0.000    0.000   -1.270
          83.185
RCC 129    0.000    0.000   64.999    0.000    0.000   -1.270
          83.185
RCC 130    0.000    0.000   52.502    0.000    0.000   -1.270
          83.185
RCC 131    0.000    0.000   40.005    0.000    0.000   -1.270
          83.185
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
RCC 132    0.000    0.000   27.508    0.000    0.000   -1.270
          83.185
RCC 133    0.000    0.000   15.011    0.000    0.000   -1.270
          83.185
RCC 134    0.000    0.000    2.515    0.000    0.000   -1.270
          83.185
RCC 135    0.000    0.000 -183.718    0.000    0.000    1.270
          82.868
RCC 136    0.000    0.000 -171.221    0.000    0.000    1.270
          82.868
RCC 137    0.000    0.000 -158.725    0.000    0.000    1.270
          82.868
RCC 138    0.000    0.000 -146.228    0.000    0.000    1.270
          82.868
RCC 139    0.000    0.000 -133.731    0.000    0.000    1.270
          82.868
RCC 140    0.000    0.000 -121.234    0.000    0.000    1.270
          82.868
RCC 141    0.000    0.000 -108.737    0.000    0.000    1.270
          82.868
RCC 142    0.000    0.000  -96.241    0.000    0.000    1.270
          82.868
RCC 143    0.000    0.000  -83.744    0.000    0.000    1.270
          82.868
RCC 144    0.000    0.000  -71.247    0.000    0.000    1.270
          82.868
RCC 145    0.000    0.000  -58.750    0.000    0.000    1.270
          82.868
RCC 146    0.000    0.000  -46.253    0.000    0.000    1.270
          82.868
RCC 147    0.000    0.000  -33.757    0.000    0.000    1.270
          82.868
RCC 148    0.000    0.000  -21.260    0.000    0.000    1.270
          82.868
RCC 149    0.000    0.000   -8.763    0.000    0.000    1.270
          82.868
RCC 150    0.000    0.000  183.718    0.000    0.000   -1.270
          82.868
RCC 151    0.000    0.000  171.221    0.000    0.000   -1.270
          82.868
RCC 152    0.000    0.000  158.725    0.000    0.000   -1.270
          82.868
RCC 153    0.000    0.000  146.228    0.000    0.000   -1.270
          82.868
RCC 154    0.000    0.000  133.731    0.000    0.000   -1.270
          82.868
RCC 155    0.000    0.000  121.234    0.000    0.000   -1.270
          82.868
RCC 156    0.000    0.000  108.737    0.000    0.000   -1.270
          82.868
RCC 157    0.000    0.000   96.241    0.000    0.000   -1.270
          82.868
RCC 158    0.000    0.000   83.744    0.000    0.000   -1.270
          82.868
RCC 159    0.000    0.000   71.247    0.000    0.000   -1.270
          82.868
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
RCC 160    0.000    0.000   58.750    0.000    0.000   -1.270
          82.868
RCC 161    0.000    0.000   46.253    0.000    0.000   -1.270
          82.868
RCC 162    0.000    0.000   33.757    0.000    0.000   -1.270
          82.868
RCC 163    0.000    0.000   21.260    0.000    0.000   -1.270
          82.868
RCC 164    0.000    0.000    8.763    0.000    0.000   -1.270
          82.868
BOX 165  116.909    4.404 -202.108   -0.061    1.399    0.000
        -233.757  -10.206    0.000    0.000    0.000  404.216
BOX 166  111.786   34.512 -202.108   -0.421    1.335    0.000
        -223.150  -70.359    0.000    0.000    0.000  404.216
BOX 167   99.044   62.268 -202.108   -0.752    1.181    0.000
        -197.336 -125.717    0.000    0.000    0.000  404.216
BOX 168   79.553   85.781 -202.108   -1.032    0.946    0.000
        -158.074 -172.508    0.000    0.000    0.000  404.216
BOX 169   54.641  103.448 -202.108   -1.242    0.646    0.000
        -108.040 -207.542    0.000    0.000    0.000  404.216
BOX 170   26.005  114.065 -202.108   -1.367    0.303    0.000
         -50.642 -228.433    0.000    0.000    0.000  404.216
BOX 171   -4.404  116.909 -202.108   -1.399   -0.061    0.000
          10.206 -233.757    0.000    0.000    0.000  404.216
BOX 172  -34.512  111.786 -202.108   -1.335   -0.421    0.000
          70.359 -223.150    0.000    0.000    0.000  404.216
BOX 173  -62.268   99.044 -202.108   -1.181   -0.752    0.000
         125.717 -197.336    0.000    0.000    0.000  404.216
BOX 174  -85.781   79.553 -202.108   -0.946   -1.032    0.000
         172.508 -158.074    0.000    0.000    0.000  404.216
BOX 175 -103.448   54.641 -202.108   -0.646   -1.242    0.000
         207.542 -108.040    0.000    0.000    0.000  404.216
BOX 176 -114.065   26.005 -202.108   -0.303   -1.367    0.000
         228.433  -50.642    0.000    0.000    0.000  404.216
RCC 177    0.000    0.000 -202.098    0.000    0.000  404.196
         117.000
 END
? UMS Transport Cask - WE17x17 - Top - Normal - v2.8
FUE   OR  +91   OR  +92   OR  +93
UPL   OR  +88  -91   OR  +89  -92   OR  +90  -93
UEF   OR  +85  -88   OR  +86  -89   OR  +87  -90
SHL   OR  +65  +63  -66  -69  -70   OR  +65  +63  +66
      -67  -71  -72   OR  +65  +63  +67  -77  -73  -74
STL  +64  +63  -65
CSS  +62  -63
PTC   OR  +69  +65  -66   OR  +70  +65  -66
PCM   OR  +71  +66  -75  -68   OR  +72  +66  -76  -68
PMC   OR  +75  +66  -68   OR  +71  +68  -67   OR  +76
      +66  -68   OR  +72  +68  -67
PBC   OR  +73  +67  -77   OR  +74  +67  -77
TWT  +97  -94  -95  -96
TWB  +98  -94  -95  -96
ANU   OR +101  -97  -85  -86  -87   OR +102  -98  -85
      -86  -87   OR  +77  +63  -85  -86  -87  -83  -97
      -98  -99 -100 -103 -104 -105 -106 -107 -108 -109
     -110 -111 -112 -113 -114 -115 -116 -117 -118 -119
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
      -120 -121 -122 -123 -124 -125 -126 -127 -128 -129
      -130 -131 -132 -133 -134 -135 -136 -137 -138 -139
      -140 -141 -142 -143 -144 -145 -146 -147 -148 -149
      -150 -151 -152 -153 -154 -155 -156 -157 -158 -159
      -160 -161 -162 -163 -164
ANM  +77  +63  +83  -85  -86  -87  -97  -98  -99 -100
      -103 -104 -105 -106 -107 -108 -109 -110 -111 -112
      -113 -114 -115 -116 -117 -118 -119 -120 -121 -122
      -123 -124 -125 -126 -127 -128 -129 -130 -131 -132
      -133 -134 -135 -136 -137 -138 -139 -140 -141 -142
      -143 -144 -145 -146 -147 -148 -149 -150 -151 -152
      -153 -154 -155 -156 -157 -158 -159 -160 -161 -162
      -163 -164
FLT  +99 -101  -97
FLB +100 -102  -98
S01 +103  -94  -95  -96
S02 +104  -94  -95  -96
S03 +105  -94  -95  -96
S04 +106  -94  -95  -96
S05 +107  -94  -95  -96
S06 +108  -94  -95  -96
S07 +109  -94  -95  -96
S08 +110  -94  -95  -96
S09 +111  -94  -95  -96
S10 +112  -94  -95  -96
S11 +113  -94  -95  -96
S12 +114  -94  -95  -96
S13 +115  -94  -95  -96
S14 +116  -94  -95  -96
S15 +117  -94  -95  -96
S16 +118  -94  -95  -96
T01 +119  -94  -95  -96
T02 +120  -94  -95  -96
T03 +121  -94  -95  -96
T04 +122  -94  -95  -96
T05 +123  -94  -95  -96
T06 +124  -94  -95  -96
T07 +125  -94  -95  -96
T08 +126  -94  -95  -96
T09 +127  -94  -95  -96
T10 +128  -94  -95  -96
T11 +129  -94  -95  -96
T12 +130  -94  -95  -96
T13 +131  -94  -95  -96
T14 +132  -94  -95  -96
T15 +133  -94  -95  -96
T16 +134  -94  -95  -96
F01 +135  -94  -95  -96
F02 +136  -94  -95  -96
F03 +137  -94  -95  -96
F04 +138  -94  -95  -96
F05 +139  -94  -95  -96
F06 +140  -94  -95  -96
F07 +141  -94  -95  -96
F08 +142  -94  -95  -96
F09 +143  -94  -95  -96
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
F10 +144  -94  -95  -96
F11 +145  -94  -95  -96
F12 +146  -94  -95  -96
F13 +147  -94  -95  -96
F14 +148  -94  -95  -96
F15 +149  -94  -95  -96
G01 +150  -94  -95  -96
G02 +151  -94  -95  -96
G03 +152  -94  -95  -96
G04 +153  -94  -95  -96
G05 +154  -94  -95  -96
G06 +155  -94  -95  -96
G07 +156  -94  -95  -96
G08 +157  -94  -95  -96
G09 +158  -94  -95  -96
G10 +159  -94  -95  -96
G11 +160  -94  -95  -96
G12 +161  -94  -95  -96
G13 +162  -94  -95  -96
G14 +163  -94  -95  -96
G15 +164  -94  -95  -96
H01 +165  +2  -9
H02 +166  +2  -9
H03 +167  +2  -9
H04 +168  +2  -9
H05 +169  +2  -9
H06 +170  +2  -9
H07 +171  +2  -9
H08 +172  +2  -9
H09 +173  +2  -9
H10 +174  +2  -9
H11 +175  +2  -9
H12 +176  +2  -9
NSS  OR  +1  -2  -5  -6  -26  -28  -30  -32  -34
     -36  -38  -40  OR  +27  +1  -26  OR  +29  +1
     -28  OR  +31  +1  -30  OR  +33  +1  -32  OR
     +35  +1  -34  OR  +37  +1  -36  OR  +39  +1
     -38  OR  +41  +1  -40
NS4  +2  -9  -5  -6  -27  -29  -31  -33  -35  -37
     -39  -41 -165 -166 -167 -168 -169 -170 -171 -172
     -173 -174 -175 -176
OSS  OR  +9  -5  -6  -14  OR  +5  -14  OR  +6
     -14
PBG  +14  +10  -11
PBS  +11  +10  -15
ISS  +15  -16
TFG  +7  -8  -16  -22  -23
LID  OR  +18  OR  +20  -42  OR  +19  OR  +21  -43
LVN  OR  +42  +22  OR  +43  +23
LBR  OR  +22  -20  -18  OR  +23  -21  -19
CAV  +16  +17  -62
LSS  OR  +44  -45  -50  OR  +49  -50  -48  OR  +48
     -7  -50
LFW  OR  +46  -48  -49  OR  +47  -48  -49
LSW  +45  -46  -47
TRV  OR  +26  +1  OR  +28  +1  OR  +30  +1  OR
```

5.5.3.2.4 PWR Top Radial End Fitting Gamma – Accident Conditions (Continued)

```
     +32  +1  OR  +34  +1  OR  +36  +1  OR  +38
     +1  OR  +40  +1
FRV  +51  +50  -1  -7
DEP  +52  -51
DEB  OR  +53  -44  OR  +53  +50  -52
DEC  +54  -53
DER  +55  -54
DED  +56  -55
DEE  +57  -56
INV  +60  -57
EXV  +61  -60
 END
'114 Regions
    1    1    1    1    4    1    1    1    3    3
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    3    3
    1    1    1    1    3    1    2    1    2    1
    2    1    2    1
'Universes
114R0
'114 Materials
    1    3    5   12   12   12   12   12 1000 1000
   12   12 1000   19   12   12   12   12   12   12
   12   12   12   12   12   12   12   12   12   12
   12   12   12   12   12   12   12   12   12   12
   12   12   12   12   12   12   12   12   17   17
   17   17   17   17   17   17   17   17   17   17
   17   17   17   17   17   17   17   17   17   17
   17   17   17   17   17   17   17   17   20   20
   20   20   20   20   20   20   20   20   20   20
   12   14   12 1000   13   12   12   12 1000 1000
 1000   12   16   15 1000 1000 1000 1000 1000 1000
 1000 1000 1000    0
0 0
0
END
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma

```
=SAS4A     PARM='SIZE=1000000'
UMS Transport-we17x17ns_10Y_45B-Bottom-Normal-Dry-Radial-Fuel-Gamma-UMS Profile
27N-18COUPLE   INFHOMMEDIUM
' WE17x17NS Material File
' $Id: we17x17ns.matl,v 1.4 1997/07/22 22:59:14 csh Exp $
'
' Reference EA790-4009 Rev. 0 Spreadsheet pwr-mat-h2o.xls
'
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
' Mat 1 1 is fuel, 2 is basket outside of fuel radius, 3 is TSC
'
UO2         1  DEN=2.1530 1.0 END
ZIRCALLOY   1  DEN=0.4494 1.0 END
SS304       1  DEN=0.3807 1.0 END
AL          1  DEN=0.0894 1.0 END
B4C         1  DEN=0.0220 1.0 END
SS304       2  DEN=0.7691 1.0 END
AL          2  DEN=0.2544 1.0 END
'
' Begin Upper Plenum
' 4 is in fuel radius, 5 is outside of fuel radius
'
ZIRCALLOY   3  DEN=0.4494 1.0 END
SS304       3  DEN=1.0318 1.0 END
SS304       4  DEN=0.6101 1.0 END
'
'BEGIN U. EF
'
SS304       5  DEN=1.2537 1.0 END
SS304       6  DEN=1.1366 1.0 END
'
'BEGIN LOWER END FITTING
'
SS304       9  DEN=1.4554 1.0 END
SS304       10 DEN=1.7805 1.0 END
' MASTER MATERIALS LIST
' $Id: master.matl,v 1.3 1997/07/09 19:56:48 csh Exp $
'
CARBONSTEEL   11  1.0  END
SS304         12  1.0  END
PB            13  1.0  END
B-10          14  0.0  8.553-5 END
B-11          14  0.0  3.422-4 END
AL            14  0.0  7.763-3 END
H             14  0.0  5.854-2 END
O             14  0.0  2.609-2 END
C             14  0.0  2.264-2 END
N             14  0.0  1.394-3 END
BALSA         15  1.0  END
REDWOOD       16  1.0  END
AL            17  1.0  END
REG-CONCRETE  18  den=2.2426  1.0  END
N             19  1.E-6 END
CU            20  0.4286 END
SS304         20  0.5714 END
END COMP
IDR=0 ITY=2 IZM=10 ISN=8 FRD=71.695 END
' WE17x17 Bottom Radial Biasing Rev. 2.8 - Normal Conditions
' IZM=10
  71.695  83.579  85.166  85.865  90.945  97.815
  97.930 104.915 116.345 116.980 END
  1 2 12  0 12 13  0 12 14 12 END
XEND
TIM=3000.0 NST=2000 NIT=1000 ISO=0 IGO=4 RAN=F05D728AF501 SFA=9.0120e+16
ICS=11 NOD=0 IPF=8 END
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
SOE
27R0.0
' W17 45B 10 year Fuel Gamma Spectrum
1.5743E+05 7.4152E+05 3.7803E+06 9.4200E+06 4.2178E+08 3.7019E+09
5.3258E+10 1.3025E+11 8.1351E+12 7.6492E+13 1.0567E+14 1.8094E+15
1.7568E+14 3.7162E+13 6.0583E+13 2.1673E+14 2.7731E+14 9.8776E+14 END
SXY 1 -83.579 83.579 -83.579 83.579   0.000 182.880 83.579
     182.880  1.0 1.0 END
' WE17x17 Bottom Model Radial Detector Desc. Rev. 2.8 - Normal Conditions
' ICS=11
CSF 104.915 116.980 116.980 116.980 116.980 135.763
     157.480 216.980 316.980 357.480 516.980 END
CSL 193.142 196.317 193.142 196.317 160.122 193.142  24.000 160.122
       0.000  24.000   0.000 196.317 196.317 304.902   0.000 404.902
       0.000 504.902   0.000 554.902   0.000 704.902 END
CSD  1  1  1  1  3 16 16  1  1 72 20  1
      8  1 16  1 16  1 16  1 16  1 END
' PWR Bottom Model Axial Profiles  - Photon Source
' IPF=8
BUB  0.0000 128.0160 137.1600 146.3040 155.4480 164.5920
     173.7360 182.8800 END
BUF  1.07900e+00  1.08000e+00  9.91200e-01  9.02300e-01  8.13500e-01
      7.24700e-01  6.35800e-01  5.47000e-01 END
GEND
UMS Transport Cask - WE17x17 - Bot - v2.8
 0 0 1 50
RCC   1      0.000     0.000 -193.142     0.000     0.000  386.283
              116.980
RCC   2      0.000     0.000 -191.872     0.000     0.000  383.743
              116.345
RCC   3      0.000     0.000 -186.792     0.000     0.000  373.583
              116.345
RCC   4      0.000     0.000 -224.257     0.000     0.000  448.513
              104.915
RCC   5      0.000     0.000 -211.557     0.000     0.000  423.113
               97.930
RCC   6      0.000     0.000 -196.317     0.000     0.000  392.633
               97.930
RCC   7      0.000     0.000 -196.317     0.000     0.000  392.633
               97.815
RCC   8      0.000     0.000 -188.563     0.000     0.000  377.126
               97.940
BOX   9     95.503   -26.747 -201.397    12.473     0.000    0.000
             0.000     0.000  402.793     0.000    53.494    0.000
RCC  10      0.000     0.000 -211.557     0.000     0.000  423.113
               90.945
RCC  11      0.000     0.000 -209.017     0.000     0.000  418.033
               90.945
RCC  12      0.000     0.000 -198.222     0.000     0.000  396.443
               85.865
RCC  13      0.000     0.000 -198.222     0.000     0.000  396.443
               85.865
WED  14    102.375   -17.780 -160.122    16.154     0.000    0.000
             0.000     0.000   36.059     0.000    35.560    0.000
WED  15    102.375   -17.780 -141.072     7.620     0.000    0.000
             0.000     0.000   17.009     0.000    35.560    0.000
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
WED  16  -102.375  -17.780 -160.122  -16.154    0.000    0.000
            0.000    0.000   36.059    0.000   35.560    0.000
WED  17  -102.375  -17.780 -141.072   -7.620    0.000    0.000
            0.000    0.000   17.009    0.000   35.560    0.000
BOX  18   102.375  -17.780 -193.167   18.694    0.000    0.000
            0.000    0.000   33.071    0.000   35.560    0.000
BOX  19  -102.375  -17.780 -193.167  -18.694    0.000    0.000
            0.000    0.000   33.071    0.000   35.560    0.000
BOX  20   108.369  -10.160 -193.167   12.700    0.000    0.000
            0.000    0.000   16.231    0.000   20.320    0.000
BOX  21  -108.369  -10.160 -193.167  -12.700    0.000    0.000
            0.000    0.000   16.231    0.000   20.320    0.000
RCC  22   108.369    0.000 -176.936   12.700    0.000    0.000
           12.700
RCC  23  -108.369    0.000 -176.936  -12.700    0.000    0.000
           12.700
WED  24   102.375  -17.780  160.122   16.154    0.000    0.000
            0.000    0.000  -36.059    0.000   35.560    0.000
WED  25   102.375  -17.780  141.072    7.620    0.000    0.000
            0.000    0.000  -17.009    0.000   35.560    0.000
WED  26  -102.375  -17.780  160.122  -16.154    0.000    0.000
            0.000    0.000  -36.059    0.000   35.560    0.000
WED  27  -102.375  -17.780  141.072   -7.620    0.000    0.000
            0.000    0.000  -17.009    0.000   35.560    0.000
BOX  28   102.375  -17.780  193.167   18.694    0.000    0.000
            0.000    0.000  -33.071    0.000   35.560    0.000
BOX  29  -102.375  -17.780  193.167  -18.694    0.000    0.000
            0.000    0.000  -33.071    0.000   35.560    0.000
BOX  30   108.369  -10.160  193.167   12.700    0.000    0.000
            0.000    0.000  -16.231    0.000   20.320    0.000
BOX  31  -108.369  -10.160  193.167  -12.700    0.000    0.000
            0.000    0.000  -16.231    0.000   20.320    0.000
RCC  32   108.369    0.000  176.936   12.700    0.000    0.000
           12.700
RCC  33  -108.369    0.000  176.936  -12.700    0.000    0.000
           12.700
RCC  34     0.000    0.000 -304.902    0.000    0.000  609.803
          157.480
RCC  35     0.000    0.000 -304.267    0.000    0.000  608.533
          156.845
RCC  36     0.000    0.000 -300.457    0.000    0.000  600.913
          125.984
RCC  37     0.000    0.000 -245.593    0.000    0.000  491.185
          156.845
RCC  38     0.000    0.000 -224.892    0.000    0.000  449.783
          105.550
RCC  39     0.000    0.000 -196.952    0.000    0.000  393.903
          156.845
RCC  40     0.000    0.000 -196.317    0.000    0.000  392.633
          157.480
RCC  41     0.000  104.925 -201.397    0.000   -3.820    0.000
            6.706
RCC  42     0.000  104.925  201.397    0.000   -3.820    0.000
            6.706
RCC  43     0.000  101.115 -201.397    0.000   -4.988    0.000
            2.540
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
RCC 44     0.000  101.115  201.397    0.000   -4.988    0.000
           2.540
RCC 45     0.000   96.136 -201.397    0.000   -5.222    0.000
           0.495
RCC 46     0.000   96.136  201.397    0.000   -5.222    0.000
           0.495
RCC 47     0.000   93.455 -201.397    0.000   -9.933    4.155
           0.495
RCC 48     0.000   93.455  201.397    0.000   -9.933   -4.155
           0.495
RCC 49     0.000    0.000 -224.257    0.000    0.000  448.513
         116.980
RCC 50     0.000    0.000 -196.327    0.000    0.000  392.653
         135.763
RCC 51     0.000    0.000 -404.902    0.000    0.000  809.803
         216.980
RCC 52     0.000    0.000 -504.902    0.000    0.000 1009.803
         316.980
RCC 53     0.000    0.000 -554.902    0.000    0.000 1109.803
         357.480
RCC 54     0.000    0.000 -604.902    0.000    0.000 1209.803
         416.980
RCC 55     0.000    0.000 -704.902    0.000    0.000 1409.803
         516.980
RCC 56     0.000    0.000 -224.257    0.000    0.000  448.513
         104.915
RCC 57     0.000    0.000 -324.257    0.000    0.000  648.513
         204.915
RCC 58     0.000    0.000 -804.902    0.000    0.000 1609.803
         616.980
RCC 59     0.000    0.000 -904.902    0.000    0.000 1809.803
         716.980
RCC 60     0.000    0.000 -198.222    0.000    0.000  396.443
          85.166
RCC 61     0.000    0.000 -193.777    0.000    0.000  387.553
          83.579
RCC 62     0.000    0.000 -193.787    0.000    0.000  387.573
          71.695
RCC 63     0.000    0.000 -182.880    0.000    0.000  365.760
          71.695
RCC 64     0.000    0.000 -182.880    0.000    0.000  365.760
          71.695
RCC 65     0.000    0.000 -193.787    0.000    0.000  387.573
          71.695
RCC 66     0.000    0.000 -193.777    0.000    0.000  387.553
          83.589
RCC 67     0.000    0.000 -182.880    0.000    0.000  365.760
          83.589
RCC 68     0.000    0.000 -182.880    0.000    0.000  365.760
          83.589
RPP 69   -25.456   25.456  -77.320   77.320 -193.787  193.787
RPP 70   -52.817   52.817  -52.817   52.817 -193.787  193.787
RPP 71   -77.320   77.320  -25.456   25.456 -193.787  193.787
RPP 72   -25.456   25.456  -77.320   77.320 -182.880  182.880
RPP 73   -52.817   52.817  -52.817   52.817 -182.880  182.880
RPP 74   -77.320   77.320  -25.456   25.456 -182.880  182.880
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
RPP  75   -25.456   25.456  -77.320   77.320 -182.880  182.880
RPP  76   -52.817   52.817  -52.817   52.817 -182.880  182.880
RPP  77   -77.320   77.320  -25.456   25.456 -182.880  182.880
RPP  78   -25.456   25.456  -77.320   77.320 -193.787  193.787
RPP  79   -52.817   52.817  -52.817   52.817 -193.787  193.787
RPP  80   -77.320   77.320  -25.456   25.456 -193.787  193.787
RCC  81     0.000    0.000 -191.237    0.000    0.000    2.540
           83.185
RCC  82     0.000    0.000  191.237    0.000    0.000   -2.540
           83.185
RCC  83     0.000    0.000 -177.470    0.000    0.000    1.270
           83.185
RCC  84     0.000    0.000 -164.973    0.000    0.000    1.270
           83.185
RCC  85     0.000    0.000 -152.476    0.000    0.000    1.270
           83.185
RCC  86     0.000    0.000 -139.979    0.000    0.000    1.270
           83.185
RCC  87     0.000    0.000 -127.483    0.000    0.000    1.270
           83.185
RCC  88     0.000    0.000 -114.986    0.000    0.000    1.270
           83.185
RCC  89     0.000    0.000 -102.489    0.000    0.000    1.270
           83.185
RCC  90     0.000    0.000  -89.992    0.000    0.000    1.270
           83.185
RCC  91     0.000    0.000  -77.495    0.000    0.000    1.270
           83.185
RCC  92     0.000    0.000  -64.999    0.000    0.000    1.270
           83.185
RCC  93     0.000    0.000  -52.502    0.000    0.000    1.270
           83.185
RCC  94     0.000    0.000  -40.005    0.000    0.000    1.270
           83.185
RCC  95     0.000    0.000  -27.508    0.000    0.000    1.270
           83.185
RCC  96     0.000    0.000  -15.011    0.000    0.000    1.270
           83.185
RCC  97     0.000    0.000   -2.515    0.000    0.000    1.270
           83.185
RCC  98     0.000    0.000  177.470    0.000    0.000   -1.270
           83.185
RCC  99     0.000    0.000  164.973    0.000    0.000   -1.270
           83.185
RCC 100     0.000    0.000  152.476    0.000    0.000   -1.270
           83.185
RCC 101     0.000    0.000  139.979    0.000    0.000   -1.270
           83.185
RCC 102     0.000    0.000  127.483    0.000    0.000   -1.270
           83.185
RCC 103     0.000    0.000  114.986    0.000    0.000   -1.270
           83.185
RCC 104     0.000    0.000  102.489    0.000    0.000   -1.270
           83.185
RCC 105     0.000    0.000   89.992    0.000    0.000   -1.270
           83.185
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
RCC 106      0.000      0.000     77.495      0.000      0.000     -1.270
            83.185
RCC 107      0.000      0.000     64.999      0.000      0.000     -1.270
            83.185
RCC 108      0.000      0.000     52.502      0.000      0.000     -1.270
            83.185
RCC 109      0.000      0.000     40.005      0.000      0.000     -1.270
            83.185
RCC 110      0.000      0.000     27.508      0.000      0.000     -1.270
            83.185
RCC 111      0.000      0.000     15.011      0.000      0.000     -1.270
            83.185
RCC 112      0.000      0.000      2.515      0.000      0.000     -1.270
            83.185
RCC 113      0.000      0.000   -171.221      0.000      0.000      1.270
            82.868
RCC 114      0.000      0.000   -158.725      0.000      0.000      1.270
            82.868
RCC 115      0.000      0.000   -146.228      0.000      0.000      1.270
            82.868
RCC 116      0.000      0.000   -133.731      0.000      0.000      1.270
            82.868
RCC 117      0.000      0.000   -121.234      0.000      0.000      1.270
            82.868
RCC 118      0.000      0.000   -108.737      0.000      0.000      1.270
            82.868
RCC 119      0.000      0.000    -96.241      0.000      0.000      1.270
            82.868
RCC 120      0.000      0.000    -83.744      0.000      0.000      1.270
            82.868
RCC 121      0.000      0.000    -71.247      0.000      0.000      1.270
            82.868
RCC 122      0.000      0.000    -58.750      0.000      0.000      1.270
            82.868
RCC 123      0.000      0.000    -46.253      0.000      0.000      1.270
            82.868
RCC 124      0.000      0.000    -33.757      0.000      0.000      1.270
            82.868
RCC 125      0.000      0.000    -21.260      0.000      0.000      1.270
            82.868
RCC 126      0.000      0.000     -8.763      0.000      0.000      1.270
            82.868
RCC 127      0.000      0.000    171.221      0.000      0.000     -1.270
            82.868
RCC 128      0.000      0.000    158.725      0.000      0.000     -1.270
            82.868
RCC 129      0.000      0.000    146.228      0.000      0.000     -1.270
            82.868
RCC 130      0.000      0.000    133.731      0.000      0.000     -1.270
            82.868
RCC 131      0.000      0.000    121.234      0.000      0.000     -1.270
            82.868
RCC 132      0.000      0.000    108.737      0.000      0.000     -1.270
            82.868
RCC 133      0.000      0.000     96.241      0.000      0.000     -1.270
            82.868
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
RCC 134    0.000    0.000   83.744    0.000    0.000   -1.270
           82.868
RCC 135    0.000    0.000   71.247    0.000    0.000   -1.270
           82.868
RCC 136    0.000    0.000   58.750    0.000    0.000   -1.270
           82.868
RCC 137    0.000    0.000   46.253    0.000    0.000   -1.270
           82.868
RCC 138    0.000    0.000   33.757    0.000    0.000   -1.270
           82.868
RCC 139    0.000    0.000   21.260    0.000    0.000   -1.270
           82.868
RCC 140    0.000    0.000    8.763    0.000    0.000   -1.270
           82.868
BOX 141  116.909    4.404 -193.142   -0.061    1.399    0.000
        -233.757  -10.206    0.000    0.000    0.000  386.283
BOX 142  111.786   34.512 -193.142   -0.421    1.335    0.000
        -223.150  -70.359    0.000    0.000    0.000  386.283
BOX 143   99.044   62.268 -193.142   -0.752    1.181    0.000
        -197.336 -125.717    0.000    0.000    0.000  386.283
BOX 144   79.553   85.781 -193.142   -1.032    0.946    0.000
        -158.074 -172.508    0.000    0.000    0.000  386.283
BOX 145   54.641  103.448 -193.142   -1.242    0.646    0.000
        -108.040 -207.542    0.000    0.000    0.000  386.283
BOX 146   26.005  114.065 -193.142   -1.367    0.303    0.000
         -50.642 -228.433    0.000    0.000    0.000  386.283
BOX 147   -4.404  116.909 -193.142   -1.399   -0.061    0.000
          10.206 -233.757    0.000    0.000    0.000  386.283
BOX 148  -34.512  111.786 -193.142   -1.335   -0.421    0.000
          70.359 -223.150    0.000    0.000    0.000  386.283
BOX 149  -62.268   99.044 -193.142   -1.181   -0.752    0.000
         125.717 -197.336    0.000    0.000    0.000  386.283
BOX 150  -85.781   79.553 -193.142   -0.946   -1.032    0.000
         172.508 -158.074    0.000    0.000    0.000  386.283
BOX 151 -103.448   54.641 -193.142   -0.646   -1.242    0.000
         207.542 -108.040    0.000    0.000    0.000  386.283
BOX 152 -114.065   26.005 -193.142   -0.303   -1.367    0.000
         228.433  -50.642    0.000    0.000    0.000  386.283
END
* UMS Transport Cask - WE17x17 - Bot - Normal - v2.8
FUE   OR  +75   OR  +76   OR  +77
LPL   OR  +72 -75  OR  +73 -76  OR  +74 -77
LEF   OR  +69  +61 -72   OR  +70 +61 -73   OR  +71
        +61  -74
CSS  +60  -61
BWT  +82  -78  -79  -80
BWB  +81  -78  -79  -80
ANV  +61  -78  -79  -80  -82  -81  -83  -84  -85  -86
     -87  -88  -89  -90  -91  -92  -93  -94  -95  -96
     -97  -98  -99 -100 -101 -102 -103 -104 -105 -106
    -107 -108 -109 -110 -111 -112 -113 -114 -115 -116
    -117 -118 -119 -120 -121 -122 -123 -124 -125 -126
    -127 -128 -129 -130 -131 -132 -133 -134 -135 -136
    -137 -138 -139 -140
S01  +83  -78  -79  -80
S02  +84  -78  -79  -80
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
S03 +85 -78 -79 -80
S04 +86 -78 -79 -80
S05 +87 -78 -79 -80
S06 +88 -78 -79 -80
S07 +89 -78 -79 -80
S08 +90 -78 -79 -80
S09 +91 -78 -79 -80
S10 +92 -78 -79 -80
S11 +93 -78 -79 -80
S12 +94 -78 -79 -80
S13 +95 -78 -79 -80
S14 +96 -78 -79 -80
S15 +97 -78 -79 -80
T01 +98 -78 -79 -80
T02 +99 -78 -79 -80
T03 +100 -78 -79 -80
T04 +101 -78 -79 -80
T05 +102 -78 -79 -80
T06 +103 -78 -79 -80
T07 +104 -78 -79 -80
T08 +105 -78 -79 -80
T09 +106 -78 -79 -80
T10 +107 -78 -79 -80
T11 +108 -78 -79 -80
T12 +109 -78 -79 -80
T13 +110 -78 -79 -80
T14 +111 -78 -79 -80
T15 +112 -78 -79 -80
F01 +113 -78 -79 -80
F02 +114 -78 -79 -80
F03 +115 -78 -79 -80
F04 +116 -78 -79 -80
F05 +117 -78 -79 -80
F06 +118 -78 -79 -80
F07 +119 -78 -79 -80
F08 +120 -78 -79 -80
F09 +121 -78 -79 -80
F10 +122 -78 -79 -80
F11 +123 -78 -79 -80
F12 +124 -78 -79 -80
F13 +125 -78 -79 -80
F14 +126 -78 -79 -80
G01 +127 -78 -79 -80
G02 +128 -78 -79 -80
G03 +129 -78 -79 -80
G04 +130 -78 -79 -80
G05 +131 -78 -79 -80
G06 +132 -78 -79 -80
G07 +133 -78 -79 -80
G08 +134 -78 -79 -80
G09 +135 -78 -79 -80
G10 +136 -78 -79 -80
G11 +137 -78 -79 -80
G12 +138 -78 -79 -80
G13 +139 -78 -79 -80
G14 +140 -78 -79 -80
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
H01 +141   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H02 +142   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H03 +143   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H04 +144   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H05 +145   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H06 +146   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H07 +147   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H08 +148   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H09 +149   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H10 +150   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H11 +151   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
H12 +152   +3   -4  -14  -16  -24  -26  -18  -19  -28
           -29
NSS   +1   -2   -4  -18  -19  -28  -29
NS4   OR   +3   -4  -14  -16  -18  -19  -24  -26  -28
           -29 -141 -142 -143 -144 -145 -146 -147 -148 -149
          -150 -151 -152   OR  +15   -4   OR  +17   -4   OR
           +25   -4   OR  +27   -4
NSE   +2   -3   -4  -18  -19  -28  -29
OSS   +4   -5  -41  -42  -43  -44  -45  -46  -47  -48
PBG   +6   -7
PBS   +7  -11
BRG   +5   -6  -10  -41  -42  -43  -44  -45  -46  -47
           -48
ISS  +11  -12  -41  -42  -43  -44  -45  -46  -47  -48
AXN  +10  -11
SPC  +12  -13
CAV  +13  -60
PL1   OR  +14   +1   -4  -15   OR  +18   +1   -4  -20
           -22
PL2   OR  +16   +1   -4  -17   OR  +19   +1   -4  -21
           -23
PU1   OR  +24   +1   -4  -25   OR  +28   +1   -4  -30
           -32
PU2   OR  +26   +1   -4  -27   OR  +29   +1   -4  -31
           -33
LSS   OR  +34  -35  -40   OR  +39  -40  -38   OR  +38
            -4  -40
LFW   OR  +36  -38  -39   OR  +37  -38  -39
LSW  +35  -36  -37
VL1   OR  +22   +1   OR  +20   +1
VL2   OR  +23   +1   OR  +21   +1
VU1   OR  +32   +1   OR  +30   +1
VU2   OR  +33   +1   OR  +31   +1
LGP  +49  +40   -1   -4
```

5.5.3.2.5 Heat Fin Evaluation - Fuel Gamma (Continued)

```
VET   OR  +43  -41   OR  +45  -43   OR  +47  -45  -12
VTC  +41  +4
VEB   OR  +44  -42   OR  +46  -44   OR  +48  -46  -12
VBC  +42  +4
DEP  +50  +40  -49
DEB   OR  +51  -34   OR  +51  +40  -50
DEC  +52  -51
DER  +53  -52
DED  +54  -53
DEE  +55  -54
INV  +58  -55
EXV  +59  -58
 END
'112 Regions
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    1    1    1    1    1
    1    1    1    1    1    3    3    3    3    4
    3    3    3    3    2    1    2    1    2    1
    2    1
'Universes
112R0
'112 Materials
    1    7    9   12   12   12   19   12   12   12
   12   12   12   12   12   12   12   12   12   12
   12   12   12   12   12   12   12   12   12   12
   12   12   12   12   12   12   12   17   17   17
   17   17   17   17   17   17   17   17   17   17
   17   17   17   17   17   17   17   17   17   17
   17   17   17   17   17   20   20   20   20   20
   20   20   20   20   20   20   20   12   14 1000
   12 1000   13   12   12   14 1000 1000   12   12
   12   12   12   16   15 1000 1000 1000 1000 1000
 1000   12 1000   12 1000 1000 1000 1000 1000 1000
 1000    0
0 0
0
END
```

5.6 References

1. 10 CFR 71, "Packaging and Transportation of Radioactive Materials," Part 71, Title 10 of the Code of Federal Regulations, 1996.

2. J. S. Tang, "SAS4: A Monte Carlo Cask Shielding Analysis Module Using an Automated Biasing Procedure," ORNL/NUREG/CSD-2/V1/R5, Volume 1, Section S4, September 1995.

3. Oak Ridge National Laboratory, "SCALE 4.3: Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," ORNL/NUREG/CSD-2, September 1995.

4. O. W. Hermann and C. V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," ORNL/NUREG/CSD-2/V1/R5, Volume 1, Section S2, September 1995.

5. J. R. Knight et al., "SAS1: A One-Dimensional Shielding Analysis Module," ORNL/NUREG/CSD-2/V1/R5, Volume 1, Section S1, September 1995.

6. W. C. Jordan, "Scale Cross-Section Libraries," ORNL/NUREG/CSD-2/V3/R5, Section M4, September 1995.

7. N. M. Greene and L. M. Petrie, "XSDRNPM: A One-Dimensional Discrete Ordinates Codes for Transport Analysis," ORNL/NUREG/CSD-2/V1/R5, Volume 1, Section S1, September 1995.

8. O. W. Hermann and R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," ORNL/NUREG/CSD-2/V2/R5, Volume 2, Part 1, Section F7, September 1995.

9. U. S. Department of Energy, "Characteristics of Spent Fuel High-Level Waste and Other Radioactive Wastes Which May Require Long Term Isolation," DOE/RW-0184, December 1987.

10. Bucholz, J. A., *et al.*, "Standard Composition Library," ORNL/NUREG/CSD-2/V2/R5, Volume 3, Section M8, September 1995.

11. J. T. West, T. J. Hoffman, and M. B. Emmett, "MORSE-SGC for the SCALE System," ORNL/NUREG/CSD-2/V2/R5, Volume 2, Part 2, Section F9, September 1995.

12. R. Cacciapouti and S. VanVolkinburg, "Axial Burnup Profile Database for Combustion Engineering 14x14 Fuel Design," YAEC-1918, Yankee Atomic Electric Company, September 1995.

13. R. Cacciapouti and S. VanVolkinburg, "Axial Burnup Profile Database for Pressurized Water Reactors," YAEC-1937, Yankee Atomic Electric Company, August 1996.

14. R. Cacciapouti, S. VanVolkinburg, and L. Hassler, "PWR Axial Burnup Profile Database," Proceedings of High Level Waste Meeting, 1994, p739.

15. EPRI NP-6191, "Testing and Analyses of the TN-24P PWR Spent Fuel Dry Storage Cask Loaded with Consolidated Fuel," February 1989.

16. PNL-6049 Vol. III, "COBRA-SFS: A Thermal-Hydraulic Analysis Computer Code," December 1986.

17. Washington Public Power Supply System, Washington 2, "Preliminary Discharged Assembly Exposure Distributions," RFP C-3/400/Addendum 03 Q-R No.5