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Replacement Pages and Drawings for the Standardized NUHOMS® System UFSAR, Revision 18 (Public Version)

NUH-003
Revision *18*
NUH003.0103

UPDATED FINAL SAFETY ANALYSIS REPORT FOR THE STANDARDIZED NUHOMS® HORIZONTAL MODULAR STORAGE SYSTEM FOR IRRADIATED NUCLEAR FUEL

By
TN Americas LLC[1]
Columbia, MD

January 2019

[1] TN Americas LLC, formerly AREVA TN, Transnuclear, Inc. (herein referred to as AREVA TN, Transnuclear, Inc., Transnuclear, or TN)

REVISION LOG

UFSAR Revision	Date	Record of Changes/FCNs	Changed Pages
9	2/3/06	FCNs 721004-045, 074, 171 R1, 172, 188, 218, 206 R1, 250, 276 R1, 294 R1, 301 R1, 309 R1, 313 R1, 317, 345, 373 and 377.	See List of Effective Pages
10	2/1/08	FCNs 721004-321 R2, 338 R1, 340, 347 R1, 352, 370, 401, 406, 410, 416, 417, 418, 428, 435, 437, 439, 445, 446, 447, 461, 467, 476 R1, 484, 494, 498, 509, 525, 528, 543	See List of Effective Pages
11	2/1/10	FCNs 721004-429 R0, 451 R0, 452 R0, 466 R0, 470 R0, 477 R0, 479 R0, 514 R0, 533 R0, 537 R0, 543 R1, 557 R0, 558 R0, 559 R1, 561 R0, 562 R0, 563 R1, 564 R2, 568 R0, 570 R0, 577 R0, 579 R0, 592 R0, 602 R0, 614 R0, 621 R0, 637 R0, 649 R1, 651 R0, 655 R0, 664 R0, 678 R1, 681 R0 ,687 R0, 688 R2, 699 R0, 705 R0, 707 R0, 708 R0, 711 R0, 715 R0, 724 R0, 732 R1, 733 R0, 738 R0, 746 R0, 753 R0, 756 R2, 788 R0, 791 R1, 793 R0, 795 R0, 798 R0, 803 R0	See changed areas of List of Effective Pages
12	1/26/12	FCNs 721004-632 R0, 665 R0, 786 R0, 797 R0, 809 R0, 814 R0, 815 R1, 821 R0, 822 R0, 828 R0, 830 R0, 831 R0, 844 R0, 847 R0, 848 R0, 851 R0, 854 R0, 856 R0, 861 R0, 866 R0, 868 R0, 872 R0, 875 R0, 883 R0, 907 R2, 908 R1, 910 R0, 911 R0, 912 R0, 923 R1, 926 R0, 955 R0, 971 R0, 972 R0, 973 R0, 984 R0, 1000 R0, 1029 R0, 1036 R0; Note that FCN 721004-951 R0 was approved on 1/20/12 and is not included.	See changed areas of List of Effective Pages

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14	8/29/14	FCNs: 721004-1073 R2, 1079 R1, 1098 R0, 1112 R0, 1117 R0, 1139 R0, 1198 R0, 1242 R0, 1258 R0, 1263 R0, 1284 R0, 1288 R0, 1295 R0, 1297 R0, 1310 R0, 1321 R0, 1334 R0, 1361 R0	See changed areas of List of Effective Pages
15	8/29/16	FCNs: 721004-817 R0,1067 R1, 1112 R1, 1295 R1, 1308 R0, 1346 R0, 1358 R0, 1381 R1, 1384 R1, 1392 R0, 1396 R0, 1397 R0, 1401 R0, 1403 R0, 1404 R0, 1412 R0, 1413 R0, 1414 R0, 1416 R1, 1424 R0, 1425 R0, 1431 R0, 1432 R1, 1437 R0, 1441 R1, 1451 R0, 1463 R0, 1466 R0, 1470 R0, 1471 R0, 1472 R0, 1484 R0, 1488 R0, 1490 R0, 1491 R0, 1496 R0, 1500 R0, 1504 R0, 1528 R0, 1532 R0; 1543 R0, 1546 R0	See changed areas of List of Effective Pages
16	7/27/17	FCNs: 1432 R2, 721004-1437 R1, 1507 R0, 1532 R1, 1545 R0, 1556 R0, 1575 R0, 1576 R0, 1581 R0, 1584 R0, 1586 R0, 1593 R0, 1594 R0, 1601 R0, 1602 R0, 1606 R0, 1612 R1, 1623 R0, 1624 R1, 1626 R0,1627 R0, 1630 R0, 1631 R0, 1634 R0, 1644 R0, 1645 R0	See changed areas of List of Effective Pages
17	3/11/18	FCNs:721004-1557 R0, 1666 R0, 1671 R0, 1675 R0, 1680 R0, 1681 R0, 1683 R0, 1686 R0	See changed areas of List of Effective Pages

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18	*1/22/19*	*FCNs:721004-1678 R0, 1712 R0, 1718 R0, 1729, R0, 1730 R0, 1734 R0*	*See changed areas of List of Effective Pages. Note that the following pages are removed in Revision 18:* • *M.3.1-8a* • *N.2-9a* • *U.2-28a* • *Z.5-6a*

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7.7-1	6	October 2001
7.7-2	7	November 2003
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8.2-21	13	January 2014

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9.6-1	16	July 2017
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10-19	14	September 2014
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12.6-2	17	March 2018
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A.3	6	October 2001
A.4	7	November 2003
A.5	7	November 2003
A.6	6	October 2001
A.7	6	October 2001
A.8	6	October 2001
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C6-2	12	February 2012
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D.1-9	6	October 2001
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E.1-2	11	February 2010
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DWG (sh. 1 of 2) NUH-03-1022-SAR	5	1/8/14
DWG (sh. 2 of 2) NUH-03-1022-SAR	5	Not shown
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DWG (sh. 3 of 3) NUH-03-1023-SAR	8	Not shown
E.1-3	11	February 2010
DWG (sh. 1 of 1) NUH-03-1029-SAR	6	1/8/14
DWG (sh. 1 of 2) NUH-03-1030-SAR	5	1/8/14
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DWG (sh. 3 of 3) NUH-03-1031-SAR	8	Not shown
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DWG (sh. 3 of 3) NUH-03-1032-SAR	7	Not shown
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DWG (sh. 1 of 3) NUH-03-1050-SAR	3	Not shown
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DWG (sh. 3 of 3) NUH-03-1050-SAR	3	Not shown
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DWG (sh. 1 of 2) NUH-03-6010-SAR	5	1/8/14
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E.3-2	11	February 2010
DWG (sh. 1 of 1) NUH-03-8000-SAR	5	1/28/10
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2 of 5 of BGE001.0024.03	None	Not shown
3 of 5 of BGE001.0024.03	None	Not shown
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6 of 5 of BGE001.0024.03	None	Not shown
7 of 5 of BGE001.0024.03	None	Not shown
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H.2	17	March 2018
H.2a	17	March 2018
H.3	6	October 2001
H.4	6	October 2001
H.5	6	October 2001
H.6	6	October 2001

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H.11	6	October 2001
H.12	6	October 2001
H.13	6	October 2001
I.0	6	October 2001
I-1	6	October 2001
"Appendix J"	6	October 2001
J.1-1	17	March 2018
J.2-1	6	October 2001
J.3-1	6	October 2001
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J.4-2	7	November 2003
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J.9-1	6	October 2001
J.10-1	6	October 2001
J.11-1	13	January 2014
J.12-1	6	October 2001
J.13-1	6	October 2001
J.14-1	6	October 2001
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ii	16	July 2017
iii	16	July 2017
iv	16	July 2017
v	16	July 2017
vi	16	July 2017
vii	16	July 2017
viii	16	July 2017
ix	16	July 2017
x	16	July 2017
xi	16	July 2017
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K.1-2	14	September 2014
K.1-3	8	June 2004

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K.1-8	8	June 2004
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DWG (sh. 1 of 2) NUH-61B-1061-SAR	5	1/8/14
DWG (sh. 2 of 2) NUH-61B-1061-SAR	5	Not shown
DWG (sh. 1 of 2) NUH-61B-1062-SAR	6	1/8/14
DWG (sh. 2 of 2) NUH-61B-1062-SAR	6	Not shown
DWG (sh. 1 of 1) NUH-61B-1063-SAR	4	1/8/14
DWG (sh. 1 of 2) NUH-61B-1064-SAR	*8*	*01/10/19*
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DWG (sh. 1 of 3) NUH-61B-1066-SAR	7	7/17/17
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K.2-3	14	September 2014
K.2-4	14	September 2014
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K.2-6	8	June 2004
K.2-7	17	March 2018
K.2-7a	17	March 2018
K.2-8	13	January 2014
K.2-9	13	January 2014
K.2-10	17	March 2018
K.2-11	8	June 2004
K.2-12	14	September 2014
K.2-13	*18*	*January 2019*
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K.2-15	13	January 2014
K.2-16	13	January 2014
K.2-17	10	February 2008
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K.2-24	11	February 2010
K.2-25	14	September 2014
K.2-26	14	September 2014

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K.3.6-19a	*18*	*January 2019*
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K.8-16	14	September 2014
K.8-17	13	January 2014
K.8-17a	13	January 2014
K.8-18	14	September 2014
K.8-19	14	September 2014
K.8-20	*18*	*January 2019*
K.8-21	8	June 2004
K.8-22	8	June 2004
K.8-23	8	June 2004
K.8-24	8	June 2004
K.8-25	8	June 2004
K.8-26	13	January 2014
K.8-27	13	January 2014
K.8-28	13	January 2014
K.8-29	13	January 2014
K.8-30	13	January 2014
K.8-31	13	January 2014
K.8-32	16	July 2017
K.9 Introduction-1	*18*	*January 2019*
K.9-1 (associated with UFSAR Rev. 11)	8	June 2004
K.9-2 (associated with UFSAR Rev. 11)	11	February 2010
K.9-3 (associated with UFSAR Rev. 11)	11	February 2010
K.9-4 (associated with UFSAR Rev. 11)	11	February 2010

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K.9-8 (associated with UFSAR Rev. 11)	11	February 2010
K.9-9 (associated with UFSAR Rev. 11)	11	February 2010
K.9-10 (associated with UFSAR Rev. 11)	11	February 2010
K.9-11 (associated with UFSAR Rev. 11)	11	February 2010
K.9.12 (associated with UFSAR Rev. 11)	11	February 2010
K.9.13 (associated with UFSAR Rev. 11)	11	February 2010
K.9.14 (associated with UFSAR Rev. 11)	11	February 2010
K.9.15 (associated with UFSAR Rev. 11)	11	February 2010
K.9-1 (associated with UFSAR Rev. 12)	8	June 2004
K.9-2 (associated with UFSAR Rev. 12)	11	February 2010
K.9-3 (associated with UFSAR Rev. 12)	11	February 2010
K.9-4 (associated with UFSAR Rev. 12)	11	February 2010
K.9-5 (associated with UFSAR Rev. 12)	11	February 2010
K.9-6 (associated with UFSAR Rev. 12)	11	February 2010
K.9-7 (associated with UFSAR Rev. 12)	11	February 2010
K.9-8 (associated with UFSAR Rev. 12)	11	February 2010
K.9-9 (associated with UFSAR Rev. 12)	11	February 2010
K.9-10 (associated with UFSAR Rev. 12)	11	February 2010
K.9-11 (associated with UFSAR Rev. 12)	11	February 2010
K.9.12 (associated with UFSAR Rev. 12)	11	February 2010
K.9.13 (associated with UFSAR Rev. 12)	11	February 2010
K.9.14 (associated with UFSAR Rev. 12)	11	February 2010
K.9.15 (associated with UFSAR Rev. 12)	12	February 2012
K.9-1 (associated with UFSAR Rev. 13)	13	January 2014
K.9-2 (associated with UFSAR Rev. 13)	13	January 2014
K.9-3 (associated with UFSAR Rev. 13)	13	January 2014
K.9-4 (associated with UFSAR Rev. 13)	13	January 2014
K.9-5 (associated with UFSAR Rev. 13)	13	January 2014
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K.9-8 (associated with UFSAR Rev. 13)	13	January 2014
K.9-9 (associated with UFSAR Rev. 13)	13	January 2014
K.9-10 (associated with UFSAR Rev. 13)	13	January 2014
K.9-11 (associated with UFSAR Rev. 13)	11	February 2010
K.9-12 (associated with UFSAR Rev. 13)	13	January 2014
K.9-13 (associated with UFSAR Rev. 13)	13	January 2014
K.9-14 (associated with UFSAR Rev. 13)	13	January 2014
K.9-15 (associated with UFSAR Rev. 13)	13	January 2014
K.9-1 (associated with UFSAR Rev. 14)	13	January 2014
K.9-2 (associated with UFSAR Rev. 14)	13	January 2014
K.9-3 (associated with UFSAR Rev. 14)	14	September 2014
K.9-4 (associated with UFSAR Rev. 14)	14	September 2014
K.9-5 (associated with UFSAR Rev. 14)	14	September 2014
K.9-5a (associated with UFSAR Rev. 14)	14	September 2014
K.9-6 (associated with UFSAR Rev. 14)	14	September 2014
K.9-7 (associated with UFSAR Rev. 14)	14	September 2014
K.9-8 (associated with UFSAR Rev. 14)	14	September 2014
K.9-9 (associated with UFSAR Rev. 14)	14	September 2014
K.9-10 (associated with UFSAR Rev. 14)	14	September 2014
K.9-11 (associated with UFSAR Rev. 14)	11	February 2010
K.9-12 (associated with UFSAR Rev. 14)	13	January 2014
K.9-13 (associated with UFSAR Rev. 14)	13	January 2014
K.9-14 (associated with UFSAR Rev. 14)	13	January 2014
K.9-15 (associated with UFSAR Rev. 14)	13	January 2014
K.9-1 (associated with UFSAR Rev. 15)	13	January 2014
K.9-2 (associated with UFSAR Rev. 15)	15	August 2016
K.9-3 (associated with UFSAR Rev. 15)	14	September 2014
K.9-4 (associated with UFSAR Rev. 15)	14	September 2014
K.9-5 (associated with UFSAR Rev. 15)	14	September 2014
K.9-5a (associated with UFSAR Rev. 15)	14	September 2014
K.9-6 (associated with UFSAR Rev. 15)	14	September 2014

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K.9-10 (associated with UFSAR Rev. 15)	14	September 2014
K.9-11 (associated with UFSAR Rev. 15)	11	February 2010
K.9-12 (associated with UFSAR Rev. 15)	13	January 2014
K.9-13 (associated with UFSAR Rev. 15)	13	January 2014
K.9-14 (associated with UFSAR Rev. 15)	13	January 2014
K.9-15 (associated with UFSAR Rev. 15)	13	January 2014
K.9-1 (associated with UFSAR Rev. 16)	13	January 2014
K.9-2 (associated with UFSAR Rev. 16)	15	August 2016
K.9-3 (associated with UFSAR Rev. 16)	14	September 2014
K.9-4 (associated with UFSAR Rev. 16)	14	September 2014
K.9-5 (associated with UFSAR Rev. 16)	16	July 2017
K.9-5a (associated with UFSAR Rev. 16)	16	July 2017
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K.9-7 (associated with UFSAR Rev. 16)	16	July 2017
K.9-7a (associated with UFSAR Rev. 16)	16	July 2017
K.9-7b (associated with UFSAR Rev. 16)	16	July 2017
K.9-8 (associated with UFSAR Rev. 16)	14	September 2014
K.9-9 (associated with UFSAR Rev. 16)	16	July 2017
K.9-10 (associated with UFSAR Rev. 16)	14	September 2014
K.9-11 (associated with UFSAR Rev. 16)	11	February 2010
K.9-12 (associated with UFSAR Rev. 16)	13	January 2014
K.9-13 (associated with UFSAR Rev. 16)	13	January 2014
K.9-14 (associated with UFSAR Rev. 16)	13	January 2014
K.9-15 (associated with UFSAR Rev. 16)	13	January 2014
K.9-1 (associated with UFSAR Rev. 17)	13	January 2014
K.9-2 (associated with UFSAR Rev. 17)	15	August 2016
K.9-3 (associated with UFSAR Rev. 17)	14	September 2014
K.9-4 (associated with UFSAR Rev. 17)	14	September 2014
K.9-5 (associated with UFSAR Rev. 17)	16	July 2017

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K.9-7a (associated with UFSAR Rev. 17)	16	July 2017
K.9-7b (associated with UFSAR Rev. 17)	16	July 2017
K.9-8 (associated with UFSAR Rev. 17)	14	September 2014
K.9-9 (associated with UFSAR Rev. 17)	16	July 2017
K.9-10 (associated with UFSAR Rev. 17)	14	September 2014
K.9-11 (associated with UFSAR Rev. 17)	11	February 2010
K.9-12 (associated with UFSAR Rev. 17)	13	January 2014
K.9-13 (associated with UFSAR Rev. 17)	13	January 2014
K.9-14 (associated with UFSAR Rev. 17)	13	January 2014
K.9-15 (associated with UFSAR Rev. 17)	13	January 2014
K.9-1 (associated with UFSAR Rev. 18)	*13*	*January 2014*
K.9-2 (associated with UFSAR Rev. 18)	*15*	*August 2016*
K.9-3 (associated with UFSAR Rev. 18)	*14*	*September 2014*
K.9-4 (associated with UFSAR Rev. 18)	*14*	*September 2014*
K.9-5 (associated with UFSAR Rev. 18)	*16*	*July 2017*
K.9-5a (associated with UFSAR Rev. 18)	*16*	*July 2017*
K.9-6 (associated with UFSAR Rev. 18)	*14*	*September 2014*
K.9-7 (associated with UFSAR Rev. 18)	*16*	*July 2017*
K.9-7a (associated with UFSAR Rev. 18)	*16*	*July 2017*
K.9-7b (associated with UFSAR Rev. 18)	*16*	*July 2017*
K.9-8 (associated with UFSAR Rev. 18)	*14*	*September 2014*
K.9-9 (associated with UFSAR Rev. 18)	*16*	*July 2017*
K.9-10 (associated with UFSAR Rev. 18)	*14*	*September 2014*
K.9-11 (associated with UFSAR Rev. 18)	*11*	*February 2010*
K.9-12 (associated with UFSAR Rev. 18)	*13*	*January 2014*
K.9-13 (associated with UFSAR Rev. 18)	*13*	*January 2014*
K.9-14 (associated with UFSAR Rev. 18)	*13*	*January 2014*
K.9-15 (associated with UFSAR Rev. 18)	*13*	*January 2014*
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K.10-2	8	June 2004
K.10-3	8	June 2004
K.10-4	12	February 2012
K.10-5	8	June 2004
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K.10.11	8	June 2004
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K.11-9a	14	September 2014
K.11-10	8	June 2004
K.11-11	14	September 2014
K.11.12	14	September 2014
K.11.13	8	June 2004
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K.11.16	8	June 2004
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K.13-1	8	June 2004
K.14-1	8	June 2004
"Appendix L"	7	November 2003
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L.1-1	17	March 2018
L.1-2	6	October 2001
L.1-3	6	October 2001
L.1-4	13	January 2014
L.1-5	6	October 2001
L.1-6	7	November 2003
L.1-7	13	January 2014
L.1-8	7	November 2003
DWG (sh. 1 of 4) NUH-03-1070-SAR	2	01/28/10
DWG (sh. 2 of 4) NUH-03-1070-SAR	2	Not shown
DWG (sh. 3 of 4) NUH-03-1070-SAR	2	Not shown
DWG (sh. 4 of 4) NUH-03-1070-SAR	2	Not shown
DWG (sh. 1 of 4) NUH-03-1071-SAR	1	Not shown
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L.1-11	6	October 2001
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L.2-2	13	January 2014
L.2-3	13	January 2014
L.2-4	17	March 2018
L.2-5	13	January 2014
L.2-6	13	January 2014
L.2-7	6	October 2001
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L.3-2	6	October 2001
L.3-3	6	October 2001
L.3-4	9	January 2006
L.3-5	9	January 2006
L.3-6	6	October 2001
L.3-7	6	October 2001
L.3-8	6	October 2001
L.3-9	6	October 2001
L.3-10	6	October 2001
L.3-11	6	October 2001
L.3-12	6	October 2001
L.3-13	13	January 2014
L.3-14	6	October 2001
L.3-15	12	February 2012
L.3-16	6	October 2001
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L.3-31	6	October 2001
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L.3-42	6	October 2001
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L.4-1	6	October 2001
L.4-2	6	October 2001
L.4-3	13	January 2014
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L.4-8	6	October 2001
L.4-9	6	October 2001
L.4-10	6	October 2001
L.4-11	6	October 2001
L.4-12	6	October 2001
L.4-13	6	October 2001
L.4-14	6	October 2001
L.4-15	6	October 2001
L.4-16	6	October 2001
L.4-17	6	October 2001
L.4-18	6	October 2001
L.4-19	13	January 2014
L.4-20	13	January 2014
L.4-21	6	October 2001
L.4-22	13	January 2014
L.4-23	13	January 2014
L.4-24	6	October 2001
L.4-25	6	October 2001
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L.4-27	6	October 2001
L.4-28	6	October 2001
L.4-29	6	October 2001
L.4-30	6	October 2001
L.4-31	6	October 2001
L.4-32	6	October 2001
L.4-33	13	January 2014
L.4-34	6	October 2001
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L.6-3	6	October 2001
L.6-4	6	October 2001
L.6-5	6	October 2001
L.6-6	6	October 2001
L.6-7	6	October 2001
L.6-8	6	October 2001
L.6-9	6	October 2001
L.6-10	6	October 2001
L.6-11	6	October 2001
L.6-12	6	October 2001
L.6-13	6	October 2001
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L.9-5	13	January 2014
L.9-6	6	October 2001
L.10.1	6	October 2001
L.11-1	13	January 2014
L.11-2	13	January 2014
L.11-3	13	January 2014
L.11-4	13	January 2014
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L.11.8	6	October 2001
L.12-1	6	October 2001
L.13-1	6	October 2001
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M-i	*18*	*January 2019*
M-ii	*18*	*January 2019*
M-iii	*18*	*January 2019*
M-iv	*18*	*January 2019*
M-v	*18*	*January 2019*
M-vi	*18*	*January 2019*
M-vii	*18*	*January 2019*
M-viii	*18*	*January 2019*
M-ix	*18*	*January 2019*
M-x	*18*	*January 2019*
M-xi	*18*	*January 2019*
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M.1-3	*18*	*January 2019*
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DWG (sh. 2 of 2) NUH-32PT-1002-SAR	5	Not shown
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M.1-11	9	January 2006
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M.2-28	11	February 2010
M.2-29	9	January 2006
M.2-30	14	September 2014
M.2-31	14	September 2014
M.2-32	9	January 2006
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M.2-43	9	January 2006
M.2-44	*18*	*January 2019*
M.2-45	*18*	*January 2019*
M.2-46	*18*	*January 2019*
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M.3.1-8	*18*	*January 2019*
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M.3.4-19	8	June 2004
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M.8-19	*18*	*January 2019*
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M.9-4 (associated with UFSAR Rev. 11)	11	February 2010
M.9-5 (associated with UFSAR Rev. 11)	11	February 2010
M.9-6 (associated with UFSAR Rev. 11)	11	February 2010
M.9-7 (associated with UFSAR Rev. 11)	11	February 2010
M.9-8 (associated with UFSAR Rev. 11)	11	February 2010
M.9-9 (associated with UFSAR Rev. 11)	11	February 2010
M.9-10 (associated with UFSAR Rev. 11)	11	February 2010
M.9-11 (associated with UFSAR Rev. 11)	11	February 2010
M.9-12 (associated with UFSAR Rev. 11)	11	February 2010
M.9-1 (associated with UFSAR Rev. 12)	8	June 2004
M.9-2 (associated with UFSAR Rev. 12)	11	February 2010
M.9-3 (associated with UFSAR Rev. 12)	11	February 2010
M.9-4 (associated with UFSAR Rev. 12)	11	February 2010
M.9-5 (associated with UFSAR Rev. 12)	11	February 2010
M.9-6 (associated with UFSAR Rev. 12)	11	February 2010
M.9-7 (associated with UFSAR Rev. 12)	11	February 2010
M.9-8 (associated with UFSAR Rev. 12)	11	February 2010
M.9-9 (associated with UFSAR Rev. 12)	11	February 2010
M.9-10 (associated with UFSAR Rev. 12)	11	February 2010
M.9-11 (associated with UFSAR Rev. 12)	11	February 2010
M.9-12 (associated with UFSAR Rev. 12)	11	February 2010
M.9-1 (associated with UFSAR Rev. 13)	13	January 2014
M.9-2 (associated with UFSAR Rev. 13)	13	January 2014
M.9-3 (associated with UFSAR Rev. 13)	13	January 2014
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M.9-11 (associated with UFSAR Rev. 13)	13	January 2014
M.9-12 (associated with UFSAR Rev. 13)	13	January 2014
M.9-13 (associated with UFSAR Rev. 13)	13	January 2014
M.9-14 (associated with UFSAR Rev. 13)	13	January 2014
M.9-1 (associated with UFSAR Rev. 14)	13	January 2014
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M.9-3 (associated with UFSAR Rev. 14)	14	September 2014
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M.9-5 (associated with UFSAR Rev. 14)	13	January 2014
M.9-6 (associated with UFSAR Rev. 14)	14	September 2014
M.9-7 (associated with UFSAR Rev. 14)	14	September 2014
M.9-8 (associated with UFSAR Rev. 14)	14	September 2014
M.9-9 (associated with UFSAR Rev. 14)	14	September 2014
M.9-10 (associated with UFSAR Rev. 14)	14	September 2014
M.9-11 (associated with UFSAR Rev. 14)	13	January 2014
M.9-12 (associated with UFSAR Rev. 14)	13	January 2014
M.9-13 (associated with UFSAR Rev. 14)	13	January 2014
M.9-14 (associated with UFSAR Rev. 14)	13	January 2014
M.9-1 (associated with UFSAR Rev. 15)	13	January 2014
M.9-2 (associated with UFSAR Rev. 15)	15	August 2016
M.9-3 (associated with UFSAR Rev. 15)	14	September 2014
M.9-4 (associated with UFSAR Rev. 15)	14	September 2014
M.9-5 (associated with UFSAR Rev. 15)	13	January 2014
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M.9-7 (associated with UFSAR Rev. 15)	14	September 2014
M.9-8 (associated with UFSAR Rev. 15)	14	September 2014
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M.9-13 (associated with UFSAR Rev. 15)	13	January 2014
M.9-14 (associated with UFSAR Rev. 15)	13	January 2014
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M.9-7 (associated with UFSAR Rev. 16)	14	September 2014
M.9-8 (associated with UFSAR Rev. 16)	14	September 2014
M.9-9 (associated with UFSAR Rev. 16)	16	July 2017
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M.9-12 (associated with UFSAR Rev. 16)	13	January 2014
M.9-13 (associated with UFSAR Rev. 16)	13	January 2014
M.9-14 (associated with UFSAR Rev. 16)	13	January 2014
M.9-1 (associated with UFSAR Rev. 17)	13	January 2014
M.9-2 (associated with UFSAR Rev. 17)	15	August 2016
M.9-3 (associated with UFSAR Rev. 17	14	September 2014
M.9-4 (associated with UFSAR Rev. 17)	14	September 2014
M.9-5 (associated with UFSAR Rev. 17)	16	July 2017
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M.9-5b (associated with UFSAR Rev. 17)	16	July 2017
M.9-6 (associated with UFSAR Rev. 17)	16	July 2017
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M.9-13 (associated with UFSAR Rev. 17)	13	January 2014
M.9-14 (associated with UFSAR Rev. 17)	13	January 2014
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M.9-6 (associated with UFSAR Rev. 18)	*16*	*July 2017*
M.9-7 (associated with UFSAR Rev. 18)	*14*	*September 2014*
M.9-8 (associated with UFSAR Rev. 18)	*14*	*September 2014*
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M.11-12	14	September 2014
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DWG (sh. 3 of 6) NUH-HBU-1000-SAR	3	Not shown
DWG (sh. 4 of 6) NUH-HBU-1000-SAR	3	Not shown
DWG (sh. 5 of 6) NUH-HBU-1000-SAR	3	Not shown
DWG (sh. 6 of 6) NUH-HBU-1000-SAR	3	Not shown
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DWG (sh. 6 of 6) NUH-HBU-1200-SAR	0	Not shown
N.1-8	8	June 2004
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N.3-31	8	June 2004
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N.3-40	8	June 2004
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N.6-40	9	January 2006
N.6-41	9	January 2006
N.6-42	9	January 2006
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N.6-49	9	January 2006
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N.6-52	14	September 2014
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N.11-11	8	June 2004
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DWG (sh. 5 of 8) NUH24PTH-72-1009	0	Not shown
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P.5-2a	*18*	*January 2019*

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P.5-5	*18*	*January 2019*
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P.5-7	*18*	*January 2019*
P.5-8	*18*	*January 2019*
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P.5-15a	*18*	*January 2019*
P.5-15b	*18*	*January 2019*
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P.5-90c	*18*	*January 2019*
P.5-90d	*18*	*January 2019*
P.5-90e	*18*	*January 2019*
P.5-90f	*18*	*January 2019*
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P.8-6	16	July 2017
P.8-7	16	July 2017
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P.8-8a	16	July 2017
P.8-9	13	January 2014
P.8-10	17	March 2018
P.8-11	13	January 2014
P.8-11a	17	March 2018
P.8-12	11	February 2010
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P.9.13 (associated with UFSAR Rev. 11)	11	February 2010
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P.9-2 (associated with UFSAR Rev. 15)	13	January 2014
P.9-3 (associated with UFSAR Rev. 15)	14	September 2014
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P.9-9 (associated with UFSAR Rev. 16)	16	July 2017
P.9-10 (associated with UFSAR Rev. 16)	14	September 2014
P.9-11 (associated with UFSAR Rev. 16)	14	September 2014
P.9-12 (associated with UFSAR Rev. 16)	13	January 2014
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P.9-7a (associated with UFSAR Rev. 17)	16	July 2017
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P.9-13 (associated with UFSAR Rev. 17)	13	January 2014
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P.9-1 (associated with UFSAR Rev. 18)	*15*	*August 2016*
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P.9-7 (associated with UFSAR Rev. 18)	*16*	*July 2017*
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P.9-7b (associated with UFSAR Rev. 18)	*16*	*July 2017*
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P.9-12 (associated with UFSAR Rev. 18)	*13*	*January 2014*
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P.9-14 (associated with UFSAR Rev. 18)	*13*	*January 2014*
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P.10-12	9	January 2006
P.10-13	9	January 2006
P.10-14	9	January 2006
P.10-15	9	January 2006
P.10-16	9	January 2006
P.10-17	9	January 2006
P.10-18	9	January 2006
P.10-19	9	January 2006
P.10.20	9	January 2006
P.10.21	9	January 2006
P.10-22	9	January 2006
P.10-23	9	January 2006
P.11-1	14	September 2014
P.11-2	9	January 2006
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P.11-6a	14	September 2014
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R.1-3	13	January 2014
R.1-4	13	January 2014
R.1-5	9	January 2006
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DWG (sh. 3 of 9) NUH-03-6400-SAR	3	Not shown
DWG (sh. 4 of 9) NUH-03-6400-SAR	3	Not shown
DWG (sh. 5 of 9) NUH-03-6400-SAR	3	Not shown
DWG (sh. 6 of 9) NUH-03-6400-SAR	3	Not shown
DWG (sh. 7 of 9) NUH-03-6400-SAR	3	Not shown
DWG (sh. 8 of 9) NUH-03-6400-SAR	3	Not shown
DWG (sh. 9 of 9) NUH-03-6400-SAR	3	Not shown
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R.1-9	9	January 2006
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R.3-53	9	January 2006
R.3-54	15	August 2016
R.3-55	15	August 2016
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R.9-2	13	January 2014
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R.13-1	9	January 2006
R.14-1	9	January 2006
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T-ii	*18*	*January 2019*
T-iii	*18*	*January 2019*
T-iv	*18*	*January 2019*
T-v	*18*	*January 2019*
T-vi	*18*	*January 2019*
T-vii	*18*	*January 2019*
T-viii	*18*	*January 2019*
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T-xi	*18*	*January 2019*
T-xii	*18*	*January 2019*
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T-xiv	*18*	*January 2019*
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T-xvii	*18*	*January 2019*
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T.1-14	11	February 2010
T.1-15	11	February 2010
T.1-16	11	February 2010
T.1-17	11	February 2010
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DWG (sh. 3 of 5) NUH61BTH-1000-SAR	4	Not shown
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DWG (sh. 5 of 5) NUH61BTH-1000-SAR	4	Not shown
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DWG (sh. 1 of 2) NUH-61BTH-2001-SAR	4	8/26/16
DWG (sh. 2 of 2) NUH61BTH-2001-SAR	4	Not shown
DWG (sh. 1 of 6) NUH-61BTH-2002-SAR	3	7/17/17
DWG (sh. 2 of 6) NUH61BTH-2002-SAR	3	Not shown
DWG (sh. 3 of 6) NUH61BTH-2002-SAR	3	Not shown
DWG (sh. 4 of 6) NUH-61BTH-2002-SAR	3	Not shown
DWG (sh. 5 of 6) NUH61BTH-2002-SAR	3	Not shown
DWG (sh. 6 of 6) NUH61BTH-2002-SAR	3	Not shown
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DWG (sh. 2 of 2) NUH61BTH-2006-SAR	2	Not shown
DWG (sh. 1 of 2) NUH61BTH-72-1105	1	7/17/17
DWG (sh. 2 of 2) NUH61BTH-72-1105	1	Not shown
DWG (sh. 1 of 1) NUH-03-8007-SAR	1	1/8/14
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T.2-3	*18*	*January 2019*
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T.2-9a	17	March 2018
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T.2-11	13	January 2014
T.2-12	*18*	*January 2019*
T.2-13	11	February 2010
T.2-14	11	February 2010
T.2-15	*18*	*January 2019*
T.2-16	16	July 2017
T.2-17	*18*	*January 2019*
T.2-18	*18*	*January 2019*
T.2-19	*18*	*January 2019*
T.2-19a	*18*	*January 2019*
T.2-20	*18*	*January 2019*
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T.2-33	14	September 2014
T.2-34	14	September 2014
T.2-34a	*18*	*January 2019*
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T.2-36	14	September 2014
T.2-37	14	September 2014
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T.2-39	14	September 2014
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T.2-41	14	September 2014
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T.2-44	*18*	*January 2019*
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T.3.1-3	12	February 2012
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T.3.5-12	11	February 2010
T.3.5-13	*18*	*January 2019*
T.3.5-14	*18*	*January 2019*
T.3.5-15	*18*	*January 2019*
T.3.5-16	*18*	*January 2019*
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T.3.5-33	*18*	*January 2019*
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T.9-10 (associated with UFSAR Rev. 11)	11	February 2010
T.9-11 (associated with UFSAR Rev. 11)	11	February 2010
T.9-12 (associated with UFSAR Rev. 11)	11	February 2010
T.9-13 (associated with UFSAR Rev. 11)	11	February 2010
T.9.14 (associated with UFSAR Rev. 11)	11	February 2010
T.9.15 (associated with UFSAR Rev. 11)	11	February 2010
T.9.16 (associated with UFSAR Rev. 11)	11	February 2010
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T.9-5 (associated with UFSAR Rev. 12)	11	February 2010
T.9-6 (associated with UFSAR Rev. 12)	11	February 2010
T.9-7 (associated with UFSAR Rev. 12)	11	February 2010
T.9-8 (associated with UFSAR Rev. 12)	11	February 2010
T.9-9 (associated with UFSAR Rev. 12)	11	February 2010
T.9-10 (associated with UFSAR Rev. 12)	11	February 2010
T.9-11 (associated with UFSAR Rev. 12)	11	February 2010
T.9-12 (associated with UFSAR Rev. 12)	11	February 2010
T.9-13 (associated with UFSAR Rev. 12)	11	February 2010
T.9.14 (associated with UFSAR Rev. 12)	11	February 2010
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T.9-5 (associated with UFSAR Rev. 13)	13	January 2014
T.9-6 (associated with UFSAR Rev. 13)	13	January 2014
T.9-7 (associated with UFSAR Rev. 13)	13	January 2014
T.9-8 (associated with UFSAR Rev. 13)	13	January 2014
T.9-9 (associated with UFSAR Rev. 13)	13	January 2014
T.9-10 (associated with UFSAR Rev. 13)	13	January 2014
T.9-11 (associated with UFSAR Rev. 13)	13	January 2014
T.9-12 (associated with UFSAR Rev. 13)	11	February 2010
T.9-13 (associated with UFSAR Rev. 13)	13	January 2014
T.9.14 (associated with UFSAR Rev. 13)	13	January 2014
T.9.15 (associated with UFSAR Rev. 13)	13	January 2014
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T.9-1 (associated with UFSAR Rev. 14)	11	February 2010
T.9-2 (associated with UFSAR Rev. 14)	14	September 2014
T.9-3 (associated with UFSAR Rev. 14)	13	January 2014
T.9-4 (associated with UFSAR Rev. 14)	14	September 2014
T.9-5 (associated with UFSAR Rev. 14)	14	September 2014
T.9-6 (associated with UFSAR Rev. 14)	13	January 2014
T.9-7 (associated with UFSAR Rev. 14)	14	September 2014
T.9-8 (associated with UFSAR Rev. 14)	14	September 2014
T.9-9 (associated with UFSAR Rev. 14)	14	September 2014
T.9-10 (associated with UFSAR Rev. 14)	14	September 2014
T.9-11 (associated with UFSAR Rev. 14)	14	September 2014
T.9-12 (associated with UFSAR Rev. 14)	11	February 2010
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T.9.14 (associated with UFSAR Rev. 14)	13	January 2014
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T.9-6 (associated with UFSAR Rev. 15)	13	January 2014
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T.9-8 (associated with UFSAR Rev. 15)	14	September 2014
T.9-9 (associated with UFSAR Rev. 15)	14	September 2014
T.9-10 (associated with UFSAR Rev. 15)	14	September 2014
T.9-11 (associated with UFSAR Rev. 15)	14	September 2014
T.9-12 (associated with UFSAR Rev. 15)	11	February 2010
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T.9.14 (associated with UFSAR Rev. 15)	13	January 2014
T.9.15 (associated with UFSAR Rev. 15)	13	January 2014
T.9.16 (associated with UFSAR Rev. 15)	13	January 2014
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T.9-11 (associated with UFSAR Rev. 16)	14	September 2014
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T.9-9 (associated with UFSAR Rev. 17)	14	September 2014
T.9-10 (associated with UFSAR Rev. 17)	16	July 2017
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T.9.14 (associated with UFSAR Rev. 17)	13	January 2014
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T.9-2 (associated with UFSAR Rev. 18)	*15*	*August 2016*
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T.9-11 (associated with UFSAR Rev. 18)	*14*	*September 2014*
T.9-12 (associated with UFSAR Rev. 18)	*11*	*February 2010*
T.9-13 (associated with UFSAR Rev. 18)	*13*	*January 2014*
T.9.14 (associated with UFSAR Rev. 18)	*13*	*January 2014*
T.9.15 (associated with UFSAR Rev. 18)	*13*	*January 2014*
T.9.16 (associated with UFSAR Rev. 18)	*13*	*January 2014*
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U-vi	*18*	*January 2019*
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U-xix	*18*	*January 2019*
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U-xxiv	*18*	*January 2019*
U-xxv	*18*	*January 2019*
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U.1-11	16	July 2017
U.1-12	16	July 2017
U.1-13	11	February 2010
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DWG (sh. 1 of 3) NUH32PTH1-1001-SAR	4	7/18/17
DWG (sh. 2 of 3) NUH32PTH1-1001-SAR	4	Not shown
DWG (sh. 3 of 3) NUH32PTH1-1001-SAR	4	Not shown
DWG (sh. 1 of 2) NUH32PTH1-1002-SAR	3	7/18/17
DWG (sh. 2 of 2) NUH32PTH1-1002-SAR	3	Not shown
DWG (sh. 1 of 5) NUH32PTH1-1003-SAR	4	7/20/17
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DWG (sh. 1 of 5) NUH32PTH1-1005-SAR	2	8/26/16
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DWG (sh. 1 of 3) NUH-08-8002-SAR	2	7/17/17
DWG (sh. 2 of 3) NUH-08-8002-SAR	2	Not shown
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U.6-191	11	February 2010
U.6-191a	16	July 2017
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U.6-287	11	February 2010
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U.8-9	15	August 2016
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U.9-5 (associated with UFSAR Rev. 11)	11	February 2010
U.9-6 (associated with UFSAR Rev. 11)	11	February 2010
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U.9-8 (associated with UFSAR Rev. 12)	11	February 2010
U.9-9 (associated with UFSAR Rev. 12)	11	February 2010
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U.9-13 (associated with UFSAR Rev. 12)	11	February 2010
U.9-14 (associated with UFSAR Rev. 12)	11	February 2010
U.9-1 (associated with UFSAR Rev. 13)	11	February 2010
U.9-2 (associated with UFSAR Rev. 13)	13	January 2014
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U.9-5 (associated with UFSAR Rev. 14)	14	September 2014
U.9-6 (associated with UFSAR Rev. 14)	13	January 2014
U.9-7 (associated with UFSAR Rev. 14)	14	September 2014
U.9-8 (associated with UFSAR Rev. 14)	14	September 2014
U.9-9 (associated with UFSAR Rev. 14)	14	September 2014
U.9-10 (associated with UFSAR Rev. 14)	14	September 2014
U.9-11 (associated with UFSAR Rev. 14)	14	September 2014
U.9-12 (associated with UFSAR Rev. 14)	11	February 2010
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U.9-5 (associated with UFSAR Rev. 15)	14	September 2014
U.9-6 (associated with UFSAR Rev. 15)	13	January 2014
U.9-7 (associated with UFSAR Rev. 15)	14	September 2014
U.9-8 (associated with UFSAR Rev. 15)	14	September 2014
U.9-9 (associated with UFSAR Rev. 15)	14	September 2014
U.9-10 (associated with UFSAR Rev. 15)	14	September 2014
U.9-11 (associated with UFSAR Rev. 15)	14	September 2014
U.9-12 (associated with UFSAR Rev. 15)	11	February 2010
U.9-13 (associated with UFSAR Rev. 15)	14	September 2014
U.9-14 (associated with UFSAR Rev. 15)	13	January 2014
U.9-1 (associated with UFSAR Rev. 16)	11	February 2010
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U.9-6a (associated with UFSAR Rev. 16)	16	July 2017
U.9-7 (associated with UFSAR Rev. 16)	16	July 2017
U.9-7a (associated with UFSAR Rev. 16)	16	July 2017
U.9-8 (associated with UFSAR Rev. 16)	14	September 2014
U.9-9 (associated with UFSAR Rev. 16)	14	September 2014
U.9-10 (associated with UFSAR Rev. 16)	14	September 2014
U.9-11 (associated with UFSAR Rev. 16)	14	September 2014
U.9-12 (associated with UFSAR Rev. 16)	11	February 2010
U.9-13 (associated with UFSAR Rev. 16)	14	September 2014
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U.9-5 (associated with UFSAR Rev. 17)	14	September 2014
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U.9-6a (associated with UFSAR Rev. 17)	16	July 2017
U.9-7 (associated with UFSAR Rev. 17)	16	July 2017
U.9-7a (associated with UFSAR Rev. 17)	16	July 2017
U.9-8 (associated with UFSAR Rev. 17)	14	September 2014
U.9-9 (associated with UFSAR Rev. 17)	14	September 2014
U.9-10 (associated with UFSAR Rev. 17)	14	September 2014
U.9-11 (associated with UFSAR Rev. 17)	14	September 2014
U.9-12 (associated with UFSAR Rev. 17)	11	February 2010
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U.9-7a (associated with UFSAR Rev. 18)	*16*	*July 2017*
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U.9-9 (associated with UFSAR Rev. 18)	*14*	*September 2014*
U.9-10 (associated with UFSAR Rev. 18)	*14*	*September 2014*
U.9-11 (associated with UFSAR Rev. 18)	*14*	*September 2014*
U.9-12 (associated with UFSAR Rev. 18)	*11*	*February 2010*
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V.2-2	13	January 2014
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V.3-1	13	January 2014
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V.3-5	10	February 2008
V.3-6	10	February 2008
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V.9-1	13	January 2014
V.9-2	13	January 2014
V.10-1	10	February 2008
V.11-1	10	February 2008
V.11-2	10	February 2008
V.11-3	11	February 2010
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V.13-1	10	February 2008
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DWG (sh. 3 of 8) NUH-03-8008-SAR	1	Not shown
DWG (sh. 4 of 8) NUH-03-8008-SAR	1	Not shown
DWG (sh. 5 of 8) NUH-03-8008-SAR	1	Not shown
DWG (sh. 6 of 8) NUH-03-8008-SAR	1	Not shown
DWG (sh. 7 of 8) NUH-03-8008-SAR	1	Not shown
DWG (sh. 8 of 8) NUH-03-8008-SAR	1	Not shown
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DWG (sh. 2 of 6) NUH-03-8009-SAR	1	Not shown
DWG (sh. 3 of 6) NUH-03-8009-SAR	1	Not shown
DWG (sh. 4 of 6) NUH-03-8009-SAR	1	Not shown
DWG (sh. 5 of 6) NUH-03-8009-SAR	1	Not shown
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W.2-13	14	September 2014

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Y.2-28	*18*	*January 2019*
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Y.2-42	*18*	*January 2019*
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Y.4-91	14	September 2014
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Y.5-3	17	March 2018
Y.5-4	16	July 2017
Y.5-5	16	July 2017
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Y.5-7	14	September 2014
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Y.9-10 (associated with UFSAR Rev. 14)	14	September 2014
Y.9-11 (associated with UFSAR Rev. 14)	14	September 2014
Y.9-12 (associated with UFSAR Rev. 14)	14	September 2014
Y.9-13 (associated with UFSAR Rev. 14)	14	September 2014
Y.9-14 (associated with UFSAR Rev. 14)	14	September 2014
Y.9-1 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-2 (associated with UFSAR Rev. 15)	15	August 2016
Y.9-3 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-4 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-5 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-6 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-7 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-8 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-9 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-10 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-11 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-12 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-13 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-14 (associated with UFSAR Rev. 15)	14	September 2014
Y.9-1 (associated with UFSAR Rev. 16)	14	September 2014
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Y.9-11 (associated with UFSAR Rev. 16)	14	September 2014
Y.9-12 (associated with UFSAR Rev. 16)	14	September 2014
Y.9-13 (associated with UFSAR Rev. 16)	14	September 2014
Y.9-14 (associated with UFSAR Rev. 16)	14	September 2014
Y.9-1 (associated with UFSAR Rev. 17)	14	September 2014
Y.9-2 (associated with UFSAR Rev. 17)	15	August 2016
Y.9-3 (associated with UFSAR Rev. 17)	14	September 2014
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Y.9-5 (associated with UFSAR Rev. 17)	14	September 2014
Y.9-6 (associated with UFSAR Rev. 17)	16	July 2017
Y.9-7 (associated with UFSAR Rev. 17)	14	September 2014
Y.9-8 (associated with UFSAR Rev. 17)	16	July 2017
Y.9-8a (associated with UFSAR Rev. 17)	16	July 2017
Y.9-9 (associated with UFSAR Rev. 17)	14	September 2014
Y.9-10 (associated with UFSAR Rev. 17)	16	July 2017
Y.9-11 (associated with UFSAR Rev. 17)	14	September 2014
Y.9-12 (associated with UFSAR Rev. 17)	14	September 2014
Y.9-13 (associated with UFSAR Rev. 17)	14	September 2014
Y.9-14 (associated with UFSAR Rev. 17)	14	September 2014
Y.9-1 (associated with UFSAR Rev. 18)	*14*	*September 2014*
Y.9-2 (associated with UFSAR Rev. 18)	*15*	*August 2016*
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Y.9-10 (associated with UFSAR Rev. 18)	*16*	*July 2017*
Y.9-11 (associated with UFSAR Rev. 18)	*14*	*September 2014*
Y.9-12 (associated with UFSAR Rev. 18)	*14*	*September 2014*
Y.9-13 (associated with UFSAR Rev. 18)	*14*	*September 2014*
Y.9-14 (associated with UFSAR Rev. 18)	*14*	*September 2014*
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Y.10-4	14	September 2014
Y.10-5	14	September 2014
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Y.11-11	14	September 2014
Y.11-12	14	September 2014
Y.11-13	14	September 2014
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Y.14-1	14	September 2014
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Z-ii	*18*	*January 2019*
Z-iii	*18*	*January 2019*
Z-iv	*18*	*January 2019*
Z-v	*18*	*January 2019*
Z-vi	*18*	*January 2019*
Z-vii	*18*	*January 2019*
Z-viii	*18*	*January 2019*
Z-ix	*18*	*January 2019*
Z-x	*18*	*January 2019*
Z-xi	*18*	*January 2019*
Z-xii	*18*	*January 2019*
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Z.1-8	14	September 2014
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Z.1-10	14	September 2014
Z.1-11	14	September 2014
Z.1-12	14	September 2014
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DWG (sh. 4 of 4) NUH37PTH-72-1001	0	Not shown
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DWG (sh. 3 of 5) NUH37PTH-72-1002	1	Not shown
DWG (sh. 4 of 5) NUH37PTH-72-1002	1	Not shown
DWG (sh. 5 of 5) NUH37PTH-72-1002	1	Not shown
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DWG (sh. 3 of 4) NUH37PTH-72-1003	1	Not shown
DWG (sh. 4 of 4) NUH37PTH-72-1003	1	Not shown
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DWG (sh. 4 of 6) NUH37PTH-72-1004	0	Not shown
DWG (sh. 5 of 6) NUH37PTH-72-1004	0	Not shown
DWG (sh. 6 of 6) NUH37PTH-72-1004	0	Not shown
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DWG (sh. 4 of 7) NUH37PTH-72-1011	0	Not shown
DWG (sh. 5 of 7) NUH37PTH-72-1011	0	Not shown
DWG (sh. 6 of 7) NUH37PTH-72-1011	0	Not shown

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DWG (sh. 5 of 7) NUH37PTH-72-1016	0	Not shown
DWG (sh. 6 of 7) NUH37PTH-72-1016	0	Not shown
DWG (sh. 7 of 7) NUH37PTH-72-1016	0	Not shown
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DWG (sh. 7 of 7) NUH37PTH-72-1017	0	Not shown
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Z.2-3	*18*	*January 2019*
Z.2-3a	*18*	*January 2019*
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Z.2-10a	17	March 2018
Z.2-11	14	September 2014
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Z.2-22	14	September 2014
Z.2-23	*18*	*January 2019*
Z.2-24	*18*	*January 2019*
Z.2-25	*18*	*January 2019*
Z.2-26	*18*	*January 2019*
Z.2-27	*18*	*January 2019*
Z.2-28	14	September 2014
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Z.5-4	*18*	*January 2019*
Z.5-5	*18*	*January 2019*

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Z.8-12a	17	March 2018
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Z.9-5 (associated with UFSAR Rev. 14)	14	September 2014
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Z.9-5 (associated with UFSAR Rev. 16)	14	September 2014
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Z.9-8 (associated with UFSAR Rev. 16)	16	July 2017
Z.9-8a (associated with UFSAR Rev. 16)	16	July 2017
Z.9-8b (associated with UFSAR Rev. 16)	16	July 2017
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Z.9-11a (associated with UFSAR Rev. 16)	16	July 2017
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Z.9-10 (associated with UFSAR Rev. 17)	16	July 2017
Z.9-11 (associated with UFSAR Rev. 17)	14	September 2014

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Z.9-12 (associated with UFSAR Rev. 17)	14	September 2014
Z.9-13 (associated with UFSAR Rev. 17)	16	July 2017
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Z.9-5 (associated with UFSAR Rev. 18)	*14*	*September 2014*
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Z.9-6a (associated with UFSAR Rev. 18)	*16*	*July 2017*
Z.9-7 (associated with UFSAR Rev. 18)	*14*	*September 2014*
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Z.9-8a (associated with UFSAR Rev. 18)	*16*	*July 2017*
Z.9-8b (associated with UFSAR Rev. 18)	*16*	*July 2017*
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Z.11-10	*18*	*January 2019*
Z.11-11	14	September 2014

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Z.13-1	14	September 2014
Z.14-1	14	September 2014

Revision 12 of this UFSAR incorporates design modifications implemented per 10 CFR 72.48 since the issuance of UFSAR Revision 11.

Revision 13 of this UFSAR incorporates design modifications implemented per 72.48 since the issuance of UFSAR Revision 12, including FCN 721004-951 that was approved prior to the issuance of Revision 12. It also incorporates changes implemented due to approval of Amendment 11 to CoC 1004.

Revision 14 of this UFSAR incorporates design modifications implemented per 72.48 since the issuance of UFSAR Revision 13. It also incorporates changes implemented due to approval of Amendment 13 to CoC 1004.

Revision 15 of this UFSAR incorporates design modifications implemented per 10 CFR 72.48 since the issuance of UFSAR Revision 14.

Revision 16 of this UFSAR incorporates design modifications implemented per 72.48 since the issuance of UFSAR Revision 15. It also incorporates changes implemented due to approval of Amendment 14 to CoC 1004.

Revision 17 of this UFSAR incorporates design modifications implemented per 10 CFR 72.48 since the issuance of UFSAR Revision 16. It also incorporates changes and CoC holder commitments resulting from the review and approval of the renewal of CoC 1004 (i.e., incorporation of aging management-related items).

Revision 18 of this UFSAR incorporates design modifications implemented per 72.48 since the issuance of UFSAR Revision 17. It also incorporates changes implemented due to approval of Amendment 15 to CoC 1004.

selected. A detailed description of the 69BTH system, including drawings, authorized payload contents and supporting safety analyses are provided in Appendix Y of this UFSAR.

Amendment No. 13 to CoC 1004 also adds the NUHOMS® 37PTH system to the standardized NUHOMS® system. The NUHOMS® 37PTH DSC is designed to store a total of 37 intact or up to four damaged and balance intact) PWR fuel assemblies with a maximum assembly average initial enrichment of 5.0 wt. % U-235, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of 3.0 years. Three heat load-zoning configurations are authorized and the system is designed to accommodate a decay heat load of up to 30.0 kW per DSC depending upon the specific configuration selected. A detailed description of the 37PTH DSC, including drawings, authorized payload contents and supporting safety analyses are provided in Appendix Z of this UFSAR.

Amendment No. 14 to CoC 1004 adds fuel qualification tables (FQTs) at 400 kgU per assembly for fuel to be stored in the 32PT, 24PTH, and 37PTH DSCs, increases the maximum uranium loading from 490 kgU per assembly to 492 kgU per assembly for fuel to be stored in the 24PTH, 32PTH1, and 37PTH DSCs, expands the fuel qualification tables for fuel to be stored in the 24PTH and 32PTH1 DSCs for burnup and enrichment combinations not previously allowed, allows for the storage of up to four failed fuel cans (FFCs) in the 32PTH1 DSC, creates a new 32PTH1 heat load zoning configuration (HLZC) capable of storing intact fuel, damaged fuel, and up to four FFCs, expands the 37PTH criticality analysis to include poison rod assemblies (PRAs), adds FQTs to allow fuel with a uranium loading as low as 170 kgU per assembly to be stored in the 61BTH and 69BTH DSCs, expands the fuel qualification tables for the 61BTH, 69BTH, and 37PTH DSCs for additional burnup and enrichment combinations, including low enrichment and low burnup fuel, allows for up to 61 damaged BWR fuel assemblies to be stored in the 61BTH DSC, includes new heat load zoning configurations for the 61BTH and 69BTH DSCs, evaluates the shielding impacts of 32PTH1 DSC stored in the HSM-H with a concrete density reduced from 145 pcf to 140 pcf, and uniform gaps at 1.5 inches between the HSM-Hs, improves language in the technical specifications (TS), such as for that of definitions, DSC/HSM/TC combinations, and dose measurements, and allows for acceptance testing for neutron absorber content to be performed by either neutron transmission or by B-10 volume density measurement.

Amendment No. 15 to CoC 1004 consolidates and standardizes the fuel qualification tables (FQTs) for four PWR systems (32PT, 24PTH, 32PTH1 and 37PTH). The consolidated/standardized FQTs in some cases expand the authorized and explicitly evaluated initial enrichment and burnup ranges allowed for the four PWR systems and cover range of heat loads allowed for each fuel assembly (FA) in the various zones identified Heat Load Zone Configurations (HLZCs) for each of the four PWR systems. The consolidated/standardized FQTs provide for minimum required cooling times, as low as two years, as a function of enrichment and burnup (BU). Further, the FQTs are generated for three MTU loadings per FA and allow for interpolation between MTU loadings and to establish cooling times for FAs that fall into the unanalyzed regions of the FQTs. New HLZCs are added for the 24PTH (HLZC #6), and 32PTH1 (HLZC #5 and HLZC #6). The reported storage module, transfer cask and site dose rates along with the occupational exposure analyses for these systems is updated to account for the bounding HLZCs and consolidated FQTs for each of the four PWR systems. AMD 15 & 72.48

Amendment No. 15 to CoC 1004 also makes the following changes to the 32PT system: 1) storage of damaged CE 14x15, WE 14x14 and WE 17x17 FAs with top and bottom endcaps in the Type A1 or A2 32PT DSC baskets, 2) the ability to store up to 8 failed fuel canisters (FFCs) in the Type A1 or A2 32PT DSC baskets 3) adds a 32MMC Poison Plate basket type (32PT-32) to the 32PT design which accommodates WE 17x17 class FAs, 4) adds the ability to use poison rod assemblies (PRAs) with the Type B and C 32PT DSC baskets with Type 1 or 2 MMC poison plates storing WE 17x17 class assemblies, 5) allow the use of Rod Control Cluster Assemblies (RCCAs) or Control Rod Assemblies (CRAs) as substitutes for PRAs in several cases, 6) expands the authorized fuel cladding materials that are qualified for storage in the 32PT system, and 7) creates a new 32PT heat load zoning configuration (HLZC) #4.

Amendment No. 15 to CoC 1004 also adds two additional BWR fuel types, GNF2 and ATRIUM-11, as authorized contents to the 61BTH system, adds an alternative loading option to HLZC #10, adds a set of FQTs for 170 MTU loadings per FA and allows for interpolation between the MTU loadings to establish cooling times for FAs that fall into the unanalyzed regions of the FQTs.

Amendment No 15 to CoC 1004 also provides clarification for use of solar shields and rain shields for the TC during transfer operations.

Chapters 1 through 8 and Appendices A through H of this FSAR provide the supporting licensing basis for the Standardized NUHOMS®-24P and -52B Systems only.

A complete description of the new systems addressed by the above listed amendments, including supporting safety analysis, is located within self-contained Appendices to this FSAR as summarized in the following table:

Amendment No.	Description	Location of Supporting Licensing Basis
3	Addition of the NUHOMS®-61BT DSC to the contents of the Standardized NUHOMS® System	Appendix K
N/A	Addition of the NUHOMS®-24PT2 DSC to the contents of the Standardized NUHOMS® System	Appendix L
4	Addition of low burnup fuel to the contents of the NUHOMS®-24P DSC	Chapter 3
5	Addition of the NUHOMS®-32PT DSC to the Standardized NUHOMS® System	Appendix M
6	Addition of the NUHOMS®-24PHB DSC to the Standardized NUHOMS® System	Appendix N
7	Addition of damaged fuel to the contents of the NUHOMS®-61BT DSC	Appendix K
8	(a) Addition of the NUHOMS® 24PTH System to the Standardized NUHOMS® System	Appendix P
	(b) Revision of the authorized contents of the 32PT DSC to include low enrichment and reconstituted fuel	Appendix M
	(c) Revision of the authorized contents of the 24PHB DSC to include additional fuel types	Appendix N

Amendment No.	Description	Location of Supporting Licensing Basis
N/A	Addition of an alternate version of the HSM, designated as HSM Model 152, to the Standardized NUHOMS® System	Appendix R
N/A	Addition of an alternate version of the HSM, designated as HSM Model 202, to the Standardized NUHOMS® System	Appendix V
N/A	Addition of an alternate version of the OS197 Transfer Cask, designated as OS197L, to the Standardized NUHOMS® System. Revised TS to conform with NUREG 1745 format.	Appendix W
9	Addition of FANP9 fuel to the contents of the 61BT DSC	Appendix K
10	Addition of Control Components to the contents of the 32PT DSC	Appendix M
	Addition of WE 15x15 Partial Length Shield Assemblies to the contents of the 24PTH DSC	Appendix P
	Addition of 61BTH system to the Standardized NUHOMS® System	Appendix T
	Addition of 32PTH1 system to the Standardized NUHOMS® System	Appendix U
11	Addition of the OS197L TC to the Standardized NUHOMS® System	Appendix W
13	Addition of 69BTH system to the Standardized NUHOMS® System	Appendix Y
	Addition of 37PTH system to the Standardized NUHOMS® System	Appendix Z
14	Add fuel qualification tables at 400 kgU per assembly for fuel to be stored in the 32PT, 24PTH, and 37PTH DSCs	Appendices M, P, and Z
	Increase the maximum uranium loading from 490 kgU per assembly to 492 kgU per assembly for fuel to be stored in the 24PTH, 32PTH1, and 37PTH DSCs	Appendices P, U, and Z
	Expand the fuel qualification tables for fuel to be stored in the 24PTH and 32PTH1 DSCs for burnup and enrichment combinations not previously allowed	Appendices P and U
	Allow for the storage of up to four FFCs in the 32PTH1 DSC	Appendix U
	Create a new 32PTH1 HLZC be capable of storing intact fuel, damaged fuel, and up to four FFCs	Appendix U
	Expand 37PTH criticality analysis to include PRAs	Appendix Z
	Add fuel qualification tables to allow fuel with a uranium loading as low as 170 kgU per assembly to be stored in the 61BTH and 69BTH DSCs	Appendices T and Y
	Expand the fuel qualification tables for the 61BTH, 69BTH, and 37PTH DSCs for additional burnup and enrichment combinations, including low enrichment and low burnup fuel	Appendices T, Y, and Z

Amendment No.	Description	Location of Supporting Licensing Basis
	Allow for up to 61 damaged BWR fuel assemblies to be stored in the 61BTH DSC	Appendix T
	Include new heat load zoning configurations for the 61BTH and 69BTH DSCs	Appendices T and Y
	Evaluate the shielding impacts of 32PTH1 DSC stored in the HSM-H with a concrete density reduced from 145 pcf to 140 pcf, and uniform gaps at 1.5 inches between the HSM-Hs	Appendix U
	Allow for acceptance testing for neutron absorber content to be performed by either neutron transmission or by B-10 volume density measurement	Appendices K, M, P, T, U, Y, and Z
15	*Added Bases information for TS 5.3.1 regarding use of a solar shield.*	*Chapter 10*
	Generically for the 32PT, 24PTH, 32PTH1, and 37PTH PWR DSCs - Unified and standardized fuel qualification tables	*Appendices M, P, U, and Z, respectively*
	32PT DSC System Changes • *storage of damaged fuel* • *storage of failed fuel* • *adds a 32MMC Poison Plate basket type (32PT-32)* • *adds the ability to use poison rod assemblies (PRAs) with the Type B and C 32PT DSC baskets* • *allow the use of Rod Control Cluster Assemblies (RCCAs) or Control Rod Assemblies (CRAs) as substitutes for PRAs* • *expands the authorized fuel cladding materials* *creates a new 32PT heat load zoning configuration (HLZC) #4.*	*Appendix M*
	24PTH DSC System Changes *Creation of a new 24PTH heat load zoning configuration (HLZC) #6*	*Appendix P*
	61BTH DSC System Changes • *Addition of alternative loading option to 61BTH heat load zoning configuration (HLZC) #10* • *Expanded fuel qualification tables* *Addition of new fuel types (GNF-2 and ATRIUM 11), and ability to store damaged fuel*	*Appendix T*
	32PTH1 DSC System Changes *Added new heat load zoning configurations (HLZCs) #5 & #6*	*Appendix U*

The 61BT DSC basket structure consists of assemblies of stainless steel fuel compartments held in place by basket rails and holddown ring. The four and nine compartment assemblies are held together by welded stainless steel boxes wrapped around the fuel compartments, which also retain the neutron absorber plates between the compartments in the assemblies. The borated aluminum or boron carbide/aluminum metal matrix composite plates (neutron absorber plates) provide the necessary criticality control and provide the heat conduction paths from the fuel assemblies to the cask cavity wall.

The 32PT DSC basket structure is a box type assembly of high strength XM-19 stainless steel surrounded by transition rails. Inside the compartments, around the fuel assemblies, the borated aluminum or Boralyn® plates (neutron poison plates) provide the necessary criticality control and provide the heat conduction paths from the fuel assemblies to the cask cavity wall. This method of construction forms a very strong structure of compartment assemblies which provide for storage of 32 fuel assemblies. Appendix M provides the details of the 32PT DSC.

The design details for the 24PTH, 61BTH and 32PTH1 69BTH and 37PTH baskets are provided in Appendices P, T, U, Y, and Z, respectively.

Aging Management Program Requirements

Aging management program (AMP) requirements for use of the 24P and 52B DSC during the period of extended storage operations are contained in Section 12.3. Applicable TLAAs performed for the initial CoC 1004 renewal application are provided in Section 12.2.

1.3.1.2 Horizontal Storage Module Model 80 and Model 102

An isometric view of the two alternate designs of a prefabricated HSM utilized to form an array of HSMs is shown in Figure 1.2-2 and 1.2-2a. Four additional alternate module designs, designated as HSM-H, HSM Model 152, HSM Model 202, *and HSM-HS* are discussed in detail in Appendices P, R, V, and U, respectively.

Each HSM provides a self-contained modular structure for storage of spent fuel canisterized in a DSC as illustrated in Figure 1.3-4. The HSM is constructed from reinforced concrete and structural steel. The thick concrete roof and walls of the HSM provide substantial neutron and gamma shielding. Contact doses for the HSM are designed to be ALARA.

The nominal thickness of the HSM roof and exterior walls of an HSM array for biological shielding is about three feet. Separate shielding walls are utilized at the end of a module row to provide the required thickness. Similarly, an additional shield wall is used at the rear of the module if the ISFSI is configured as single module rows. Sufficient shielding between HSMs in an HSM array to prevent scatter in adjacent HSMs during loading and retrieval operations is provided by thick concrete side walls. The inlet and outlet vents are designed to take advantage of the self-shielding of adjacent HSMs.

3.3 Safety Protection System

3.3.1 General

The NUHOMS® system is designed for safe and secure, long-term containment and dry storage of SFAs. The components, structures, and equipment which are designed to assure that this safety objective is met are shown in Table 3.3-1. The key elements of the NUHOMS® system and its operation which require special design consideration are:

A. Minimizing the contamination of the DSC exterior by fuel pool water.

B. The double closure seal welds on the DSC shell to form a pressure retaining containment boundary and to maintain a helium atmosphere.

C. Minimizing personnel radiation exposure during DSC loading, closure, and transfer operations.

D. Design of the transfer cask and DSC for postulated accidents.

E. Design of the HSM passive ventilation system for effective decay heat removal to ensure the integrity of the fuel cladding.

F. Design of the DSC basket assembly to ensure subcriticality.

These items are addressed in the following subsections.

3.3.2 Protection by Multiple Confinement Barriers and Systems

3.3.2.1 Confinement Barriers and Systems

The radioactive material which the NUHOMS® ISFSI confines is the spent fuel assemblies and the associated contaminated materials. These radioactive materials are confined by the multiple barriers listed in Table 3.3-2.

During fuel loading operations, the radioactive material in the plant's fuel pool is prevented from contacting the DSC exterior by filling the cask/DSC annulus with uncontaminated, demineralized water prior to placing the cask and DSC in the fuel pool. This places uncontaminated water in the annulus between the DSC and cask interior. In addition, the cask/DSC annulus opening at the top of the cask is sealed using an inflatable seal to prevent pool water from entering the annulus. This procedure

7.2.3.2 Determining Fuel Assembly Minimum Required Cooling Times Using Fitting Equations

Fuel qualification tables (FQTs) are developed to provide the minimum required cooling times needed for the authorized fuel assemblies for a given decay heat limit and/or radiological sources (combined total dose rates). As described in Section M.5.*2.6*, the FQTs are developed for the heavy metal loadings *of 380, 475 and 492 kgU. (The 492 kgU FQTs do not apply to the 32PT DSC.)* As described in Chapter T.5 and Y.5, the FQTs are developed for the 61BTH and 69BTH DSCs, respectively, for two heavy metal loadings – *170* KgU and 198 KgU.

This section provides a method for calculating the minimum required cooling time for a given fuel assembly, with an intermediate heavy metal loading that is in between the two discrete heavy metal load values mentioned above. It is demonstrated that the minimum required cooling times can be calculated by using a simple fitting equation. Section 7.2.3.2.1 provides details on the fitting equation and how to determine the various terms of the equations. Section 7.2.3.2.2 provides several examples on how to use the fitting equation. Section 7.2.3.2.3 provides verification and validation information concerning the accuracy of the fitting equation results.

7.2.3.2.1 Fitting Equations for Determining Minimum Required FA Cooling Times

Determining the minimum required cooling time for a given FA using this method is a two-step process. Step one involves clearly identifying the variable value inputs needed for the fitting equation for the applicable system, some of which are user inputs and others are looked up in FQTs. Step two involves determining the minimum required cooling time for the FA in question in the applicable system by using the looked up variable value inputs with the applicable fitting equation.

First, it is necessary to clearly identifying the variable value inputs needed for the fitting equation. These values are listed in Table 7.2-12 below.

Table 7.2-12
Input Parameters Needed for Fitting Equation

	Low kgU/FA Data	Unknown Cool Time Intermediate value kgU/FA Data	High kgU/FA Data
Metal Loading (kgU)	*380* for PWR 170 for BWR (kgU_{low})	*380* < kgU_{new} < 475 (32PT) *380 < kgU_{new} < 475, or* *475 < kgU_{new} < 492* (Other PWR Systems) 170 < kgU_{new} < 198 (BWR Systems)	475 for 32PT *492* for other PWR Systems 198 for BWR Systems (kgU_{high})
Burnup (GWD/MTU)	BU (user input) **(This value must be identical for all three FAs)**		
Wt. % U235 (%)	w/o (user input) **(This value must be identical for all three FAs)**		
Decay Heat (kWt/FA)	DH (user input) **(This value must be identical for all three FAs)**		
Min. Cooling Time (years)	CT_{low} (Find in FQT using Bu, wt. %, and decay heat)	CT_{new} (Determined by fitting equation)	CT_{high} (Find in FQT using Bu, wt. %, and decay heat)

Next, the generic form of the fitting equation is defined. This equation determines the intermediate metal loading FA cooling time.

Generic Fitting Equation 1

$$CT_{new} = \frac{CT_{high} * \ln(kgU_{new}/kgU_{low}) + CT_{low} * [\ln(kgU_{high}/kgU_{low}) - \ln(kgU_{new}/kgU_{low})]}{\ln(kgU_{high}/kgU_{low})}$$

7.2.3.2.2 Fitting Equation Examples

This fitting equation methodology is currently applicable to the 32PT, *32PTH1,* 24PTH, 37PTH, 61BTH, and 69BTH systems only. Although this Appendix describes the 32PT system, this section contains examples on how to apply this method to each of the above mentioned systems. These examples will be referenced in the technical specifications for utility user guidance.

7.2.3.2.2.1 32PT

For the 32PT example, a 3.2 *wt. %* without FA with 430 kgU of metal loading at 55 GWD/MTU of burnup is used. The FA needs to be loaded into a heat loading zone, which cannot exceed 0.87 kWt/FA. The minimum required cooling times for PWR FAs in 32PT DSC are tabulated in FQTs shown in Appendix Chapter M.2. Using these FQTs, the known values are identified in Table 7.2-13.

Table 7.2-13
Input Parameters Needed for 32PT Example

	Low kgU/FA Data	Unknown Cool Time Intermediate value kgU/FA Data	High kgU/FA Data
Metal Loading (kgU)	*380* (kgU_{low})	430 (kgU_{new})	475 (kgU_{high})
Burnup (GWD/MTU)	55		
Wt. % U235 (%)	3.2		
Decay Heat (kWt/FA)	0.87		
Min. Cooling Time (years)	CT_{low} (*10.3*)	CT_{new} (TBD)	CT_{high} (*17.8*)

Generic Fitting Equation 1 may be simplified by substituting the known low and high kgU/FA values of *380* and 475 into it. This simplification results in Fitting Equation 2, which can be used to determine the intermediate FA MTU cooling time for the 32PT system.

Fitting Equation 2 for 32PT

$$CT_{new} = 4.4814*[CT_{high}*ln(kgU_{new}/380) - CT_{low}*ln(kgU_{new}/475)]$$

Substituting the known input values into Equation 2, the value of minimum cooling time for the 430 kgU FA is found.

Equation 2 Example Solution

$$CT_{new} = 4.4814*[17.8*ln(430/380) - 10.3*ln(430/475)] = 14.45 \text{ years}$$

The minimum required cooling time determined for the 430 kgU FA is *14.5* years. The result of the fitting equation shall be rounded up to the next highest single decimal place. Fitting Equation 2 will be used in the notes and examples of the FQTs for the 32PT system for calculating minimum required cooling times for FAs with an intermediate heavy metal loading.

7.2.3.2.2.2 24PTH, 32PTH1, and 37PTH

For the 24PTH, 32PTH1, and 37PTH DSCs, FQTs are available at 380 kgU, 475 kgU, and 492 kgU. Therefore, if 380 kgU < kgU_{new} < 475 kgU, Fitting Equation 2 applies. If 475 kgU < kgU_{new} < 492 kgU, a fitting equation may be derived using the same method, as illustrated in the following example.

In this example, kgU_{new} = 480 kgU, enrichment = 5.0 wt. %, burnup = 62 GWd/MTU, and decay heat = 1.00 kW/FA. From the FQTs, CT_{low} = 15.7 years and CT_{high} = 17.1 years. The necessary data is summarized in Table 7.2-14.

Table 7.2-14
Input Parameters Needed for 37PTH Example

	Low kgU/FA Data	Unknown Cool Time Intermediate value kgU/FA Data	High kgU/FA Data
Metal Loading (kgU)	*475* (kgU_{low})	*480* (kgU_{new})	*492* (kgU_{high})
Burnup (GWD/MTU)		*6.2*	
Wt. % U235 (%)		*5.0*	
Decay Heat (kWt/FA)		*1.0*	
Min. Cooling Time (years)	CT_{low} (*15.7*)	CT_{new} (TBD)	CT_{high} (*17.1*)

Generic Fitting Equation 1 may be simplified by substituting the known low and high kgU/FA values of 475 and 492 kgU. This simplification results in Fitting Equation 3:

Fitting Equation 3 for 24PTH, 32PTH1, and 37PTH DSCs, 475 kgU < kgUnew < 492 kgU

$$CT_{new} = 28.4382*[CT_{high}*ln(kgU_{new}/475) - CT_{low}*ln(kgU_{new}/492)]$$

Substituting the known input values into Equation 3, the value of the minimum cooling time for the 480 kgU fuel assembly is found:

Equation 3 Example Solution

$$CT_{new} = 28.4382*[17.1*ln(480/475) - 15.7*ln(480/492)] = 16.12\ years$$

The result of the fitting equation shall be rounded up to the next highest single decimal place, or 16.2 years.

7.2.3.2.2.3 61BTH and 69BTH

For the 61BTH and 69BTH examples, a 3.4 *wt. %* FA with 180 kgU of metal loading at 62 GWD/MTU of burnup is used. The FA needs to be loaded into a heat loading zone, which cannot exceed 0.9 kWt/FA. The minimum required cooling times for BWR FAs in the 61BTH DSC are tabulated in FQTs shown in Appendix Chapter T.2. The minimum required cooling times for BWR FAs in the 69BTH DSC are tabulated in FQTs shown in Appendix Chapter Y.2. Using the 61BTH FQTs, the known values are identified in Table 7.2-15.

Table 7.2-15
Input Parameters Needed for 61BTH Example

	Low kgU/FA Data	Unknown Cool Time Intermediate value kgU/FA Data	High kgU/FA Data
Metal Loading (kgU)	170 (kgU_{low})	180 (kgU_{new})	198 (kgU_{high})
Burnup (GWD/MTU)		62	
Wt. % U235 (%)		3.4	
Decay Heat (kWt/FA)		0.9	
Min. Cooling Time (years)	CT_{low} (4.5)	CT_{new} (TBD)	CT_{high} (5.0)

Generic Fitting Equation 1 may be simplified by substituting the known low and high kgU/FA values of 170 and 198 into it. This simplification results in Fitting Equation 4, which can be used to determine the intermediate FA MTU cooling time for the 61BTH or 69BTH systems.

Fitting Equation 4 for 61BTH and 69BTH

$$CT_{new} = 6.56 * \{[(ln(kgU_{new}) - 5.14] * CT_{high} - [ln(kgU_{new}) - 5.29] * CT_{low}\}$$

Substituting the known input values into Equation 4, the value of minimum cooling time for the 180 kgU FA is found.

Equation 4 Example Solution

$$CT_{new} = 6.56 * \{[(\ln(180) - 5.14] * 5.0 - [\ln(180) - 5.29] * 4.5 = 4.69 \text{ years}\}$$

The minimum required cooling time determined for the 180 kgU FA is 4.7 years. The result of the fitting equation shall be rounded up to the next highest single decimal place. Fitting Equation 4 will be used in the notes and examples of the FQTs for the 61BTH and 69BTH systems for calculating minimum required cooling times for FAs with an intermediate heavy metal loading.

7.2.3.2.3 Verification of Fitting Equation Methodology

7.2.3.2.3.1 PWR Fuel in 24PTH and 37PTH

Table M.5-45 provides examples of cooling time determinations for various kgU and burnup/enrichment using the fitting equation and the linear interpolation using cooling times determined in FQTs explicitly generated at *380* kgU and *492* kgU.

As shown in Table M.5-45, the predicted cooling times obtained using Fitting Equation 3 for 24PTH and 37PTH are in agreement with those obtained by the linear interpolation approach. For certain cases, the predicted cooling times by the fitting equations are slightly higher.

7.2.3.2.3.2 BWR Fuel in 61BTH and 69BTH

Table M.5-47 provides examples of cooling time determinations for various kgU and burnup/enrichment using the fitting equations and the linear interpolation using cooling times determined in FQTs explicitly generated at 170 kgU and *198* kGU.

As shown in Table M.5-47, the predicted cooling times obtained using the fitting equations are in agreement with those obtained by the linear interpolation approach. For certain cases, the predicted cooling times by the fitting equations are slightly higher.

For the 61BT DSC, borated aluminum, MMC, or Boral® shall be supplied in accordance with UFSAR Sections K.9.1.7.1, K.9.1.7.2, K.9.1.7.3, K.9.1.7.4, portions of Section K.9.1.7.7, portions of Section K.9.1.7.8.4, and all of Sections K.9.1.7.8.5, K.9.1.7.9.1, and K.9.1.7.9.2, with the minimum B-10 areal density specified in Table 1-1k. These sections of the UFSAR are hereby incorporated into the NUHOMS® 1004 CoC.

The NUHOMS®-61BTH DSC is designed for unirradiated fuel with a maximum lattice average enrichment of 5.0 wt. % U-235 as shown in Table 1-1t, taking credit for the boron content in the poison plates of the DSC basket, as shown in Table 1-1v for intact fuel, Table 1-1w and Table 1-1x for damaged fuel, and Table 1-1w1 for failed fuel. The NUHOMS®-61BTH DSC (similar to 61BT DSC) is designated as Type 1 and Type 2 depending upon the rails used in the basket.

Each 61BTH DSC type is provided with six alternate basket configurations, based on the boron content in the poison plates, as listed in Table 1-1v or Table 1-1w or Table 1-1w1 (designated as "A" for the lowest B-10 loading to "F" for the highest B-10 loading).

For the 61BTH DSC, borated aluminum, MMC, or Boral® shall be supplied in accordance with UFSAR Sections T.9.1.7.1, T.9.1.7.2, T.9.1.7.3, T.9.1.7.4, portions of Section T.9.1.7.7, portions of Section T.9.1.7.8.4, and all of Sections T.9.1.7.8.5, T.9.1.7.9.1, and T.9.1.7.9.2, with the minimum B-10 areal density specified in Table 1-1v or Table 1-1w. These sections of the UFSAR are hereby incorporated into the NUHOMS® 1004 CoC.

The NUHOMS®-69BTH DSC is designed for unirradiated fuel with a maximum lattice average enrichment of 5.0 wt. % U-235 as shown in Table 1-1gg. Credit is taken for the boron content in the poison plates of the DSC basket, as shown in Table 1-1jj for intact and Table 1-1kk for damaged fuel. The NUHOMS®-69BTH DSC is provided with six alternate basket configurations, based on the boron content in the poison plates, as listed in Table 1-1jj or Table 1-1kk (designated as "A" for the lowest B-10 loading to "F" for the highest B-10 loading).

For the 69BTH DSC, borated aluminum, MMC, or Boral® shall be supplied in accordance with UFSAR Sections Y.9.1.7.1, Y.9.1.7.2, Y.9.1.7.3, Y.9.1.7.4, portions of Section Y.9.1.7.7, portions of Section Y.9.1.7.8.4, and all of Sections Y.9.1.7.8.5, Y.9.1.7.9.1, and Y.9.1.7.9.2, with the minimum B-10 areal density specified in Table 1-1jj or Table 1-1kk. These sections of the UFSAR are hereby incorporated into the NUHOMS® 1004 CoC.

The NUHOMS®-32PT is designed for unirradiated fuel with an initial fuel enrichment of up to 5.0 wt. % U-235 as shown in Table *1-1e*, taking credit for poison rod assemblies (PRAs), poison plates, and soluble boron in the DSC cavity water during loading operations *as shown in Table 1-1g and Table 1-1g1 for intact fuel, Table 1-1g2 for damaged fuel and Table 1-1g3 for damaged or failed fuel. The PRAs are designated as B4C PRAs or AIC PRAs based on the absorber material as specified in Table 1-1h. Intact fuel maybe loaded in the 32PT DSC basket with no PRAs or with 4, 8 or 16 PRAs as shown in Figure 1-5, Figure 1-6 or Figure 1-7. Each of these configurations represents a basket type. Additional basket types are based on the number of poison plates per basket (16, 24 or 32) as shown in Table 1-1g and Table 1-1g1 and fixed poison loading in the poison plates (three types) as shown in Table 1-1h.*. A 32PT DSC basket may contain 0, 4, 8 or 16 PRAs

beyond those calculated for the 24P, 24PHB, 52B, 61BT, 32PT, 24PTH, 61BTH, or 32PTH1, 69BTH, or 37PTH canister full of design basis fuel assemblies with or without CCs.

Technical Specification 1.1 defines INTACT FUEL ASSEMBLY as an assembly containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pin hole leaks. Non-cladding material damage is acceptable to the extent that the fuel assembly can be handled by normal means and the fuel assembly is retrievable after all normal and off-normal conditions. This is applicable to fuel assemblies to be loaded in the 24P, 24PHB, 52B, 61BT, 32PT, 24PTH, 61BTH, 32PTH1, 69BTH or 37PTH DSCs. The bases for this definition is that the criticality and confinement functions are maintained under normal, off-normal and accident conditions of storage. The criticality analyses documented for these DSCs considers that the fuel assembly geometry remains unchanged under accident conditions and sub-criticality is assured.

Technical Specification Table 1-1a, Table 1-1b, Table 1-1c, Table 1-1j, Table 1-1e, Table 1-1i, Table 1-1l, Table 1-1t, Table 1-1aa ,Table 1-1gg, and Table 1-1ll provide the key fuel parameters that require confirmation prior to loading fuel assemblies within a specific standardized DSC model. *Definitions for DAMAGED and FAILED FUEL ASSEMBLIES are provided for each DSC, as applicable in these tables.* Each of these Technical Specification Tables lists additional Technical Specification Tables and Figures which provide requirements which also must be met prior to loading.

FUNCTIONAL AND OPERATING LIMITS VIOLATIONS
If Functional and Operating Limits are violated, the limitations on the fuel assemblies in the canister have not been met. Actions must be taken to place the affected fuel assemblies in a safe condition. This safe condition may be established by returning the affected fuel assemblies to the spent fuel pool. However, it is acceptable for the affected fuel assemblies to remain in the canister if that is determined to be a safe condition.

For the TC/DSC Lifting Heights as a Function of Low Temperature and Location, the basis for the low temperature and height limits is ANSI N14.6-1986 and -1993 paragraph 4.2.6, which requires at least 40 °F higher service temperature than nil ductility transition (NDT) temperature for the TC. In the case of the standardized TC, the test temperature is -40 °F; therefore, although the NDT temperature is not determined, the material will have the required 40 °F margin if the ambient temperature is 0 °F or higher. This assumes the material service temperature is equal to the ambient temperature.

The basis for the low temperature limit for the DSC is NUREG/CR-1815. The basis for the handling height limits is the NRC evaluation of the structural integrity of the DSC to drop heights of 80 inches and less.

For the NUHOMS®-24P, 52B and 61BT *Systems*, the basis for the high temperature limit is PNL-6189 for the fuel clad limit, the manufacturer's specification for neutron shield, and the design basis pressure of the TC internal cavity pressure. For the NUHOMS®-32PT, 24PHB and 24PTH Systems, the fuel cladding limits are based on ISG-11, Revision 2. For the NUHOMS®-61BTH *System* and the NUHOMS®-61BT *System* with FANP 9x9-2 fuel assemblies, the fuel cladding limits are based on ISG-11 Revision 3.

For the NUHOMS®-69BTH and 37PTH Systems, the fuel cladding limits are based on NUREG-1536, Revision 1.

For the TC/DSC Transfer at High Ambient Temperatures (32PTH1 DSC Only), the fuel cladding limits are based on ISG-11 Revision 3 (Reference 4).

The basis for using a solar shield during transfer operations when ambient temperatures exceed 100 °F (106 °F for 32PTH1) is to prevent direct solar radiation on the exterior surface of the cask and to ensure that component temperatures remain below the allowable limits described in the UFSAR. With the solar shield in place, the temperatures on the exterior surface of the cask remain below those evaluated in the UFSAR.

The solar shield while not required for transfer operations with ambient temperatures below 100 °F (106 °F for 32PTH1) may also be used to protect the cask from rain or snow or other

B 10.5.3.2 Cask Drop

BASES

The basis for this specification is Section 8.2.5, "Accidental Cask Drop."

B 10.5.3.3 TRANSFER CASK Alignment with HSM or HSM-H

BASES

The basis for the true position alignment tolerance is the clearance between the DSC shell, the transfer cask cavity, the HSM or HSM-H/HSM-HS access opening, and the DSC support rails inside the HSM or HSM-H.

12.2.3 Horizontal Storage Module Concrete and Dry Shielded Canister Steel Support Structure Thermal Fatigue, Corrosion, and Temperature Effects Evaluation

12.2.3.1 Summary Description

This TLAA evaluates time-dependent aging mechanisms associated with potential degradation of the HSM component. The HSMs subject to these evaluations are the *Standardized HSM (*HSM Models 80, 102, 152*, and 202)*, and HSM-H (*including the high seismic option designated as* HSM-HS). The HSM-HS and HSM Model 202 are the same as the HSM-H for purposes of these evaluations. The evaluations presented herein are bounding for the six HSM types. All the storage modules are identified as HSM in these evaluations unless the specific HSM type is noted.

The effect of radiation on the HSM materials is evaluated in Section 12.2.5 for a service life of **[ ]**. The evaluation concludes that there are no adverse effects on the HSM structure due to neutron fluence or gamma radiation over the period of extended operation.

The HSM is evaluated for sustained elevated temperature effects, thermal fatigue, and corrosion of the reinforcing steel.

The DSC support structure is evaluated for the effects of corrosion and thermal fatigue. The sliding surfaces of the DSC support structure are fabricated from NITRONIC® 60 austenitic stainless steel, which is coated with a dry film lubricant to minimize friction during insertion and retrieval of the DSC. The effect of radiation is evaluated for the lubricant.

The heat shields are designed to limit concrete temperature to within acceptable limits. The heat shields are evaluated for corrosion effects.

12.2.3.2 Conclusions

This time-limited aging analysis evaluated the effects of temperature, thermal cycling fatigue, and corrosion on the HSM components. The evaluation conclusions are as follows:

- Degradation due to elevated temperature is not an aging effect requiring management for the HSM concrete. The long-term temperatures corresponding to maximum design basis heat loads trend toward the allowed long-term temperature limits of the ACI 349 Code.
- Thermal fatigue is not an aging effect requiring management for the HSM concrete. A fatigue usage factor of 0.25 due to daily and seasonal temperature fluctuations is conservatively calculated. Thermal cycling fatigue is not considered an aging effect requiring management.
- Even though environmental degradation of the HSM concrete due to rebar corrosion is not expected to occur, corrosion of rebar is considered an aging effect requiring management for the HSM concrete.
- Loss of material due to general corrosion of carbon steel and crevice and pitting corrosion of stainless steel DSC steel support structure is considered an aging effect requiring management for the HSM DSC steel support structure.

used in lieu of the confinement evaluation by the general licensee, although the inputs utilized in this appendix should be evaluated for applicability by each site. A site-specific confinement evaluation may be performed by the general licensee to modify key input parameters in order to meet the dose limits specified in 10 CFR 72.104(a) and 10 CFR 72.106(b).

12.2.7 Thermal Performance of Horizontal Storage Modules for the Period of Extended Operation

12.2.7.1 Summary Description

This appendix evaluates the confinement weld temperatures on the outer surface of the NUHOMS® dry shielded canisters (DSCs) and the concrete temperatures over the extended period of storage as input to aging management reviews (AMRs) and time limited aging analyses (TLAAs) for the DSC and the horizontal storage module (HSM). The DSCs subject to this evaluation are stored in one of the *following* storage module types; Standardized HSM (HSM Models 80, *102, 152, or 202*), *or*, HSM-H (*including the high seismic option designated as* HSM-HS). All storage modules are identified as HSM in this evaluation unless the specific type of the HSM is noted. Studies of chloride-induced stress corrosion cracking (CISCC) show that stainless steel welds are susceptible to CISSC when the weld temperature is in the range of 30 °C to 80 °C.

To evaluate the confinement weld temperatures various heat loads at steady state condition and average annual temperature of 70 °F are considered in the models. The evaluated heat loads are in the range of 2 kW to 22 kW for DSCs loaded in the Standardized HSM (HSM *Models 80, 102, 152 or 202*) or HSM Model 152 and in the range of 2 kW to 32 kW for DSCs loaded in the HSM-H *(including the high seismic option designated as HSM-HS).*

[

]

For these evaluations, half symmetric, three-dimensional computational fluid dynamics (CFD) models generated in ANSYS FLUENT [12.11] are used to simulate the air flow and heat transfer within the each HSM type separately. These evaluations also provide the DSC confinement weld temperature as a function of the heat load and storage time for a DSC during long-term storage time.

12.2.7.2 Conclusions

This TLAA provides the canister and concrete temperature ranges during extended storage, to be used as input for the stress corrosion cracking AMR [12.19, Appendix 5B] and the concrete TLAA (Section 12.2.3).

8 7 6 5 4 3 2 1

H G F E D C B A

Proprietary and Security Related Information for Drawing NUH-61B-1064-SAR, Rev. 8 Withheld Pursuant to 10 CFR 2.390

H G F E D C B A

8 7 6 5 4 3 2 1

K.2.1 Spent Fuel To Be Stored

The NUHOMS®-61BT DSC is designed to store 61 intact, or up to 16 damaged and the remainder intact, for a total of 61, standard BWR fuel assemblies with or without fuel channels. The NUHOMS®-61BT DSC can store intact BWR fuel assemblies with the characteristics described in Table K.2-1, or damaged and intact BWR fuel assemblies with the characteristics described in Table K.2-2, which include a variety of cooling times, enrichment and maximum bundle average burnup. Damaged BWR fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. *The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means.* Missing cladding and/or crack size in the fuel pins is to be limited such that a fuel pellet is not able to pass through the gap created by the cladding opening during handling and retrievability is assured following Normal/Off-Normal conditions.

The NUHOMS® -61BT DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

The NUHOMS®-61BT DSC may store BWR fuel assemblies with a maximum decay heat of 300 watts/assembly, or a total of 18.3 kW. The NUHOMS®-61BT DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight with channel is 705 lbs.

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, heat load and confinement. The fuel assemblies considered are listed in Table K.2-3. It was determined that the GE 7x7 is the enveloping fuel design for the shielding source term calculation. However, for criticality safety, the GE 10x10 assembly is the most reactive, and is evaluated for configurations that bound all normal, off-normal and accident conditions.

The NUHOMS®-61BT DSC has three basket configurations, based on the boron content in the poison plates. The maximum lattice average enrichment authorized for Type A, B and C NUHOMS®-61BT DSCs is 3.7, 4.1 and 4.4 weight percent (wt. %) U-235, respectively.

Intact BWR fuel assemblies may be stored in any of the three NUHOMS®-61BT DSC Types provided the loading meets the maximum lattice average enrichment limit for the NUHOMS®-61BT DSC type, as given on Table K.2-4. Damaged BWR fuel assemblies may only be stored in Type C NUHOMS®-61BT DSCs with endcaps installed on each four compartment assembly where a damaged fuel assembly is stored.

Fuel assemblies with various combinations of burnup, enrichment and cooling time can be stored in the NUHOMS®-61BT DSC as long as the fuel assembly parameters fall within the design limits specified in Table K.2-1 or Table K.2-2, and Table K.2-4. A simplified approach, using

Table K.2-2
BWR Fuel Specification of Damaged Fuel to be Stored in the Standardized NUHOMS®-61BT DSC

PHYSICAL PARAMETERS: Fuel Design:	7x7, 8x8 BWR damaged fuel assemblies manufactured by General Electric or Exxon/ANF or equivalent reload fuel that are enveloped by the Fuel assembly design characteristics listed in Table K.2-3 for the 7x7 and 8x8 designs only. Damaged fuel assemblies beyond the definition contained below are not authorized for storage.
Cladding Material:	Zircaloy
Fuel Damage:	Damaged BWR fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks, *or assemblies with minor damage to spacer grids as long as the assembly is able to be handled by normal means.* Missing cladding and/or crack size in the fuel pins is to be limited such that a fuel pellet is not able to pass through the gap created by the cladding opening during handling and retrievability is assured following Normal/Off-Normal conditions. Damaged fuel shall be stored with Top and Bottom Caps. Damaged fuel may only be stored in the 2x2 compartments of the "Type C" NUHOMS®-61BT Canister described in Table K.2-4.
Channels:	Fuel may be stored with or without fuel channels.
Maximum Assembly Length (unirradiated)	176.2 in
Nominal Assembly Width (excluding channels)	5.44 in
Maximum Assembly Weight	705 lbs
RADIOLOGICAL PARAMETERS(1):	No interpolation of Radiological Parameters is permitted between groups.
Group 1: Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	5-years(2)
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Assembly Average Enrichment:	2.0 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 2: Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	8-years(2)
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Assembly Average Enrichment:	2.65 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 3: Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	6.5-years(2)
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Assembly Average Enrichment:	3.38 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly

Section K.4 provides the off-normal thermal analyses for storage and transfer mode for the NUHOMS®-61BT DSC. The resulting stress intensities for the NUHOMS®-61BT are acceptable.

K.3.6.2.3 Damaged Fuel Integrity Assessment for Off-Normal Loads

The evaluation of the damaged fuel for off-normal loads is discussed in Section K.3.6.3.

K.3.6.3 Damaged Fuel Integrity Assessment for Normal and Off-Normal Loads

Per the definition in Table K.2-2, damaged BWR fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. Missing cladding and/or crack size in the fuel pins is to be limited such that a fuel pellet is not able to pass through the gap created by the cladding opening during handling and retrievability is assured following Normal/Off-Normal conditions.

This section summarizes the evaluations performed to demonstrate structural integrity of the damaged fuel under normal and off-normal operations loads. The evaluations consider the effects of cladding defect size, cladding rupture geometry, and reduced cladding thickness due to oxidation effects.

The structural analysis documented in this section conservatively evaluates a limiting configuration with a single rod and the spacer grids in designated locations without any support from fuel compartment to provide assurance of limiting additional cladding damage. The changes to fuel assembly configuration do not have impact on retrievability due to damage to spacer grids as long as the assembly is able to be handled by normal means and the retrievability is assured following the normal and off-normal conditions. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability. The criticality analysis documented in Section K.6.4 also considers the impact of damage to the fuel assembly that includes missing and damaged grid spacers, which results in limiting the enrichment of these fuel assemblies. Therefore, additional configurations for damaged grid spacers are not evaluated herein. Licensees can perform specific evaluations to demonstrate retrievability using actual configurations.

Normal operation loads for storage conditions include stresses due to dead weight, thermal, and handling loads resulting from DSC fuel loading/unloading, fuel transfer to the ISFSI, and DSC insertion/retrieval from the HSM. These handling/transfer operations are performed slowly by trained operations personnel and follow detailed procedures. The applicable off-normal load for storage conditions is the off-normal handling load (i.e. jammed canister condition).

Because the 61BT DSC is a dual-purpose canister, it has been evaluated for loads that bound the Part72 normal and off-normal storage conditions. The fracture mechanics evaluations are also based on loads that bound the storage conditions.

Both linear-elastic stress analysis and linear-elastic fracture mechanics methods are employed to evaluate the integrity of the fuel cladding. BWR fuel with 7x7 and 8x8 arrays is considered. Table K.3.6-5 shows a summary of the fuel characteristics and design parameters used in these

evaluations. A cladding thickness reduction of 200 μm has been conservatively assumed in the structural integrity evaluations to account for water side and inner surface oxidation.

The linear elastic stress analyses use basic stress equations, conservation of energy principles, and fundamental kinematic relationships to calculate cladding stresses due to normal and off-normal loads. The handling/transfer loads produce the controlling stresses from normal operation loads. The controlling off-normal load is the jammed canister load. The computed maximum stresses for the controlling loads are summarized in Table K.3.6-6. The computed maximum stresses are compared to the irradiated cladding yield stress, and a stress ratio is calculated. As shown in the table, the maximum stress ratios correspond to the hypothetical one-foot end and side drops. Substantial margins exist for all loads considered. All the stresses summarized in Table K.3.6-6 are compressive stresses with the exception of the one-foot side drop case, which produces tension stresses due to bending.

29. Visually inspect the lifting hooks or the yoke to insure that they are properly positioned on the trunnions.

30. Drain up to approximately 1100 gallons of water from the DSC as required to meet plant crane limits.

31. The cask should be lifted just far enough to allow the weight of the transfer cask to be distributed onto the yoke lifting hooks. Inspect the lifting hooks to insure that they are properly positioned on the trunnions.

32. Install suitable protective material onto the bottom of the transfer cask to minimize cask contamination. Move the cask to the fuel pool.

33. Prior to lowering the cask into the pool, adjust the pool water level, if necessary, to accommodate the volume of water which will be displaced by the cask during the operation.

34. Lower the cask into the fuel pool leaving the top surface of the cask approximately one foot above the surface of the pool water.

35. Fill the DSC with pool water.

36. Position the cask over the cask loading area in the fuel pool.

37. Lower the cask into the pool. As the cask is being lowered, the exterior surface of the cask should be sprayed with clean demineralized water.

38. Disengage the lifting yoke from the cask and lift the top shield plug from the DSC.

39. Remove the holddown ring. If the DSC contains damaged fuel assemblies, remove the top end caps. Remove the fuel from the DSC and place the fuel into the spent fuel racks.

40. Lower the top shield plug onto the DSC.

41. Visually verify that the top shield plug is properly positioned onto the DSC.

42. Engage the lifting yoke onto the cask trunnions.

43. Visually verify that the yoke lifting hooks are properly engaged with the cask trunnions.

44. Lift the cask by a small amount and verify that the lifting hooks are properly engaged with the trunnions.

K.9 Acceptance Tests and Maintenance Program

Background for this particular UFSAR chapter:

Beginning with CoC 1004 Amendment 10, which was incorporated into UFSAR Revision 11, Chapter K.9, "Acceptance Tests and Maintenance Program," contained information which was incorporated by reference into the Technical Specifications (TS) associated with a particular amendment. It is known that certain general licensees reconcile the CoC 1004 UFSAR revisions provided to them to their loaded systems, pursuant to 10 CFR 72.48 and 10 CFR 72.212. In doing so they sometimes find the changed UFSAR portions incorporated by reference into the TS to be impossible to reconcile because the 10 CFR 72.48 regulation does not allow proposed activities which involve changes to the TS.

In order to facilitate this reconciliation process by general licensees, the following statements are provided, addressing the licensing basis for certain amendments, as they relate to certain UFSAR chapters which contain TS incorporated by reference. Additionally, so that the actual information is contained in the current CoC 1004 UFSAR, to facilitate the reconciliation by general licensees, the UFSAR Revision 11, 12, 13, and 14 versions of Chapter K.9 are inserted and annotated in this part of the UFSAR. For clarity, this includes annotating the version of Chapter K.9 directly associated with the latest UFSAR revision in which a change to Chapter K.9 occurred.

- Systems loaded to CoC 1004 Amendment 10 have Technical Specifications incorporated by reference from UFSAR Revisions 11 and 12 in Chapter K.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to systems loaded to Amendment 10.
- Systems loaded to CoC 1004 Amendment 11 have Technical Specifications incorporated by reference from UFSAR Revision 13 Chapter K.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 11.
- Note that CoC 1004 Amendment 12 was submitted and docketed, associated with a U.S. Department of Energy project, but due to a lack of review funding the NRC returned it without a review.
- Systems loaded to CoC 1004 Amendment 13 have Technical Specifications incorporated by reference from UFSAR Revisions 14 and 15 Chapter K.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 13.
- Systems loaded to CoC 1004 Amendment 14 have Technical Specifications incorporated by reference from UFSAR Revisions 16 and 17 Chapter K.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 14.
- *Systems loaded to CoC 1004 Amendment 15 have Technical Specifications incorporated by reference from UFSAR Revision 18 Chapter K.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 15.*

K.9 Tests and Maintenance Program

K.9.1 Acceptance Tests

The pre-operational testing requirements for the *Standardized* NUHOMS® system are given in Section 9.0 with the exceptions described in the following sections. The NUHOMS®-61BT DSC has been enhanced to provide leaktight confinement and the basket includes an updated poison plate design. Additional acceptance testing of the NUHOMS®-61BT DSC welds and of the poison plates are described. AMD 11

K.9.1.1 Visual Inspection

No change *to Section 4.5.1.*

K.9.1.2 Structural.

The NUHOMS®-61BT DSC confinement welds are designed, fabricated, tested and inspected in accordance with ASME B&PV Code Subsection NB [9.1] with exceptions as listed in Section K.3.1. The following requirements are unique to the NUHOMS®-61BT DSC:

- The inner bottom cover weld is inspected in accordance with Article NB-5231.
- The outer bottom cover weld root and cover are penetrant tested.
- The canister shell longitudinal and circumferential welds are 100% radiographically inspected.
- The outer top cover plate weld root, middle and cover are penetrant tested.

The NUHOMS®-61BT DSC basket is designed, fabricated, and inspected in accordance with ASME B&PV Code Subsection NG [9.1] with exceptions as listed in Section K.3.1. The following requirements are unique to the NUHOMS®-61BT DSC:

- The fuel compartment wrapper welds are inspected in accordance with Article NG-5231.
- The fuel compartment welds are inspected in accordance with Article NG-5231.

K.9.1.3 Leak Tests

The NUHOMS®-61BT DSC confinement is leak tested to verify it is leaktight in accordance with ANSI N14.5 [9.2].

The leak tests are typically performed using the helium mass spectrometer method. Alternative methods are acceptable, provided that the required sensitivity is achieved.

K.9.1.4 Components

The Standardized NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. No other components of the Standardized NUHOMS® system require testing, except as discussed in this Appendix.

K.9.1.5 Shielding Integrity

No change to Section 4.3.9 and Appendix U, Section U.9.1.5.

K.9.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section K.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section K.9.1.7.6.

K.9.1.7 Poison Acceptance

CAUTION

Sections K.9.1.7.1 through K.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 1) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated aluminum

(b) Boron carbide / aluminum metal matrix composite (MMC)

(c) BORAL®

The 61BT DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content of these three types of materials is given in Table K.9-1.

References to metal matrix composites throughout this appendix are not intended to refer to BORAL®, which is described later in this section.

K.9.1.7.1 Borated Aluminum

See the Caution in Section K.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating primarily as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy (other boron compounds, such as AlB_{12}, can also occur). For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section K.9.1.7.7. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

K.9.1.7.2 Boron Carbide / Aluminum Metal Matrix Composites (MMCs)

See the Caution in Section K.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, *molten metal infiltration* or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding *or produced by molten metal infiltration* shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B_4C particles in MMCs shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 61BT DSC, MMCs shall pass the qualification testing specified in Section K.9.1.7.8, and shall subsequently be subject to the process controls specified in Section K.9.1.7.9.

All changes on this page are AMD 13

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section K.9.1.7.7. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

K.9.1.7.3 **BORAL®**

See the Caution in Section K.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns. The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of BORAL®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken from the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

K.9.1.7.4 **Visual Inspections of Neutron Absorbers**

See the Caution in Section K.9.1.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. ***Blisters shall be treated as non-conforming. For clad MMCs and for BORAL®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet. Material that does not meet these criteria shall be reworked, repaired, or scrapped.***

K.9.1.7.5 Other Visual Inspections Criteria (non-Technical Specifications)

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

K.9.1.7.6 Thermal Conductivity Testing

Acceptance testing shall conform to ASTM E1225[1], ASTM E1461[2], or equivalent method, performed at roomtemperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, and may be reduced if the first five tests meet the specified minimum thermal conductivity. For cast products, the lot shall be defined by the heat or ingot. For other products, the lot shall be defined as material produced in a single production campaign using the same heat or lots of aluminum and boron carbide feed materials.

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The measured thermal conductivity values shall satisfy the minimum required conductivities as specified in Section K.4.3.

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section K.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

K.9.1.7.7 Specification for Acceptance Testing of Neutron Absorber *Content*

Acceptance testing for Neutron Absorber content shall be performed by either neutron transmission or by B-10 volume density measurement.

CAUTION

Portions of Section K.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 1) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

[1] ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique."

[2] ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method."

K.9.1.7.7.1 *Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission*

a) Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

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The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam up to 1.1 inch diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard.

Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than the statistically derived minimum thickness from K.9.1.7.7 a) or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing

operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

K.9.1.7.7.2 *Specification for Acceptance Testing of Neutron Absorbers by B-10 Volume Density Measurement*

a) B-10 volume density measurement acceptance testing procedures shall be subject to approval by the certificate holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, as long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for B-10 volume density measurements shall be such that there is at least one density measurement for each 2000 square inches of final product in each lot.

Areal density is determined by measuring the B-10 volume density in test samples and converting the measured values to areal density. The method of measurement of B-10 volume density shall be subject to approval by the certificate holder. The method of measurement of B-10 volume density shall be qualified against neutron transmission testing. Results of the two test methods shall be compared and a penalty shall be derived to account for the performance based results of neutron transmission testing.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B-10 areal densities are determined by volume density as described above. The lower tolerance limit of B-10 volume density is then determined, defined as the mean value of B-10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6]. Finally, the minimum specified value of B-10 areal density is divided by the lower tolerance limit of B-10

volume density to arrive at the minimum plate thickness that provides the specified B-10 areal density.

Any plate that is thinner than the statistically derived minimum thickness from K.9.1.7.7.2 a) or the minimum design thickness, whichever is greater, shall be treated as nonconforming, with the following exception. Local depressions are acceptable, as long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

K.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

CAUTION

Portions of Section K.9.1.7.8.4 and Section K.9.1.7.8.5 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 1) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

K.9.1.7.8.1 Applicability and Scope

MMCs acceptable for use in the 61BT DSC are described in Section K.9.1.7.2.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section K.9.1.7.9 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the Certificate Holder.

K.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in Section K.9.1.7.8.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section K.9.1.7.8.5.

K.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for unclad MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below

842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for MMCs (clad or unclad) consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

K.9.1.7.8.4 Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity

At least three samples, one each from approximately the two ends and middle of the qualification material run shall be subject to:

a) **room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:**

- **Minimum yield strength, 0.2% offset: 1.5 ksi**
- **Minimum ultimate strength: 5 ksi**
- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

b) **Testing to verify more than 98% of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a means to be approved by the Certificate Holder. The maximum interconnected porosity is 0.5 volume %.**

Delamination Testing of Clad MMC

c) ***Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage.*** *An example of such a test would be: (1) immerse a specimen at least 6 x 6 inches in water under pressure ≥30 psig for at least 24 hours, (2) place the specimen in a vacuum furnace preheated to at least 300°F and evacuate the furnace. Acceptance criterion: no blistering or delamination of the cladding.*

K.9.1.7.8.5 Required Tests and Examinations to Demonstrate B10 Uniformity

3 Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998

4 Boralyn testing submitted to the NRC under docket 71-1027, 1998

5 ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products

6 ASTM E290, Standard Methods for Bend Testing of Materials for Ductility

Uniformity of the boron distribution shall be verified either by:

a) Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

b) Quantitative testing for the B10 areal density, B10 density, the boron carbide weight fraction, *or the boron weight fraction,* on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section K.9.1.7.7, or by chemical analysis for boron carbide *or boron* content in the composite.

K.9.1.7.8.6 Qualification Report

Qualification report shall be prepared by, or subject to approval by the Certificate Holder.

K.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION

Sections K.9.1.7.9.1 and K.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications (paragraph 4.1 ((Note 1)) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

K.9.1.7.9.1 **Applicability and Scope**

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section K.9.1.7.8 is required, depending on the characteristics of the material that could be affected by the process change.

K.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, reduce corrosion resistance, reduce the mechanical strength or ductility of the MMC.

K.9.1.7.9.3 Identification and Control of Key Process Changes

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section K.9.1.7.9.2.

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7 ASTM E94, Recommended Practice for Radiographic Testing

8 ASTM E142, Controlling Quality of Radiographic Testing

9 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that are established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average *(d50)* particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a magnesium-alloyed aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, and

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

K.9.2 Maintenance Program

NUHOMS®-61BT system is a totally passive system and therefore will require little, if any, maintenance over the lifetime of the ISFSI. Typical NUHOMS®-61BT system maintenance tasks will be performed in accordance with Section 4.

K.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition including 1999 addenda.

9.2 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.3 *Deleted.*

9.4 *Deleted.*

9.5 "Aluminum Standards and Data, 2003" The Aluminum Association.

9.6 Natrella, "Experimental Statistics," Dover, 2005.

9.7 *Deleted.*

9.8 *Deleted.*

9.9 *Deleted.*

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Table K.9-1
B10 Specification for the NUHOMS® 61BT Poison Plates

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Basket Type	*Specified Minimum B10 Areal Density for Borated Aluminum/MMC for 90% credit (g/cm²)*	*Specified Minimum B10 Areal Density for BORAL® for 75% credit (g/cm²)*
Type 1 DSC		
A	*0.021*	*0.025*
B	*0.032*	*0.038*
C	*0.040*	*0.048*
For Damaged Fuel		
C	*0.040*	*0.048*

This version of Chapter K.9 is associated with CoC 1004 Amendment 15 and is from UFSAR Revision 18. Please see Page K.9 Introduction - 1 for a discussion as to why certain versions of Chapter K.9 are being maintained in the UFSAR.

Table K.9-2
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All changes on this page are AMD 11

This version of Chapter K.9 is associated with CoC 1004 Amendment 15 and is from UFSAR Revision 18. Please see Page K.9 Introduction - 1 for a discussion as to why certain versions of Chapter K.9 are being maintained in the UFSAR.

Table K.9-3
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M.1. General Discussion

This Appendix M to the NUHOMS® Updated Final Safety Analysis Report (UFSAR) addresses the Important to Safety aspects of storing spent fuel including high burnup fuel (up to *62* GWd/MTU)in the NUHOMS®-32PT system.

The NUHOMS®-32PT System is designed to accommodate up to 32 intact, up to 28 damaged, or up to 8 failed fuel cans, with characteristics as described in Chapter M.2.

The NUHOMS®-32PT system consists of a NUHOMS®-32PT Dry Shielded Canister (DSC) stored in a Model 80 or Model 102 or Model 152 or Model 202 NUHOMS® Horizontal Storage Module (HSM). The NUHOMS® HSM Models 152 and 202 are described and evaluated in Appendix R and Appendix V, respectively. In addition, an upgraded version of the HSM, designated as HSM-HS, is also provided to allow storage of the NUHOMS®-32PT DSC in locations where higher seismic levels exist. The HSM-HS design configuration, described in Appendix U.1, is modified to accommodate the smaller diameter of the NUHOMS®-32PT DSC.

The NUHOMS®-32PT DSC is transferred in an OS197 or OS197H Transfer Cask (TC). The NUHOMS®-32PT is also qualified for transfer in an OS197L TC, within certain limitations, as described in Appendix W.1. Finally, the NUHOMS®-32PT DSC is also transferred in a modified version of the OS200 TC described in UFSAR Appendix U.1. The OS200 TC is fitted with an aluminum sleeve and a spacer to accommodate the smaller diameter of the 32PT DSC.

The format of this Appendix follows the guidance provided in NRC Regulatory Guide 3.61 [1.1]. The analysis presented in this Appendix shows that the NUHOMS®-32PT system meets all the requirements of 10CFR72 [1.1]. A separate analysis will be submitted to address the safety related aspects of transporting spent fuel in the NUHOMS®-32PT DSC in accordance with 10CFR71 [1.3].

The NUHOMS®-32PT system provides confinement, shielding, criticality control and passive heat removal independent of any other facility structures or components. The NUHOMS®-32PT DSC also maintains structural integrity of the fuel during storage.

Note: References to sections or chapters within this Appendix are identified with a prefix M (e.g., Section M.2.3 or Appendix M.2 or Chapter M.2). References to sections or chapters of the UFSAR outside of this Appendix (main body of the UFSAR) are identified with the applicable UFSAR section or chapter number (e.g., Section 2.3 or Chapter 2).

Aging Management Program Requirements

AMP requirements for use of the 32PT System during the period of extended storage operations are contained in Section 12.3. Applicable TLAAs performed for the initial CoC 1004 renewal application are provided in Section 12.2.

M.1.1 Introduction

The NUHOMS® System provides a modular canister based spent fuel storage and transport system. The system includes DSCs, HSMs, and the TC.

This Appendix M provides the supporting safety analysis for the addition of the 32PT DSC system. Only those features that are being revised or added to the NUHOMS® System are addressed and evaluated in this Appendix. The HSM and TC designs remain unchanged. The NUHOMS®-32PT DSC is similar to the existing 24P DSCs with the following exceptions:

- The basket has a capability to store 32, rather than 24, Pressurized Water Reactor (PWR) fuel assemblies.
- The canister shell thickness is reduced from 0.625 inches to 0.5 inches.
- The canister has been upgraded to provide a leak tight confinement.
- The basket represents a new design.
- The canister shell length and the thickness of the top and bottom end closure assemblies have been modified to accommodate the new basket design and the revised payload.

The NUHOMS®-32PT DSC system is designed to store intact *and/or damaged and/or failed* standard PWR fuel assemblies with or without Control Components (CCs). The NUHOMS®-32PT DSC system is designed for a maximum heat load of 24 kW/canister and a maximum of *2.2* kW/assembly when heat load zoning is considered. The fuel which may be stored in the NUHOMS®-32PT DSC is presented in Section M.2.

Provisions have been made for storage of up to 28 damaged fuel assemblies in lieu of an equal number of intact assemblies in cells, other than the four cells located at the center of the 32PT basket as described in Section M.2. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

Provisions have also been made for storage of up to 8 FFCs in cells located at the outside corners compartment cells of the 32PT basket as described in Chapter M.2.

M.1.2 General Description of the NUHOMS®-32PT DSC

M.1.2.1 NUHOMS®-32PT DSC Characteristics

Each NUHOMS®-32PT DSC consists of a fuel basket and a canister body (shell, canister inner bottom and top cover plates and shield plugs). A sketch of the 32PT DSC components is shown in Figure M.1-1. A set of reference drawings is presented in Section M.1.5.

As shown in Table M.1-1, the 32PT DSC system consists of four design configurations or Types as follows:

- 32PT-S100, Short Canister (186.2 inch length)
- 32PT-L100, Long Canister (192.2 inch length)
- 32PT-S125, Short Canister (186.2 inch length)
- 32PT-L125, Long Canister (192.2 inch length)

These four design configurations allow flexibility to accommodate the payload fuel types described in Section M.2, with and without CCs. Dimensions and estimated weights of the NUHOMS®-32PT DSC are shown in Table M.1-1.

The thickness for the individual plate components of the top and bottom end cover plates has been increased to accommodate the higher internal pressure, while the top and bottom end shield plug thickness has been reduced relative to the 24P DSC configuration. The NUHOMS®-32PT DSC shell thickness is 0.50 inches instead of 0.625 inches as used for the NUHOMS®-24P or – 52B DSC designs. The materials used to fabricate the DSC are shown in the Parts List on Drawings NUH-32PT-1001-SAR, -1002-SAR, -1003-SAR, -1004-SAR, -1006-SAR, *and -1007-SAR.*

The failed fuel assemblies are to be placed in individual failed fuel cans (FFCs). Each FFC is constructed of sheet metal and is provided with a welded bottom closure and a removable top closure, which allows lifting of the FFC with the enclosed damaged assembly/debris. The FFC is provided with screens at the bottom and top to contain fuel debris and allow filling/drainage of water from the FFC during loading operations. The FFC is protected by the fuel compartment tubes and its only function is to confine the failed fuel. The FFC geometry and the materials used for its fabrication are shown on drawing NUH-32PT-1007-SAR included in Section M.1.5.

The confinement vessel for the NUHOMS®-32PT DSC consists of a shell which is a welded stainless steel cylinder with an integrally-welded, stainless steel bottom closure assembly; and a stainless steel top closure assembly, which includes the vent and drain system.

There are no penetrations through the confinement vessel. The draining and venting systems are covered by the seal welded outer top closure plate and vent and siphon port plugs. To preclude air in-leakage, the canister cavity is inerted and pressurized above atmospheric pressure with helium. The NUHOMS®-32PT DSCs are designed and tested to meet the leak tight criteria of ANSI N14.5-1997.

The basket structure consists of a grid assembly of welded stainless steel plates or tubes that make up a grid of 32 fuel compartments. Each fuel compartment accommodates aluminum and/or neutron absorbing plates (which are made of either borated aluminum or metal matrix composites such as Boralyn®, Metamic® or equivalent) that provide the necessary criticality control and heat conduction paths from the fuel assemblies to the canister shell. The space between the fuel compartment grid assembly and the perimeter of the DSC shell is bridged by transition rail structures. The transition rails are solid aluminum segments that support the fuel

compartment grid assembly and transfer mechanical loads to the DSC shell. They also provide the thermal conduction path from the basket assembly to the canister shell wall, making it efficient in rejecting heat from its payload. This method of construction forms a robust structure of compartment assemblies which provides for storage of 32 fuel assemblies. The nominal clear dimension of each fuel compartment opening is 8.7 in. x 8.7 in., which provides clearance around the fuel assemblies.

During dry storage of the spent fuel in the NUHOMS®-32PT *System*, no active systems are required for the removal and dissipation of the decay heat from the fuel. The NUHOMS®-32PT DSC is designed to transfer the decay heat from the fuel to the basket, from the basket to the canister body and ultimately to the ambient via the HSM or TC.

Each canister is identified by a Mark Number as follows: WWW32PT-XXX-YYY-ZZZ, where: XXX is the canister type designation (S100/L100/S125/L125), YYY is the basket type designation, while WWW and ZZZ are designated by TN. Each canister is also marked with the patent number. The basket type designation, YYY, consists of a letter (A, B, C, D to designate 0, 4, 8 or 16 *poison road assemblies (*PRAs*)*) and 2 numerals (16, 24*, or 32* to designate the configuration of poison plates). The Type A 32PT DSC basket (0 PRAs) consists of 3 subtypes: A, A1 or A2, depending on the *B*-10 content of the poison plates.

M.1.2.2 Operational Features

M.1.2.2.1 General Features

The NUHOMS®-32PT DSCs are designed to safely store 32 intact *and/or damaged and/or failed* PWR fuel assemblies with or without CCs. The NUHOMS®-32PT DSC is designed to maintain the fuel cladding temperature below allowable limits during storage, short-term accident conditions, short-term off-normal conditions and fuel transfer operations.

The criticality control features of the NUHOMS®-32PT DSC are designed to maintain the neutron multiplication factor k-effective less than the upper subcritical limit equal to 0.95 minus benchmarking bias and modeling bias under all conditions.

M.1.2.2.2 Sequence of Operations

The sequence of operations to be performed in loading fuel into the NUHOMS®-32PT DSCs is presented in Appendix M.8.

M.1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

M.1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry, soluble boron in spent fuel pool and by utilizing fixed neutron poison material in the fuel basket. If required, depending on fuel assembly design and initial enrichment, PRAs, as shown in Figure M.1-2 are also used for criticality control. The 32PT basket may contain 0, 4, 8 or 16 PRAs and is called a Type A, Type B, Type C, or Type D, respectively. *Two types of PRAs are specified depending on the absorber material - B_4C-PRAs or AIC-PRAs as described in Table M.2-4a.*

The Type A basket (0 PRAs) consists of 3 subtypes: A, A1 or A2, depending on the *B*-10 content of the poison plates. *The Type B basket consists of subtypes: B, B1/B2 (4 B_4C-PRAs), and B1-r/B2-r (4 AIC PRAs), depending on the B-10 content of the poison plates. The Type C basket consists of subtypes: C, C1/C2 (8 B_4C-PRAs), and C1-r/C2-r (8 AIC PRAs).* These features are only necessary during the loading and unloading operations that occur in the loading pool (underwater). However, the PRAs are left in place following the completion of the DSC draining and drying operations, which are discussed in M.8.1.3. During storage, with the DSC cavity dry and sealed from the environment, criticality

control measures within the installation are not necessary because of the low reactivity of the fuel in the dry NUHOMS®-32PT DSC and the assurance that no water can enter the DSC cavity during storage.

M.1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the NUHOMS®-32PT system.

M.1.2.2.3.3 Operation Shutdown Modes

The NUHOMS®-32PT DSC system is a totally passive system so that consideration of operation shutdown modes is unnecessary.

M.1.2.2.3.4 Instrumentation

No change to Section 5.1.3.4.

M.1.2.2.3.5 Maintenance Techniques

No change to Section 5.1.3.5.

M.1.2.3 Cask Contents

The NUHOMS®-32PT DSC system is designed to store 32 intact *and/or damaged and/or failed* PWR fuel assemblies with or without CCs. Each NUHOMS®-32PT DSC is designed for a maximum heat load of 24 kW/canister and *2*.2 kW/assembly if zoning for heat load is used. The fuel that may be stored in the NUHOMS®-32PT DSC is presented in Chapter M.2. 72.48 AMD 15

Chapter M.5 provides the shielding analysis. Chapter M.6 covers the criticality safety of the NUHOMS®-32PT DSC system and its contents, listing material densities, moderator ratios, and geometric configurations.

M.1.5 Supplemental Data

The following Transnuclear drawings are enclosed:

1. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Main Assembly, Drawing NUH-32PT-1001-SAR.

2. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Shell Assembly, Drawing NUH-32PT-1002-SAR.

3. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Basket Assembly Plate Options 1, Drawing NUH-32PT-1003-SAR.

4. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Basket Assembly Tube Options 2, Drawing NUH-32PT-1004-SAR.

5. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Aluminum Transition Rails, Drawing NUH-32PT-1006-SAR.

6. *NUHOMS®-32PT Transportable Storage Canister for PWR Fuel Failed Fuel Can Assembly, Drawing NUH32PT-1007-SAR.*

7. *NUHOMS®-32PT Transportable Canister for PWR Fuel Damaged Fuel End Caps, Drawing NUH32PT-1008-SAR.*

Proprietary and Security Related Information for Drawing NUH-32PT-1003-SAR Rev. 8 Withheld Pursuant to 10 CFR 2.390

Proprietary and Security Related Information for Drawing NUH-32PT-1004-SAR Rev. 7 Withheld Pursuant to 10 CFR 2.390

8 | 7 | 6 | 5 | 4 | 3 | 2 | 1

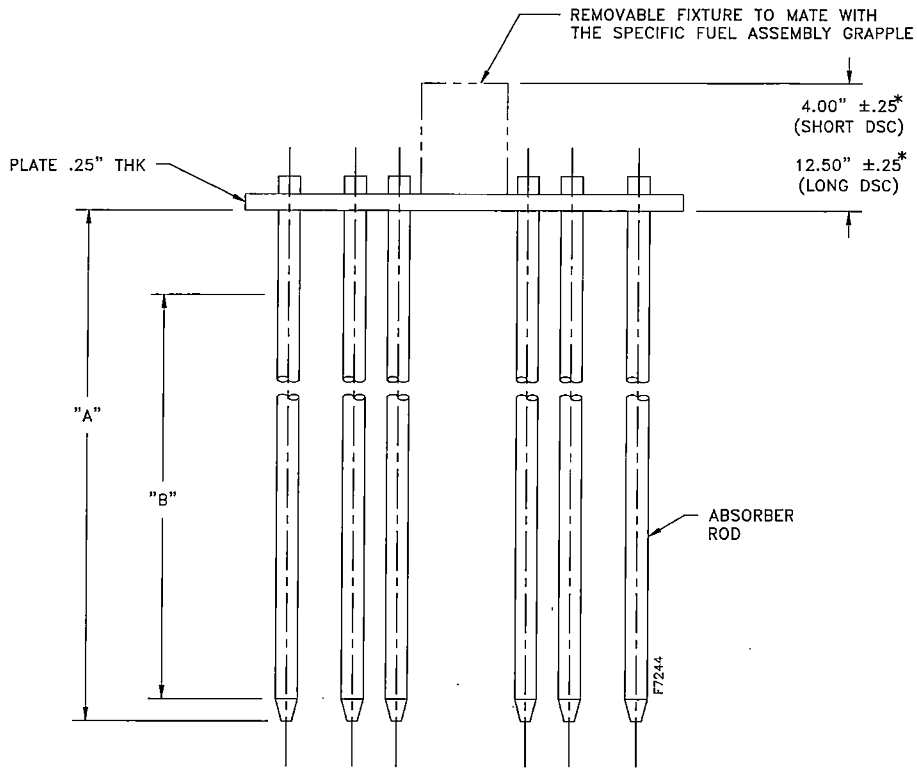
H G F E D C B A

Proprietary and Security Related Information for Drawing NUH32PT-1007-SAR Rev. 0 Withheld Pursuant to 10 CFR 2.390

H G F E D C B A

8 | 7 | 6 | 5 | 4 | 3 | 2 | 1

Proprietary and Security Related Information for Drawing NUH32PT-1008-SAR Rev. 0 Withheld Pursuant to 10 CFR 2.390



*THESE DIMENSIONS ARE FOR USE WITH WESTINGHOUSE 17x17 FUEL ASSEMBLIES. DIMENSIONS WILL VARY AS REQUIRED BY FUEL ASSEMBLY TYPE.

DIMENSION	ABSORBER TYPE					
	B_4C					AIC
	FUEL ASSEMBLY TYPE					
	WE 17x17	B&W 15x15	WE 15x15	CE 14x14	WE 14x14	WE 17x17
ABSORBER ROD OD NOMINAL (IN)	.362	.438	.450	.975	.432	.381
ABSORBER ROD DIMENSION "A"(IN)	156	160	156	143	156	156
ABSORBER STACK HEIGHT, "B" (IN)	151	151	150	129	150	150
CLAD THICKNESS NOMINAL (IN)	.018	.022	.023	.049	.022	.018
No. OF RODS	24	16	20	5	16	24
CLAD MATERIAL	304 SST	304 SST	304 SST	304 SST	304 SST	304 SST
ABSORBER PELLET NOMINAL OUTER DIAMETER (IN)	–	–	–	–	–	.341

Figure M.1-2
Poison Rod Assemblies (*B_4C-PRAs or AIC*-PRAs)

M.2.1 Spent Fuel To Be Stored

There are four design configurations for the NUHOMS®-32PT DSC, two "short" canister configurations (the 32PT-S100 and 32PT-S125), and two "long" canister configurations (the 32PT-L100 and 32PT-L125). The main difference between the –S100/-L100 and –S125/-L125 configuration designs is the thicknesses of shield plugs and DSC cover plates. The basket layout for these two configurations is identical except for the length of the components. Each of the DSC configurations is designed to store 32 intact (including reconstituted) *and/or damaged and/or failed* standard PWR high burnup fuel assemblies. *The 32PT DSC is also designed to store either up to 28 damaged fuel assemblies or 8 Failed Fuel Cans (FFCs) with the remaining locations containing intact fuel assemblies with the Type A 32PT DSC basket.* The 32PT-L100 and 32PT-L125 are also designed to store 32 intact *and/or damaged and/or failed* standard PWR fuel assemblies with or without control components (CCs). The NUHOMS®-32PT DSCs can store intact *and/or damaged and/or failed* PWR fuel assemblies and CCs with the characteristics described in Table M.2-1. The fuel to be stored is limited to a maximum assembly average enrichment of 5.0 wt. %. ^{235}U. The maximum *basket* assembly average *burnup must be less than or equal* to 55 GWd/MTU, *with the maximum permissible individual fuel assembly (FA) average burnup limited to 62 GWd/MTU.* The minimum required cooling time for fuel to be stored with *380* kgU/FA and 475 kgU/FA is explicitly specified as a function of burnup and enrichment in *Technical Specifications Tables 1-3a through 1-3p.* For fuel with a kgU/FA loading between these two values, the minimum required cooling time for fuel to be stored as a function of burnup and enrichment is determined by using the interpolation methodology specified in the notes and examples following *Technical Specifications Table 1-3p. Note that the 492 kgU tables are not applicable to the 32PT DSC.*

The CCs include *control spiders,* burnable poison rod assemblies (BPRAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), neutron source assemblies (NSAs), neutron sources, *and integral fuel burnable absorber assemblies (IFBAs).* Furthermore, non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as guide tube or instrument tube tie rods or anchors, guide tube inserts, BPRA spacer plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs. The NUHOMS®-32PT DSC may store PWR fuel assemblies arranged in any of *four* alternate heat zoning configurations with a maximum decay heat of *2.2*kW per assembly and a maximum heat load of 24 kW per canister. The heat load zoning configurations are shown in Figure M.2-1 through Figure M.2-3*a*. The NUHOMS®-32PT DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight with a CC is 1682 lb which is the same as the NUHOMS®-24P DSC design.

The maximum fuel cladding temperature limit of 400 °C (752 °F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-32PT DSC per the guidance provided in NUREG-1536, Revision 1 [2.7]. In addition, NUREG-1536 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65 °C (117 °F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570 °C (1058 °F) is applicable to accidents or off-normal thermal transients [2.7].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, heat load and confinement. These evaluations are performed in Appendices M.5, M.6, M.4 and M.7. The fuel assembly types considered are listed in Table M.2-2. It was determined that the B&W 15x15 *may be used as a representative fuel assembly for dose rate calculations.* For criticality safety, the B&W 15x15 assembly is the most reactive assembly type for a given enrichment. This assembly is used to determine the most reactive configuration in the DSC. Using this most reactive configuration, criticality analysis for all other fuel assembly classes is performed to determine the maximum enrichment allowed as a

function of *basket type and soluble boron concentration*. For thermal analysis, the WE 14x14 fuel assembly is limiting, since it results in the lowest fuel conductivity. The confinement analysis is based on B&W 15x15 fuel assembly, since it results in the smaller free volume inside the DSC cavity more than a 14x14 fuel assembly.

A 32PT DSC basket may contain 0, 4, 8, or 16 *poison rod assemblies (*PRAs*)* and is designated a Type A, Type B, Type C or Type D basket, respectively. *Two types of PRAs are specified depending on the absorber material - B4C-PRAs or AIC-PRAs as described in Table M.2-4a.* Each of these basket types has poison plates with the same minimum *B*-10 content of 0.007 gm/cm^2. Two additional basket types, designated as Type A1 and A2, have poison plates with a higher minimum *B*-10 content of 0.015 and 0.020 gm/cm^2, respectively. Type A1 and A2 basket configurations are qualified for *both the 24 poison plate and 32 poison plate basket design options. Baskets Type A1-32 and A2-32 are identical to baskets Type A1 and A2 except there are 32 poison plates with 0.015 gm/cm^2 and 0.020 gm/cm^2 minimum B-10 content respectively. Baskets Types B1-r and C1-r contain 4 and 8 silver-indium-cadmium (AIC)-PRAs, respectively, with poison plates at 0.015 gm/cm^2 minimum B-10 content while Types B2-r and C2-r may contain 4 and 8 AIC PRAs, respectively, with poison plates at 0.020 gm/cm^2 minimum B-10 content.*

The criticality analysis is based on 90% credit. The use of 90% credit is allowed because poison material coupons are to be tested via neutron transmission plus statistical analysis of the neutron transmission results. *The PRA, AIC PRA and basket types are described in Table M.2-4, M.2-4a and M.2-4b.*

Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiation and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding, and criticality evaluations for these rods are bounded by those of the regular fuel rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Reconstituted fuel assemblies with up to 56 solid stainless steel rods or unlimited number of lower enrichment UO_2 rods that replace fuel rods are acceptable for the 32PT DSC payload. CE 15x15 fuel assemblies with plugging clusters have also been evaluated.

The NUHOMS®-32PT DSCs can also accommodate up to a maximum of 28 damaged fuel assemblies placed in cells located in the DSC as shown in Figures M.2-1 through M.2-3a. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks. The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means. The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is ensured following normal and off-normal conditions. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to ensure retrievability.

The structural analysis for damaged fuel cladding described in Appendix M.3 demonstrates that the cladding does not undergo additional degradation under normal and off-normal conditions of storage. The criticality analysis described in Appendix M.6 limits the allowable contents for damaged fuel assemblies based on worst case geometry and material reconfigurations. The shielding analysis described in Appendix M.5 demonstrates that the worst configuration for damaged fuel assemblies is bounded by that of the intact fuel assemblies.

The NUHOMS®-32PT DSC is designed to accommodate up to a maximum of 8 failed fuel assemblies encapsulated in individual failed fuel cans (FFCs) and placed in cells located at the outer edge of the DSC as shown in Figure M.2-2. Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Failed fuel assemblies may contain breached rods, grossly breached rods, and other defects, such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means.

Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a rod storage basket are also considered as failed fuel. Loose fuel debris not contained in a rod storage basket may also be placed in an FFC for storage, provided the size of the debris is larger than the failed fuel can screen mesh opening and it is located at least 10 inches above the top of the bottom shield plug of the DSC.

Several different configurations with varying number of damaged fuel assemblies or failed fuel assemblies (balanced by intact fuel assemblies) are allowed for loading in the NUHOMS®-32PT DSC as shown in Figure M.2-7 through Figure M.2-9.

The NUHOMS®-32PT DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Throughout this Appendix BPRAs are considered as being representative of all CCs, unless specifically excluded.

For calculating the maximum internal pressure in the NUHOMS®-32PT DSC, it is assumed that 1% of the fuel rods are damaged for normal conditions, up to 10% of the fuel rods are damaged for off normal conditions, and 100% of the fuel rods will be damaged following a design basis accident event. A minimum of 100% of the fill gas and 30% of the fission gases (e.g., H-3, Kr and Xe) within the ruptured fuel rods are assumed to be available for release into the DSC cavity, consistent with NUREG-1536 [2.1].

The maximum design basis internal pressures for the NUHOMS®-32PT DSC are 15, 20 and 125 psig (for low burnup fuel – up to 45 GWd/MTU, 105 psig) for normal, off-normal and accident conditions of storage, respectively.

M.2.1.1 General Operating Functions

No change to Chapter 3, Section 3.1.2.

M.2.5 Summary of NUHOMS®-32PT DSC Design Criteria

The additional principal design criteria for the NUHOMS®-32PT DSC are presented in Table M.2-19. The NUHOMS®-32PT DSC is designed to store 32 intact *damaged and/or failed* PWR fuel assemblies with or without *CCs* with assembly average burnup, initial enrichment and cooling time as described in Table M.2-1. The maximum total heat generation rate of the stored fuel is limited to *2.2* kW per fuel assembly and 24 kW per NUHOMS®-32PT DSC in order to keep the maximum fuel cladding temperature below the limit [2.7] necessary to ensure cladding integrity. The fuel cladding integrity is assured by the NUHOMS®-32PT DSC and basket design which limits fuel cladding temperature and maintains a nonoxidizing environment in the cask cavity as described in Section M.4.

The NUHOMS®-32PT DSC (shell and closure) is designed and fabricated as a Class 1 component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, and the alternative provisions to the ASME Code as described in Table M.3.1-1.

The NUHOMS®-32PT DSC is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. A combination of fixed neutron absorbers, soluble boron in the pool and favorable geometry are employed to maintain the upper subcritical limit of 0.9411. The fixed neutron absorbers are in the form of borated metallic plates, *PRA,* and *AIC PRAs,* which are inserted in the guide tubes of certain assemblies in the basket. The basket is designed and fabricated in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG-3200 and the alternative provisions to the ASME Code as described in Table M.3.1-2.

The NUHOMS®-32PT DSC design, fabrication and testing are covered by TN's Quality Assurance Program, which conforms to the criteria in Subpart G of 10CFR72.

The NUHOMS®-32PT DSC is designed to withstand the effects of severe environmental conditions and natural phenomena such as earthquakes, tornadoes, lightning and floods. Section M.11 describes the NUHOMS®-32PT DSC behavior under these accident conditions.

Table M.2-1

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1e.

Table M.2-2
PWR Fuel Assembly Design Characteristics for the NUHOMS®-32PT DSC

Assembly Class	B&W 15x15	WE 17x17	CE 15x15[3], [4]	WE 15x15	CE 14x14	WE 14x14
DSC Configuration	**Maximum Unirradiated Length (in)**					
32PT-S100/32PT-S125	165.75[1]	165.75[1]	165.75	165.75[1]	165.75[1]	165.75[1]
32PT-L100/32PT-L125	171.71[1]	171.71[1]	171.71	171.71[1]	171.71[1]	171.71[1]
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Maximum MTU/assembly[2]	0.475	0.475	0.475	0.475	0.475	0.475
Maximum Number of Fuel Rods	208	264	216	204	176	179
Maximum Number of Guide/ Instrument Tubes	17	25	9	21	5	17

[1] Maximum Assembly + CC Length (unirradiated).
[2] The maximum MTU/assembly is based on the shielding analysis.
[3] CE 15x15 assemblies with stainless steel plugging clusters installed are acceptable.
[4] Control Components that extend into the active fuel region are not authorized for storage with CE 15x15 class assemblies.

Table M.2-2a

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1ee.

Table M.2-3

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1g.

Table M.2-4
Poison Rod Assembly (PRA) Description

Assembly Class	Minimum Number of Rods/PRA	Modeled B_4C Content per Rod (g/cm) (75% Credit)	Minimum B_4C Content per Rod (g/cm)
WE 17x17	24	0.59	0.79
B&W 15x15	16	0.72	0.96
WE 15x15	20	0.72	0.96
CE 14x14	5	3.14	4.19
WE 14x14	16	0.72	0.96

Table M.2-4a
AIC PRA Description

Assembly Class	*Number of Rods/AIC PRA*	*Nominal Poison Length (in.)*	*Nominal Pellet diameter (in.)*	*Minimum Required Ag Content per Rod (g/cm)* (1)
WE 17x17	*24*	*150.0*	*0.341*	*2.6 (or 6.6 g/in.)*

(1) The Ag content specified is based on a [

]

Table M.2-4*b*
B-10 Specification for the NUHOMS®-32PT Poison Plates

NUHOMS®-32PT DSC Basket Type	*Number of B_4C PRAs*	*Minimum B-10 Areal Density, gm/cm²*
A	*0*	*0.0070*
A1, A1-32	*0*	*0.0150*
A2, A2-32	*0*	*0.0200*
B	*4*	*0.0070*
B1	*4*	*0.0150*
B2	*4*	*0.0200*
C	*8*	*0.0070*
C1	*8*	*0.0150*
D	*16*	*0.0070*
NUHOMS®-32PT DSC Basket Type	*Number of AIC PRAs*	*Minimum B-10 Areal Density, gm/cm²*
B1-r	*4*	*0.0150*
B2-r	*4*	*0.0200*
C1-r	*8*	*0.0150*
C2-r	*8*	*0.0200*

Table M.2-5
Deleted

Table M.2-6
Deleted

Table M.2-7
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Table M.2-8
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Table M.2-9
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Notes: Table M.2-5 through M.2-9 Deleted

Table M.2-19
Additional Design Criteria for NUHOMS®-32PT DSC

The Gross Weight (rounded) of the NUHOMS®-32PT DSC:	
32PT-S100	88,200 (1) lbs. / 98,300 (2) lbs.
32PT-S125	90,300 (1) lbs. / 100,400(2) lbs.
32PT-L100	89,200 (1) lbs. / 99,300 (2) lbs.
32PT-L125	91,300 (1) lbs./ 101,400 (2) lbs.
Payload Capacity:	up to 32 intact *and/or damaged and/or failed* PWR assemblies (acceptable assemblies listed in Table M.2-2) and up to 32 CCs
Spent Fuel Characteristics:	See Table M.2-1 through Table M.2-3a.

(1) Based on fuel weight of 1365 lbs. per assembly.
(2) Based on fuel weight of 1682 lbs. per assembly.

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-2.

Figure M.2-1

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-3.

Figure M.2-2

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-4.

Figure M.2-3

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-4a.

Figure M.2-3a

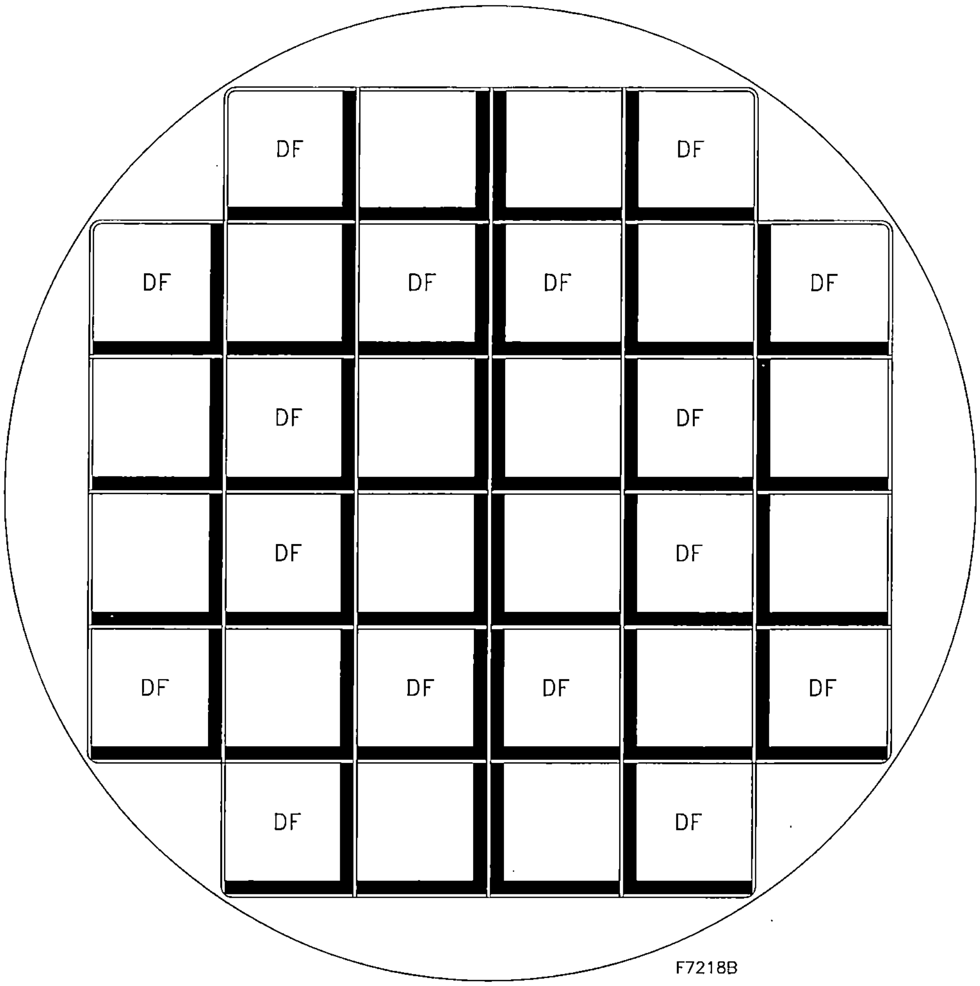

Figure M.2-7
Loading Configuration with 16 Damaged Fuel Assemblies
("DF": Damaged Assembly)

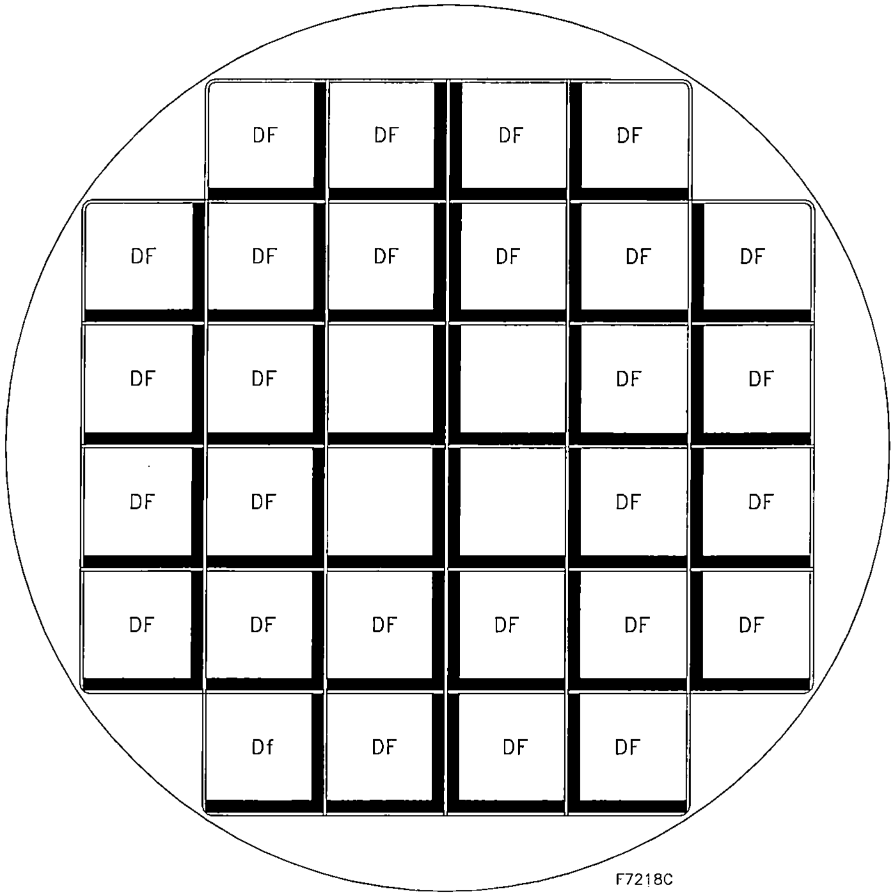

Figure M.2-8
Loading Configuration with 28 Damaged Fuel Assemblies
("DF": Damaged Assembly)

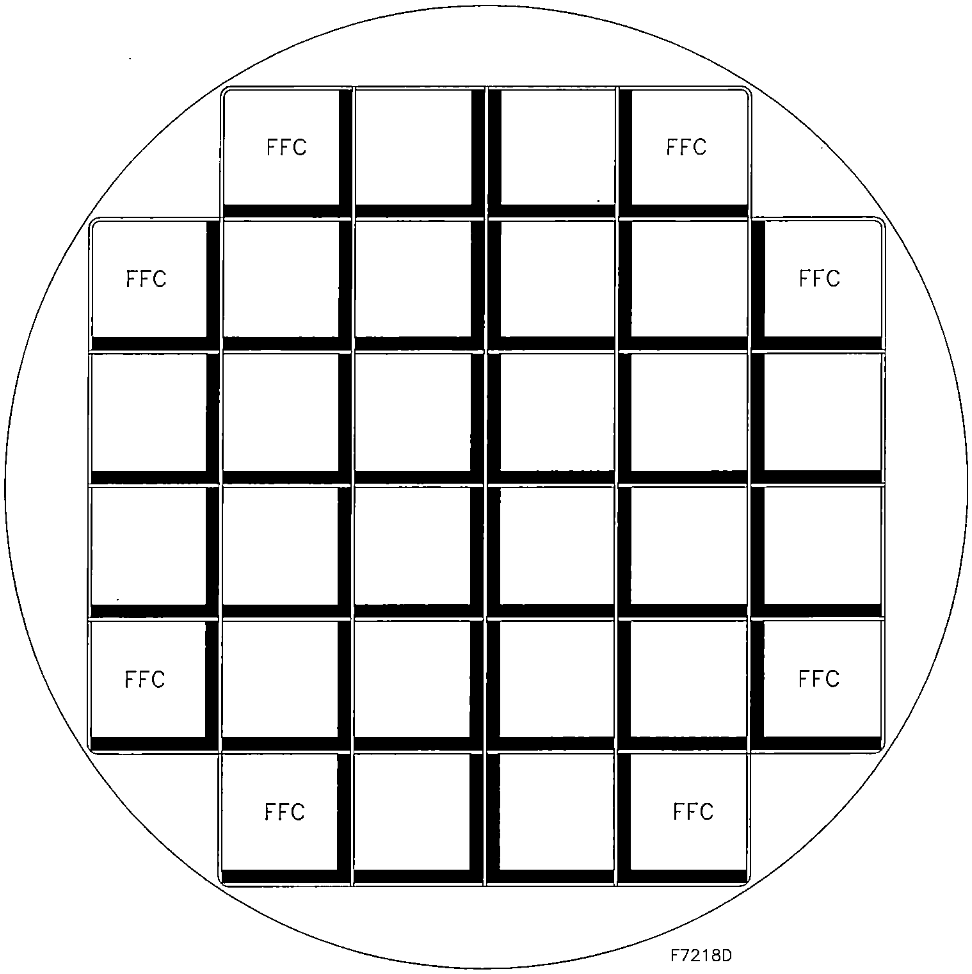

Figure M.2-9
Loading Configuration with 8 Failed Fuel Cans
("FFC": Failed Fuel Can)

Table M.3.1-1

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, Alternatives to the ASME Code for the NUHOMS-32PT DSC Confinement Boundary.

Table M.3.1-2

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, Alternatives to the ASME Code for the NUHOMS-32PT DSC Basket Assembly.

M.3.5 Fuel Rods

All the fuel assembles that are designed to be loaded into the NUHOMS® 32PT DSC are similar to those evaluated for the NUHOMS® 37PTH DSC. The geometry, model data, loads, and boundary conditions used for the 37PTH DSC are all identical and applicable for the NUHOMS® 32PT DSC. Therefore, bounding fuel side and corner drop accident conditions for the 37PTH DSC performed in Sections Z.3.5.2 and Z.3.5.3, respectively, are also applicable to the 32PT DSC. Similarly, the normal and off-normal condition damaged fuel evaluation performed in Section Z.3.6.3 for the 37PTH DSC are considered bounding for the 32PT DSC.

The 37PTH DSC uses material allowable evaluated at 713 °F for side drop evaluations. The 32PT fuel cladding is made of a zirconium based alloy material, which includes "ZIRLO" and "Optimized ZIRLO." The maximum fuel cladding temperature for the 32PT DSC is 720 °F as provided in Section M.4.9.1.2. Therefore, the side drop results from Table Z.3.5-4 (at 713 °F) are compared against the reduced allowable yield strength of 93,950 psi for the zirconium based alloy at 725 °F. Results for the 32PT DSC are shown in Table M.3.5-1.

Corner drop and damaged fuel are evaluated at 750 °F for the 37PTH DSC. This condition bounds the maximum fuel cladding temperature of 720 °F for the 32PT DSC. Therefore, evaluations performed in Sections Z.3.5.3 and Z.3.6.3 for the 37PTH DSC for corner drop are applicable to the 32PT DSC. Similarly the structural evaluations performed for damaged fuel cladding in section Z.3.6.3 for the 37PTH DSC are applicable to the 32PT DSC. Therefore, the retrievability of the damaged fuel assemblies is assured under normal and off-normal loads.

The structural analysis documented in Appendix Z.3.6.3 conservatively evaluates a limiting configuration with a single rod and the spacer grids in designated locations without any support from fuel compartment to provide assurance of limiting additional cladding damage. The changes to fuel assembly configuration do not have impact on retrievability due to damage to spacer grids, as long as the assembly is able to be handled by normal means and the retrievability is ensured following the normal and off-normal conditions. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to ensure retrievability. The criticality analysis documented in Section M.6.4 also considers the impact of damage to the fuel assembly that includes missing and damaged grid spacers, which results in limiting the enrichment of these fuel assemblies. Therefore, additional configurations are not evaluated herein. Licensees can perform specific evaluations to demonstrate retrievability using actual configurations.

Table M.3.5-1
Summary of Stress Results for Side Drop

75g Accident Load	*WE 14x14 Std/ZCA*	*WE 15x15 Std/ZC*	*CE 16x16 SCE*	*WE 17x17 OFA/LOPAR/ RFA*
Max Bending Stress, S_b (psi)	*73,255*	*70,237*	*32,308*	*70,640*
Internal Pressure (psi)	*2,235*	*2,235*	*2,235*	*2,235*
S_{press} (psi)	*11,200*	*10,271*	*8,729*	*8,381*
Combined Stress (psi)(1)	*84,455*	*80,508*	*41,037*	*79,021*
Yield Stress at 725 °F (psi)(2)	*93,950*	*93,950*	*93,950*	*93,950*

Notes:
(1) Maximum combined stress results are taken from Table Z.3.5-4.
(2) The yield strength of 93,950 psi for zircaloy cladding tube at 725 °F.

B. Vertical Deadweight

Deadweight load conditions include: (1) vertical deadweight during fuel loading operations, (2) horizontal deadweight in the TC with support through the cask rails at ±18.5°, and (3) horizontal deadweight in the HSM with support through the HSM rails at ±30°.

Under axial loads, the fuel assemblies and fuel compartment are supported by the bottom of the cask. Thus, the fuel assemblies react directly against the bottom of the canister/cask and do not load the basket structure. Stresses under axial loading are from self weight of the basket structure. Maximum axial compressive stresses occur at the supported end of the basket.

Vertical deadweight was evaluated using hand calculations and by comparing the calculated axial compression stresses to stability allowables developed considering both stability criteria and the general membrane criteria (P_m) from Subsection NG (see M.2.2.5.1.2). Calculated stresses for the vertical deadweight condition also applied to the vacuum drying case.

Calculated stresses are listed in Table M.3.6-5 along with the appropriate compressive allowables. As shown by the table, stresses for this load condition are small.

C. Horizontal Deadweight

Horizontal deadweight cases were evaluated using the ANSYS model described in M.3.6.1.3.1. As appropriate the elements representing the support rails were located at either ±18.5° or ±30° from bottom center for support by the TC or HSM, respectively. Separate analyses were performed for the 32PT fuel grid supported by the solid 1-piece or 3-piece R90 transition rails. Thus, the following four (4) analysis cases were evaluated:

1. 32PT DSC with solid 1-piece R90 transition rails supported at ±18.5° (Figure M.3.6-10 and M.3.6-11)

2. 32PT DSC with solid 3-piece R90 transition rails supported at ±18.5° (Figure M.3.6-10A and M.3.6-11A)

3. 32PT DSC with solid 1-piece R90 transition rails supported at ±30° (Figure M.3.6-12 and M.3.6-13)

4. 32PT DSC with solid 3-piece R90 transition rails supported at ± 30° (Figure M.3.6-12A and M.3.6-13A)

Primary plus secondary stresses were evaluated by combining deadweight stresses with the thermal stresses resulting from the 117 °F in cask temperature distribution. Maximum stresses are summarized in Table M.3.6-5 along with a comparison to Level A allowables from Subsection NG. *For the aluminum transition rails, reported stresses were taken at temperatures corresponding to the maximum stress on the most highly loaded rail (large R90 rail at the bottom of the basket).*

D. Vacuum Drying

As described above, the axial compression stresses under the vacuum drying condition are equal to the axial compression stresses under vertical deadweight.

As described in M.3.4.4.3, maximum stresses from the vacuum drying temperature distribution are listed in Tables M.3.4-1 and M.3.4-2. These thermal stresses are classified as secondary by the Code and, as shown by the tables, these stresses are small.

E. Handling/On-Site Transfer Loads

These cases include the loads associated with loading (and unloading) the 32PT DSC into an HSM and the inertial loads associated with on-site handling. The insertion/retrieval loads do not directly impact the 32PT basket assembly and do not require additional consideration. The inertia loads to be considered are:

- DW + 1g Axial
- DW + 1g Transverse.
- DW + 1g Vertical
- DW + 0.5g Axial + 0.5 Transverse + 0.5 Vertical

These loads are enveloped by a 2g resultant acceleration applied in the most critical orientation.

The 2.0g resultant axial load is evaluated using hand calculations and the same methodology used for the vertical deadweight analyses. Maximum compressive stresses resulting from this load case are listed in Table M.3.6-6 along with a comparison to the axial stability criteria described in M.2.2.5.1.2.

For the aluminum transition rails, reported stresses were taken at temperatures corresponding to the maximum stress on the most highly loaded rail (large R90 rail at the bottom of the basket).

Loads transverse to the axis of the DSC are evaluated using the ANSYS models described in M.3.6.1.3.1. Loads are enveloped by selection of maximum stresses from the analysis load cases listed in Table M.3.6-7. Table M.3.6-7 lists the on-site handling analyses performed for the 32PT basket considering the basket transition rail configuration and DSC support conditions in the TC and HSM. Enveloping 32PT basket stresses are summarized in Table M.3.6-6 along with a comparison to Service Level A allowables.

F. Evaluation of Results

ANSYS plots showing typical analysis results for the 32PT basket are provided in Figure M.3.6-10, Figure M.3.6-11, Figure M.3.6-12 and Figure M.3.6-13 for baskets with solid aluminum 1-piece R90 aluminum transition rails and Figure M.3.6-10A, Figure M.3.6-11A, Figure M.3.6-12A and Figure M.3.6-13A for baskets with solid aluminum 3-piece R90 transition rails. These figures and summary tables in the previous sections show that the basket stress criteria is met.

Welds in the fuel support structure are sized to maintain full moment capacity of the plates across all welded connections.

Table M.3.6-5
Summary of Results for 32PT Basket Assembly Deadweight Analyses

Vertical Deadweight

Component	Stress (Axial Compression)			Notes[(1)]
	Calculated	Allowable	Ratio	
Fuel Support Grid	.06 ksi	23.7 ksi	< .01	XM-19, 800 °F
Aluminum Transition Rails	.02 ksi	4.2 ksi	< .01	6061 Al., ≈600 °F

Notes:

1. For vacuum drying, the maximum transition rail temperature is less than 610°F. It occurs in the R90 rails and is localized at the point closest to the basket center, the average temperature is less than 600 °F.

Horizontal Deadweight

Component	Stress Category	Stress Intensity		Stress Ratio	Notes[(1)]
		Calculated	Allowable		
1-Piece R90 Transition Rails					
Fuel Support Grid	P_m	.57 ksi	28.2 ksi	.02	XM-19, 800 °F
	$P_m + P_b$	2.72 ksi	42.3 ksi	.06	
	$P_m + P_b + Q$	7.41 ksi	84.6 ksi	.09	
Transition Rail Cover Plates	P_m	.30 ksi	28.2 ksi	.01	XM-19, 800 °F
	$P_m + P_b$	2.50 ksi	42.3 ksi	.06	
	$P_m + P_b + Q$	2.50 ksi	84.6 ksi	.03	
6061 Aluminum Transition Rails	Maximum Stress	1.32 ksi	6.0 ksi	.22	Al. 6061, 450 °F
3-Piece R90 Transition Rails					
Fuel Support Grid	P_m	.70 ksi	28.2 ksi	.02	XM-19, 800 °F
	$P_m + P_b$	4.73 ksi	42.3 ksi	.11	
	$P_m + P_b + Q$	8.07 ksi	84.6 ksi	.10	
Transition Rail Cover Plates	P_m	.41 ksi	28.2 ksi	.01	XM-19, 800 °F
	$P_m + P_b$	5.32 ksi	42.3 ksi	.13	
	$P_m + P_b + Q$	5.32 ksi	84.6 ksi	.06	
6061 Aluminum Transition Rails	Maximum Stress	1.72 ksi	6.0 ksi	.29	Al. 6061, 450 °F

Note:

1. *For the steel components, stress checks were performed at the enveloping temperatures listed. For the aluminum transition rails, stress checks were performed at temperatures corresponding to the maximum stress point. Temperatures listed are for the maximum stress points of the most highly loaded rail (the large R90 transition rail at the "bottom" of the basket).*

Table M.3.6-6
Summary of Results for 32PT Basket Assembly On-Site Handling (2.0g Loads)

Vertical Handling/Seismic

Component	Stress (Axial Compression)			Notes
	Calculated	Allowable	Ratio	
Fuel Support Grid	.11 ksi	23.7 ksi	< .01	XM-19, 800°F
Aluminum Transition Rails	.04 ksi	4.2 ksi	.01	6061 Al., 600°F

Horizontal/45° Handling/Seismic

Component	Stress Category	Stress Intensity		Stress Ratio	Notes[1]
		Calculated	Allowable		
1-Piece R90 Transition Rails					
Fuel Support Grid	P_m	2.68 ksi	28.2 ksi	.10	XM-19, 800°F
	$P_m + P_b$	18.3 ksi	42.3 ksi	.43	
	$P_m + P_b + Q$	18.3 ksi	84.6 ksi	.22	
Transition Rail Cover Plates	P_m	2.19 ksi	28.2 ksi	.08	XM-19, 800°F
	$P_m + P_b$	11.1 ksi	42.3 ksi	.26	
	$P_m + P_b + Q$	11.1 ksi	84.6 ksi	.13	
6061 Aluminum Transition Rails	Maximum Stress	4.64 ksi	6.0 ksi	0.77	Al. 6061, 450°F
3-Piece R90 Transition Rails					
Fuel Support Grid	P_m	1.52ksi	28.2 ksi	.05	XM-19, 800°F
	$P_m + P_b$	20.7 ksi	42.3 ksi	.49	
	$P_m + P_b + Q$	20.7 ksi	84.6 ksi	.24	
Transition Rail Cover Plates	P_m	1.05 ksi	28.2 ksi	.04	XM-19, 800°F
	$P_m + P_b$	10.1 ksi	42.3 ksi	.24	
	$P_m + P_b + Q$	10.1 ksi	84.6 ksi	.12	
6061 Aluminum Transition Rails	Maximum stress	2.06 ksi	6.0 ksi	.34	Al. 6061, 450°F

Note: 1. For the steel components, stress checks were performed at the enveloping temperatures listed. For the aluminum transition rails, stress checks were performed at temperatures corresponding to the maximum stress point. Temperatures listed are for the maximum stress points of the most highly loaded rail (the large R90 transition rail at the "bottom" of the basket).

- Any impact on temperatures due to the potential changes in the thermal parameters between the TN-24 temperature regime and the NUHOMS®-32PT DSC temperature regime is fully bounded by the conservatism demonstrated in the results of the benchmark analysis.

Therefore, a thermal model that has been properly constructed and predicts conservatively high temperatures in comparison with the TN-24 test data can be fully expected to yield accurate results at the higher temperature levels similar of the NUHOMS®-32PT DSC design.

Several thermal design criteria are established for the thermal analysis of the 32PT DSC basket as discussed below.

- Maximum temperatures of the confinement structural components must not adversely affect the confinement function,
- Maximum fuel cladding temperature limit of 400 °C (752 °F) is applicable to normal conditions of storage and all short term operations including vacuum drying and helium backfilling of the 32PT DSC per Interim Staff Guidance (ISG) No. 11, Revision 3 [4.24]. In addition, ISG-11 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65 °C (117 °F) during drying and backfilling operations,
- Maximum fuel cladding temperature limit of 570 °C (1058 °F) is applicable to accidents or off-normal thermal transients [4.24],
- The maximum DSC cavity internal pressures during normal, off-normal and accident conditions must be below the design pressures of 15 psig, 20 psig and 125 psig (for low burnup fuel – up to 45 GWd/MTU, 105 psig), respectively, and
- Figure M.4-1, Figure M.4-2, Figure M.4-3, *and Figure M.4-3a* show the heat load zoning configurations used in the NUHOMS®-32PT DSC design. The maximum total heat load per DSC is 24 kW or 22.4 kW depending on the specific heat load zoning configuration.

The analyses consider the effect of the decay heat flux varying axially along a fuel assembly. The axial heat flux profile for a PWR fuel assembly is shown in Figure M.4-4 and is based on [4.3].

A description of the detailed analyses performed for normal storage and transfer conditions is provided in Section M.4.4, off-normal conditions in Section M.4.5, accident conditions in Section M.4.6, and loading/unloading conditions in Section M.4.7. The thermal evaluation concludes that with a design basis heat load of up to 24 kW per DSC, all design criteria are satisfied.

The effective thermal conductivity of the fuel assemblies used in the 32PT DSC thermal analysis is based on the conservative assumption of radiation and conduction heat transfer only where any convection heat transfer is neglected. In addition, the fuel assembly with the lowest effective thermal conductivity at the maximum heat load, WE 14x14, is selected as the basis for the thermal analysis. Section M.4.8 presents the calculations that determined WE 14x14 to be the fuel assembly with the lowest effective thermal conductivity in a helium or vacuum environment.

The thermal analysis model conservatively neglects convection heat transfer in the basket regions.

The basket design used for the NUHOMS®-32PT DSC is similar to the basket design of the TN-32 cask [Docket 72-1021]. Both designs use a tube and support rail type of basket, which is significantly different than the spacer disc and guide sleeve type of basket design used for the NUHOMS®-24P DSCs.

The storage modules HSM (models 80, 102, 152, and 202) and HSM-HS can be used to store 32PT DSC. The thermal performances of 32PT DSC in HSM models 80 and 102 are evaluated in this appendix. The thermal performances of 32PT DSC in HSM models 152 and 202 are evaluated in Appendix R, Section R.4.4 and Appendix V, Section V.4.4, respectively. The design and structures of air channels and heat transfer features of HSM-HS are identical to the HSM model 202 described in Appendix V, Section V.1.5. Therefore, the thermal evaluation of 32PT DSC in HSM 202 described in Appendix V is applicable to HSM-HS and no further thermal evaluation is required.

In addition to the design basis thermal evaluation presented in Section M.4.4 through Section M.4.7, Section M.4.12 presents additional thermal evaluations:

1) *Thermal Evaluation for Damaged Fuel Assemblies*
2) *Thermal Evaluation for Failed Fuel Assemblies*
3) *Thermal Evaluation for Heat Load Zoning Configuration (HLZC) # 4*
4) *Thermal Evaluation for 32 Single-Poison-Plate Configuration.*

M.4.2 Summary of Thermal Properties of Materials

The analyses use interpolated values where appropriate for intermediate temperatures. The interpolation assumes a linear relationship between the reported values. The use of linear interpolation between temperature values in the tables for determining intermediate value of property is justified by the near-linear behavior as a function of temperature for the range of interest.

The emissivity of stainless steel is 0.587 [4.7]. For additional conservatism an emissivity of 0.46 for stainless steel is used for the basket steel plates in the analysis. *For the base configuration of the basket assembly, the* emissivity values assumed in the analysis for the aluminum sheets (Type 1100) and aluminum based neutron poison plates is 0.85 which is achieved by either anodizing or other processes. The emissivity of the oxidized Zircaloy surface is 0.8 [4.14]. Emissivity for the aluminum rail material (Type 6061) is not required for the analysis because radiation between the rails and the DSC shell is conservatively neglected in the analysis. For the two alternate basket configuration shown in Figure M.4-22, a minimum emissivity value of 0.8 is assumed for the *aluminum sheets and* neutron poison plates.

The tables below provide the thermal properties of materials used in the analysis of the NUHOMS®-32PT DSC.

1. PWR Fuel with Helium Backfill

The effective thermal conductivity is the lowest calculated value for the various PWR fuel assembly types that may be stored in this DSC and corresponds to the WE 14x14 PWR assembly. Section M.4.8 presents the calculations that determined WE 14x14 to be the fuel assembly with the lowest effective thermal conductivity in both helium and vacuum environment.

Temperature (°F)	K (Btu/min-in-°F)	ρ (lb_m/in^3)	T (°F)	C_p (Btu/lb_m-°F)
Fuel in Helium, Transverse [Section M.4.8]				
138	2.894E-04	0.1166	80	0.0592
233	3.317E-04		260	0.0654
328	3.968E-04		692	0.0725
423	4.744E-04		1502	0.0778
519	5.668E-04			
616	6.715E-04			
714	7.879E-04			
812	9.208E-04			

Temperature (°F)	K (Btu/min-in-°F)	ρ (lb_m/in^3)	T (°F)	C_p (Btu/lb_m-°F)
Fuel in Helium, Axial [Section M.4.8]				
200	7.949E-04	0.1166	See values above	
300	8.387E-04			
400	8.824E-04			
500	9.189E-04			
600	9.554E-04			
800	1.036E-03			

Each aluminum rail may be fabricated as a single piece or in separate segments (3 maximum). Rails consisting of three segments are assumed for thermal analyses. An axial gap of 0.125 in. is considered between any two pieces of aluminum rail. The elements representing the XM-19 grid structure include an adjustment to the conductivity to account for gaps between the basket components.

The 3-D model is extended to approximately half the length of the DSC cavity, or 83.8 in. to model the bottom half of the canister. A symmetry boundary is applied on the axial top of the model. The heat generations are applied over the active fuel, starting from 8.625 in. from the bottom of the fuel regions and extending all the way to the top of the model. The placement of the active fuel and the model size results in slight overprediction of temperatures since the symmetry boundary at (167/2-8.625)=74.875 in. from the beginning of active fuel is located well beyond half of the active fuel, or 141.8/2=70.9 in., where peak temperatures would be expected. Active length of B&W 15x15 assembly is shorter than that of WE 14x14 assembly. Therefore using the active length of B&W assembly results in higher heat generating rates in the model, which is conservative. In addition, active fuel of WE 14x14 assembly begins about 4" from the bottom of fuel assembly. Shifting the active fuel length to 8.625" from the bottom, shifts the active fuel length away from regions of dropping DSC shell temperature in the model. This is conservative, since being away from the regions of dropping DSC shell temperatures causes higher fuel cladding temperature. Longer DSC cavity configurations (32PT-L100 and 32PT-L125) provide larger radial surface areas for heat dissipation and are therefore bounded by the shorter cavity (32PT-S100 and 32PT-S125) DSC configurations.

A basket model with alternate poison plate configuration has been developed with the following changes.

1. The emissivity of the anodized *aluminum or* poison plate has a minimum value of 0.8, which will cause a reduction in the transverse fuel effective thermal conductivities.
2. The R90 aluminum transition rails are split into three sections in the radial direction.
3. The intermittent weld with 8 in. sections of weld out of every 10 inches of basket length is explicitly modeled. Figure M.4-23 shows the gap in the rail and the intermittent weld as modeled.
4. All the poison plate/aluminum plates "chevrons" have been rotated so that they have an "L" or reversed "L" orientation. There are two types of poison plate configurations developed. One is a 16 poison plate and the second is a 24 poison plate configuration as shown in Figure M.4-22. The second alternate configuration consists of either 24 poison plates paired with aluminum chevrons or 24 single poison plates.

The analysis performed for the basket with alternate poison plate configuration concludes that the resultant maximum component temperatures and DSC internal pressures are all bounded by the basket with original poison plate configuration.

M.4.6.5 Maximum Internal Pressures

The maximum accident pressure condition for the DSC occurs during the transfer accident case with the loss of the sun shield and liquid neutron shielding in the TC under extreme ambient temperature conditions of 117 °F and maximum insolation. For this transfer accident condition, the average helium temperature is 703 °F (1,163 °R). *However, 705 °F (1,165 °R) is conservatively used.* In accordance with NUREG 1536, 100% of the fuel pins are assumed to rupture during this event. During the blocked vent case, the average gas temperature is 623 °F. However, since no DSC drop events can occur in conjunction with a blocked vent event, the maximum fraction of fuel pins that can be ruptured is limited.

A summary of the maximum accident operating pressures for the various 32PT DSC configurations are presented in Table M.4-15.

M.4.6.6 Evaluation of Cask Performance During Accident Conditions

The temperatures in the NUHOMS® HSM and TC are bounded by the existing analyses because of the same heat load for the NUHOMS®-24P DSC design. The NUHOMS®-32PT DSC shell and basket are evaluated for calculated pressures and temperatures in Section M.3.

The maximum fuel cladding temperature of 863 °F is below the short-term limit of 1058 °F (570 °C). The accident pressure in the NUHOMS®-32PT DSC of 107.3 psig remains below the accident design pressure of 125 psig. It is concluded that the NUHOMS®-32PT *System* maintains confinement during the postulated accident condition.

M.4.7 Thermal Evaluation for Loading/Unloading Conditions

All fuel transfer operations occur when the NUHOMS®-32PT DSC and TC are in the spent fuel pool. The fuel is always submerged in free-flowing pool water permitting heat dissipation. After fuel loading is complete, the cask and DSC are removed from the pool and the DSC is drained, dried, backfilled with helium and sealed.

M.4.7.1 Maximum Fuel Cladding Temperatures During Vacuum Drying

The loading condition evaluated for the NUHOMS®-32PT DSC is the heatup of the DSC before its cavity is backfilled with helium. This typically occurs during the performance of the vacuum drying operation of the DSC cavity with the Cask in the vertical configuration inside the fuel handling building, and the annulus between the Cask and the DSC full of water.

The vacuum drying of the DSC generally does not reduce the pressure sufficiently to reduce the thermal conductivity of the water vapor and helium in the DSC cavity [4.22] and [4.32]. Radiation in the gaps within the basket and rail components is conservatively neglected. Analyses are performed to determine the steady state temperatures during the vacuum drying condition.

Thermal analysis is performed using the three-dimensional model developed in Section M.4.4.1, decay heat loads for heat load zoning configuration 1, 2, 3, *and 4,* and a maximum DSC temperature of 215 °F. The initial temperature of the DSC, basket and fuel is assumed to be 215 °F, based on the boiling temperature of the fill water. Due to the use of helium for blowdown, the results of this analysis are bounded by the off-normal thermal analysis for transfer presented in Section M.4.5.2. Table M.4-17 and Table M.4-18 provide the maximum calculated temperatures for the fuel cladding and basket components, respectively, for all *four* configurations. The maximum cladding temperature reached during vacuum drying is less than 715 °F, which is below the limit of 752 °F [4.24].

M.4.7.2 Evaluation of Thermal Cycling of Fuel Cladding During Vacuum Drying, Helium Backfilling and Transfer Operations

ISG-11 also states that thermal cycling is to be minimized and imposes a limit of 65 °C (118 °F) on thermal cycling (reduction in fuel clad temperature from previous peak temperature). The basis for the limit is that as the cladding temperature is reduced more than 65 °C the concentration of hydrogen available for hydride reorientation becomes significant.

The thermal analysis of the 32PT DSC during blowdown operation assumes helium is used to drain the water from 32PT DSC cavity and subsequent vacuum drying occurs with a helium environment. This configuration eliminates a fuel cladding temperature drop that would take place during helium backfilling of the 32PT DSC subsequent to vacuum drying in a nitrogen environment and it eliminates the need for a time limit on the vacuum drying operation, since the thermal conductivity of helium does not change with pressure during vacuum drying operations. As shown in Table M.4-17 the maximum fuel cladding temperature limit of $T_{ISG\ limit}$ = 400 °C (752 °F) in ISG-11 [4.24] is satisfied for the 32PT DSC.

M.4.12 Additional Thermal Evaluations

The following sections provide additional thermal evaluations:

M.4.12.1 Thermal Evaluations for Damaged Fuel Assemblies

The NUHOMS®-32PT DSC can accommodate a combination of intact fuel assemblies along with damaged fuel assemblies. It can be loaded with up to 28 damaged fuel assemblies as noted in Section M.2.1. This section evaluates the thermal performance of the NUHOMS® -32PT DSC loaded with intact and damaged fuel assemblies during storage and transfer conditions.

M.4.12.1.1 Thermal Evaluation for Normal and Off-Normal Conditions

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M.4.12.1.2 Thermal Evaluation for Accident Conditions

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All changes on this page are Amd 15.

Proprietary Information on Pages M.4-51b through M.4-51c
Withheld Pursuant to 10 CFR 2.390.

[

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Table M.4.12.1-1
Maximum Component Temperatures for 32PT DSC with up to 28 Damaged FAs under the Bounding Accident Condition

Transfer Condition	[			]
DSC Type	*32PT*			
Heat Load, kW	*24 (HLZC # 1)*			
Fuel Assembly Type	***32 Intact***	[ ]	[ ]	[ ]
Component	*Maximum Temperature (°F)*			
Intact Fuel Assemblies	*863 (Table M-4-13)*	[		
Fuel Rubble	--			
Fuel Compartment	*852 (Table M.4-14)*			
Neutron Absorber	*852 (Table M.4-14)*			
R45/R90 Rails	*631 (Table M.4-14)*			]

[

] *This is lower than the average helium temperature of 705 °F (1165 °R) (See Section M.4.6.5) used to determine the maximum internal pressure for the bounding hypothetical accident condition with intact FAs.* [

] *This maximum internal pressure is well below the allowable limit of 125 psig as discussed in Section M.4.6.6 for maximum average burnups up to 55 GWd/MTU noted in Table M.4-15.*

Therefore, the maximum fuel cladding temperature and maximum internal pressure remain below the limits and meet thermal design criteria specified in Section M.4.1 in the worst accident scenario for the storage of up to 28 damaged fuel assemblies in the 32PT DSC.

It is concluded that the NUHOMS®-32PT DSC for the storage of up to 28 damaged FAs maintains confinement during the postulated accident condition.

M.4.12.2 Thermal Evaluations for Failed Fuel Assemblies

The NUHOMS®-32PT DSC can accommodate up to 8 failed fuel cans (FFCs) in the corner locations of heat load zoning configuration (HLZC) #2 as shown in Figure M.4-2. Failed FAs are encapsulated in individual FFCs that are designed to fit into the existing 32PT basket fuel compartments at the corner locations. For this configuration, the maximum heat load per FFC is limited to 0.8 kW.

This section evaluates the thermal performance of the NUHOMS® -32PT DSC loaded with up to 8 FFCs during storage and transfer conditions.

M.4.12.2.1 Thermal Evaluation for Normal and Off-Normal Conditions

[

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Boundary conditions for the limiting case (normal transfer, 100 °F ambient with insolation) with HLZC #2 are taken from Section M.4.4.1.8. The DSC shell temperature profile based on the total DSC heat load of 24 kW from Section M.4.4.1.8 is conservatively applied for the limiting case with failed FAs.

Table M.4.12.2-1 lists the maximum temperatures of fuel cladding for intact FAs and the 32PT DSC components for the limiting case (normal transfer, 100 °F ambient with insolation) with 8 FFCs in HLZC#2 compared with the design basis values (normal transfer, 100 °F ambient with insolation) with all intact FAs in HLZC # 2 reported in Section M.4.4.2.

Table M.4.12.2-1
Maximum Temperature of Fuel Cladding and 32PT DSC Components without and with Failed FAs in FFCs for Limiting Case HLZC # 2

	32 intact FAs *(Tables M.4-2, M.4-4)*	***24 intact FAs and 8 FFCs***	***Maximum ΔT***
Intact FA, °F	*705*	[	]
Grid, °F	*691*		
Rail, °F	*473*		
Poison, °F	*691*		
Shell, °F	*445*		
FFC wall, °F	*n/a*	[	]

[

]

For the limiting case (normal transfer, 100 °F ambient with insolation) with 8 FFCs in HLZC#2, the average temperature of helium in the DSC cavity is [ ]. *This temperature is below the design basis value of 550 °F used in the maximum normal operating pressure evaluation from Section M.4.4.4.6 and maximum off-normal operating pressure evaluation from Section M.4.5.4.*

Therefore, the design basis internal pressures in the 32PT DSC cavity for normal and off-normal operating conditions reported in Table M.4-7 and Table M.4-12 are bounding for the 32PT DSC with any combination of intact FAs and up to 8 FFCs in HLZC#2.

It is concluded that the NUHOMS®-32PT DSC for storage of up to 8 FFCs with HLZC # 2 meet all applicable normal and off-normal thermal requirements.

M.4.12.2.2 Thermal Evaluation for Accident Conditions

For accident storage and transfer conditions, the accident transfer condition with loss of sunshade and neutron shield with 24 kW heat load per DSC is the limiting case as discussed in Section M.4.6. The same boundary conditions for accident transfer condition from Section M.4.6 are applied in the 32PT DSC full-length model.

[ ]

Table M.4.12.2-2 lists the maximum temperatures of fuel cladding for intact FAs and 32PT DSC components for the limiting accident conditions with 8 FFCs in HLZC #2 compared with the bounding design basis values with all intact FAs from Section M.4.6.4.

Table M.4.12.2-2
Maximum Temperature of Fuel Cladding and 32PT DSC Components without and with Fuel Rubbles in FFCs for Accident Condition

	32 intact FAs *(Tables M.4-13, M.4-14)*	***24 intact FAs and 8 FA rubbles***
Intact FA, °F	*863*	[ ]
FA rubble, °F	*n/a*	[ ]
Grid, °F	*852*	[ ]
Rail, °F	*631*	[ ]
Poison, °F	*852*	[ ]
Shell, °F	*600*	[ ]
FFC wall, °F	*n/a*	[ ]

[ ]

For the limiting accident condition with 8 FFCs in HLZC#2, the average temperature of helium in the DSC cavity of [ ] is below the design basis value of 705 °F (see Section M.4.6.5). Therefore, the design basis values of the internal pressures in the 32PT DSC cavity for accident conditions as reported in Table M.4-15 are bounding for the 32PT DSC with HLZC #2 with 8 FFCs.

It is concluded that the NUHOMS®-32PT DSC for the storage of up to 8 failed FAs in FFCs of HLZC # 2 maintains confinement during the postulated accident condition.

M.4.12.3 Thermal Evaluations for Heat Load Zoning Configuration (HLZC) # 4

This section evaluates the thermal performance of the NUHOMS® -32PT DSC loaded in HLZC # 4 during storage and transfer conditions. The NUHOMS®-32PT DSC can be loaded with the maximum heat load per FA up to 2.2 kW in HLZC #4 as shown in Figure M.4-3a. For this configuration, the maximum heat load per DSC is limited to 24 kW.

The maximum heat load of 24 kW for HLZC #4 is equal to the maximum allowable heat load specified for HLZCs #1 and #2 as shown in Figures M.4-1 and M.4-2. Since no other changes are considered to the 32PT DSC except for the HLZC, the thermal evaluation of the 32PT DSC with HLZC #4 is based on a sensitivity analysis. A review of the maximum fuel cladding and basket component temperatures listed in Tables M.4-2 through M.4-5 and Tables M.4-8 through M.4-11 indicates that, among all the normal and off-normal conditions of storage and transfer operations, the normal transfer operation with HLZC #1 (DSC horizontal in cask, 100 °F ambient) represents the bounding case. Therefore, a sensitivity analysis based on the normal transfer operation (DSC horizontal in cask, 100 °F ambient) is selected to evaluate the thermal performance of the 32PT DSC with HLZC #4.

All thermal properties for the materials used in this sensitivity analysis are the same as those described in Section M.4.2.

Boundary conditions for the limiting case (DSC horizontal in cask, 100 °F ambient) with HLZC #4 are taken from Section M.4.4.1.8.

A review of the HLZC #4 in Figure M.4-3a indicates that if each fuel assembly is loaded at the maximum allowed heat load, the total heat load of the DSC would be 30.8 kW and will exceed the maximum allowable heat load of 24 kW for the 32PT DSC. However, to bound the various options that could be used to adjust the maximum heat load to 24 kW in HLZC # 4, the maximum heat load per fuel assembly listed in Figure M.4-3a is used in this evaluation. This is conservative since the total heat load of the DSC considered is 30.8 kW compared to the maximum allowable heat load of 24 kW.

Heat generation rates are calculated using the same methodology as described in Section M.4.4.1.4. Peaking factors and the minimum active fuel length are also the same as those described in Section M.4.4.1.4 with a correction factor of 1.013575 applied in the full-length model.

Table M.4.12.3-1 presents the maximum temperatures of fuel cladding and DSC components for the 32PT DSC with HLZC #4 from the sensitivity analysis compared to the design basis values for the normal transfer operation (DSC horizontal in cask, 100 °F ambient) with HLZC #1 as discussed in Section M.4.4.2 and listed in Tables M.4-2 and M.4-3.

As seen from Table M.4.12.3-1, the maximum fuel cladding temperature for the 32PT DSC with HLZC #4 is 705 °F with significant margin to the temperature limit of 752 °F. Further, the maximum fuel cladding temperature is also 15 °F lower than that determined previously for the design basis evaluation with HLZC #1. Therefore, the maximum fuel cladding temperatures of the 32PT DSC with HLZC #1 listed in Table M.4-2 remain bounding for the 32PT DSC with HLZC #4 for normal conditions.

Table M.4.12.3-1
Maximum Component Temperatures for 32PT DSC for Normal Transfer, 100°F Ambient

HLZC	$T_{fuel,max}$ (°F)	$T_{grid,max}$ (°F)	$T_{rail,max}$ (°F)	$T_{Al,max}$ (1) (°F)
HLZC #1, Design Basis (See Tables M.4-2 and M.4-3)	720	705	471	705
HLZC #4	705	694	491	694
ΔT, °F ($T_{HLZC\ \#4} - T_{Design\ Basis}$)	-15	-11	20	-11

Similarly, for the basket components, the maximum temperatures determined for the 32PT DSC with HLZC #4 remain bounded by the maximum temperatures determined for HLZC #1 except for the rail wherein a temperature increase of 20 °F is observed. However, there is no impact of this increase on the performance of this component as discussed in Section M.3.6.1.3.2.

Table M.4.12.3-1 shows that normal and off-normal transfer operations of the 32PT DSC with HLZC #4 are bounded by design basis load cases for normal and off-normal conditions. Similarly for accident conditions, the maximum temperatures of fuel cladding and DSC components listed in Table M.4-13 and M.4-14 for accident conditions remain bounding for the 32PT DSC with HLZC #4.

The average helium temperature in DSC cavity for the 32PT DSC with HLZC # 4 is 559 °F (1019 °R). This is a 9 °F increase compared to the design basis average helium temperature of 550°F (1010 °R) as noted in Section M.4.4.4.6. This small change has a negligible effect for the design basis maximum internal pressures listed in Table M.4-7, Table M.4-12 and Table M.4-15 for normal, off-normal and accident conditions.

Thermal evaluation of the 32PT DSC with damaged fuel assemblies in Appendix M.4.12.1 shows that up to 28 damaged fuel assemblies can be stored in the 32PT DSC with HLZC #1 through HLZC # 3.

Based on the discussion presented in Appendix M.4.12.1.1, storage of damaged FAs within the 32PT DSC does not impact either the normal or the off-normal conditions. Similarly, there is no impact of storing damaged FAs per HLZC # 4 for normal/ off-normal conditions.

However as discussed in Appendix M.4.12.1.2, there is a potential for damaged FAs to further degrade during accident conditions. While storing damaged FAs, HZLC # 4 differs from HLZC #1 evaluated in Appendix M.4.12.1 as follows:

1. *HLZC # 4 only allows up to 20 damaged FAs compared to 28 for HLZC # 1 through 3. A review of the results presented in Table M.4.12.1-1 shows that the number of damaged FAs has a significant impact on the maximum fuel cladding temperature of the intact FAs. Since the number of damaged FAs is significantly lower for HLZC # 4, this change is conservative.*

2. *Within HLZC # 4, the maximum decay heat load per FA in Zone 1 is 0.4 kW compared to 0.63 kW per FA in HLZC # 1 analyzed in Appendix M.4.12.1.*

   A reduction in the maximum decay heat load per FA in Zone 1 is conservative and will help further reduce the maximum fuel cladding temperature during accident conditions.

3. *Within HLZC # 4, the maximum decay heat load per FA in locations identified as Zone 3 and 4 is higher compared to that of HLZC # 1.*

   For HLZC # 4, the maximum decay heat load per FA is 2.2 kW in Zone 3 and 1.7 kW in Zone 4 whereas the maximum decay heat load per FA in Zone 2 for HLZC # 1 is 0.87 kW. However, as seen from Table M.4.12.3-1, the maximum fuel cladding temperature for HLZC # 1 is higher. This is due to the reduction in the decay heat load per FA of the inner compartments while increasing the decay heat load per FA of the outer compartments. Since the outer compartments with higher decay heat load per FA in HLZC # 4 are limited to storage of intact fuel assemblies, a similar behavior will be observed.

Therefore, the thermal evaluation of the 32PT DSC with damaged fuel assemblies in Appendix M.4.12.1 remains bounding for HLZC # 4.

Based on the above discussion, no further evaluations are required for the 32PT DSC with HLZC #4 and all design criteria described are satisfied.

M.4.12.4 Thermal Evaluation for 32 Single-Poison- Plate Configuration

As discussed in Section M.4.4.1.1, the thermal evaluation of the 32PT DSC for the second alternate poison plate configuration consisting of either 24 poison plates paired with aluminum chevrons or 24 single poison plates is bounded by the thermal evaluation of the basket with the original poison plate configuration in Section M.4.4 through Section M.4.7. This section evaluates the thermal performance of the third alternate poison plate configuration consisting of 32 single-poison-plates.

All thermal properties used for the evaluation are the same as those listed in Section M.4.2. The material properties for the 32 single poison plates are the same as those specified in Section M.4.3.

Similar to the previously evaluated 24 single-poison-plate configuration shown in Figure M.4-22, the design change for the 32 single-poison-plate configuration is limited to the replacement of eight aluminum chevrons with eight single poison plates. Since no other changes are considered to the 32PT DSC, the thermal evaluation of the 32PT DSC with this design change is based on a sensitivity study. A review of HLZC #1 through HLZC #4 shown in Figure M.4-1 through Figure M.4-3a indicates that HLZC #3 is the bounding HLZC since it has the highest heat load per fuel assembly stored in the inner compartments among all HLZCs.

A review of the maximum fuel cladding temperatures for HLZC # 3 listed in Tables M.4-1, M.4-2, M.4-8 and M.4-13, the normal transfer operation (DSC horizontal transfer in cask, 100 °F, HLZC # 3) listed in Table M.4-2 represents the limiting case due to its smallest margin to the fuel cladding temperature limit among all storage and transfer conditions. Therefore, a sensitivity analysis based on the limiting case is selected to re-evaluate the thermal performance of the 32PT DSC with 32 single-poison-plate configuration.

The 32PT DSC thermal model described in Section M.4.4.1.1 is modified by

- *selecting the elements representing the single aluminum plates or the poison plate paired with aluminum chevrons and the contact gaps between them , and*
- *modifying the material property to that of poison plates listed in Section M.4.3.*

The boundary condition and heat generation for the limiting case (DSC horizontal transfer in cask, 100 °F, HLZC # 3) described in Section M.4.4.1 are used in the sensitivity evaluation.

Table M.4.12.4-1 compares the maximum temperatures of fuel cladding and DSC components for the 32PT DSC with 32 single-poison-plate configuration from the sensitivity evaluation to the design basis values listed in Table M.4-2 and Table M.4-5 for the limiting case (DSC horizontal in cask, 100 °F, HLZC # 3).

Table M.4.12.4-1
Maximum Component Temperatures for 32PT DSC for Normal Transfer, 100°F Ambient, HLZC # 3

	$T_{fuel,max}$ (°F)	$T_{grid,max}$ (°F)	$T_{rail,max}$ (°F)	$T_{Al,max}$ (°F)
Design Basis (See Tables M.4-2 and M.4-5)	*720*	*704*	*456*	*704*
32 Single-Poison-Plate Configuration	*723*	*703*	*454*	*703*
ΔT ($T_{32SinglePoison} - T_{Design\ Basis}$)	*3*	*-1*	*-2*	*-1*

As seen from Table M.4.12.4-1, the maximum fuel cladding temperature for the 32PT DSC with 32 single-poison-plate configuration is 723 °F with a 3 °F increase compared to the design basis temperatures. However, the maximum fuel cladding temperature remains below the temperature limit of 752 °F. Since this evaluation is based on the bounding condition, the fuel cladding temperature will remain below 723 °F for all normal storage and short term transfer operations. For the various basket components, the maximum temperatures for the 32PT DSC with 32 single-poison-plate configuration remain bounded by the design basis values.

For the off-normal storage and accident conditions, a review of the maximum fuel cladding temperatures listed in Table M.4-8 and Table M.4-13 shows that the accident transfer condition listed in Table M.4-13 has the lowest margin of 195 °F to the fuel cladding temperature limit of 1058 °F. Due to this large margin of 195 °F, a temperature increase of 3°F in a 32PT DSC with 32 single-poison-plate configuration has a negligible effect on the fuel cladding integrity.

Therefore, the maximum temperatures of fuel cladding for the 32PT DSC under all normal, off-normal, and accident conditions remain below the limits defined by ISG-11 [4.24].

The average helium temperature in DSC cavity for the 32PT DSC with 32 single-poison-plate configuration is 533 °F. This is lower than the design basis average helium temperature of 550 °F shown in Section M.4.4.4.6. Therefore, there is no impact on the internal pressure evaluation.

Based on the above discussion, no further evaluations are required for the 32PT DSC with 32 single-poison-plate configuration and all design criteria described in Section M.4.1 are satisfied.

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-2.

Figure M.4-1

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-3.

Figure M.4-2

All changes on this page are Amd 15.

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-4.

Figure M.4-3

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-4a.

Figure M.4-3a

M.5 Shielding Evaluation

The radiation shielding evaluation for the Standardized NUHOMS® System (during loading, transfer and storage) for the 24P and 52B canisters is discussed in Sections 3.3.5, 7.0 and 8.0. The following radiation shielding evaluation specifically addresses the dose rates due to design-basis PWR fuel and control components (CCs) loaded in a NUHOMS®-32PT DSC. The shielding analysis is carried out for the four DSC configurations of the NUHOMS®-32PT System described in Section M.2.1. The basket layout for these configurations is identical except for the length of the DSC components. For shielding purposes, the only difference between the 32PT-S100/32PT-L100 and 32PT-S125/32PT-L125 versions is the thickness of the shield plug designs. The 32PT-S100/32PT-L100 versions have somewhat thinner shield plugs than the 32PT-S125/32PT-L125 versions. Each of the configurations is designed to store up to 32 intact standard PWR fuel assemblies. The 32PT-L100 and 32PT-L125 are also designed to store up to 32 intact standard PWR fuel assemblies with or without *CCs*. Therefore, for shielding purposes, AMD 15
the two long-cavity versions bound the short-cavity versions because of the additional gamma source due to the CCs, such as burnable poison rod assemblies (BPRAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), thimble plug assemblies (TPAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), neutron source assemblies (NSAs), and neutron sources. Therefore, the shielding evaluation presented herein is performed only for the 32PT-L100 and 32PT-L125 with fuel plus *CCs*. To ensure that this evaluation is conservative, the fuel source terms are not adjusted to AMD 15
account for the additional decay required to accommodate the CCs.

The NUHOMS® 32PT DSC is stored in the standardized NUHOMS® horizontal storage module (HSM). In addition, the NUHOMS® 32PT DSC can also be stored within an upgraded HSM model, designated as HSM-HS as described in Appendix U of the UFSAR.

The NUHOMS® 32PT DSC is transferred in the OS197 or OS197H transfer cask (TC). The NUHOMS® 32PT DSC is also qualified for transfer in the OS197L TC, within certain limitations, as described in Appendix W. Finally, the NUHOMS® 32PT DSC is also transferred in a modified version of the OS200 TC as described in Appendix U of the UFSAR. The OS200 TC is fitted with an aluminum sleeve to accommodate the smaller diameter 32PT DSC.

The design-basis PWR fuel source terms are derived from the bounding fuel, B&W 15x15 Mark B assembly design as described in Section M.5.2. 72.48

The NUHOMS®-32PT DSCs can store intact (including reconstituted) PWR fuel assemblies and *CCs* with the characteristics described in Table M.2-1. The NUHOMS®-32PT DSC may store PWR fuel assemblies arranged in any of *four* alternate heat zoning configurations with a AMD 15
maximum decay heat of *2.2* kW per assembly and a maximum heat load of 24 kW per canister. The heat load configurations are shown in Figure M.2-1 *through Figure M.2-3a*. Note that while the B&W, CE, and Westinghouse fuel designs are specifically listed, storing reload fuel designed by other manufacturers is also allowed provided an analysis is performed to demonstrate that the limiting features listed in Table M.2-1 bound the specific manufacturers replacement fuel. The limiting features are burnup, initial enrichment, cooling time, fissile material type, number of fuel rods, number of guide tube/instrument tube holes, cobalt impurities in the hardware and initial heavy metal.

The design-basis fuel, 0.475 MTU of heavy metal weight, source terms for this evaluation are defined as the source terms from fuel with the burnup/initial enrichment/cooling time combination given in *Technical Specifications Tables 1-3a through 1-3p* (with or without CCs) and located in the basket as shown in Figure M.2-1 *through Figure M.2-3a*, that gives the maximum dose rate on the surface of the HSM and/or OS197/OS197H TC. This approach is consistent with the method used to generate the fuel qualification tables for the Standardized NUHOMS®-24P and -52B canister designs as described in Section 7.2.3.

The fuel qualification tables (FQTs) are provided in Tables 1-3a through 1-3p of the Technical Specifications. FQTs are developed for heavy metal loadings of 0.475 MTU and 0.380 MTU with the ORIGEN-ARP module of SCALE6.0 [5.14]. Because the FQTs are developed based upon fuel assembly decay heat limits, the cooling times are shorter for lower heavy metal loadings. Section 7.2.3.2 of Chapter 7 provides the methods for determining minimum required cooling times using fitting equations or linear interpolation for a given MTU between *0.380* MTU and 0.475 MTU.

In the original analysis, Heat Load Zoning Configuration 2 (HLZC#2, Figure M.2-2) was the configuration that produced the highest dose rates on the surface of the HSM and TCs. Source terms consistent with HLZC#2 were used in conjunction with the DORT 2-D discrete ordinance software package to compute dose rates for both the HSM and TC. In order to model Heat Load Zoning

Configuration 2, all sixteen assemblies in the outer ring of the DSC are modeled with source terms consistent with 1.2 kW. Therefore, the source terms result in fairly conservative dose rates because the shielding analysis is based on a 28.8 kW heat load compared to the 24 kW heat load limit.

The design basis dose rate calculations for the HSM and TC are performed using the 2-D DORT discrete ordinates program and design basis source terms computed by the SAS2H/ORIGEN-S module of SCALE4.4 [5.1]. However, because the original SCALE4.4/SAS2H FQTs have been replaced by FQTs generated with SCALE6.0/ORIGEN-ARP over a larger range of burnups, enrichments, and heavy metal loadings, the original DORT analysis is no longer bounding. Also, the 32PT DSC HLZC#4, which is bounding for dose rate analysis, was not available when the original DORT analysis was performed. To minimize rework, the DORT analysis is maintained as the analysis of record; however, scaling factors are applied to all DORT analysis results to generate dose rates that bound HLZC#4 and the FQTs (Technical Specifications Tables 1-3a through 1-3p). This updated shielding analysis that determines the effect of uranium loading (380 kgU per assembly) on the dose rates is summarized in Section M.5.4.16.

The bounding burnup, minimum initial enrichment and cooling time combinations used in *the DORT* analysis are as follows:

- 30 GWd/MTU, 2.5 wt. % U-235, 8-year cooled – Inner sixteen assemblies in the HSM models,
- 41 GWd/MTU, 3.1 wt. % U-235, 5-year cooled – Outer sixteen assemblies in the HSM and TC models, and
- 45 GWd/MTU, 3.3 wt. % U-235, 23-year cooled - Inner sixteen assemblies in the TC models.

Source terms developed with SCALE6.0/ORIGEN-ARP consistent with the FQTs (Technical Specifications Tables 1-3a through 1-3p) are defined in Section M.5.2.7.

The design-basis source terms for the authorized CCs are taken from Appendix J. The design-basis source terms are based on three bounding CC designs: (1) B&W 15x15 Burnable Absorber Assemblies with up to 2 cycles burnup and 5-year cooled, (2) WE 17x17 Pyrex Burnable Absorber, 2-24 Rodlets with up to 2 cycles burnup and 10-year cooled, and (3) WE 17x17 WABA Burnable Absorber, 3-24 Rodlets with up to 2 cycles burnup and 10-year cooled. The properties used in Appendix J to calculate the design-basis source terms are reproduced in Table M.5-2.

The design basis CC source term that envelops all CCs allowed in the 32PT DSCs is taken from Appendix J for BPRAs with burnups up to 36 GWd/MTU. While Appendix J was developed to specifically address the additional source from a BPRA, this source term is selected as the bounding source term for all CCs. The TPAs and ORAs do not extend into the active fuel region of a fuel assembly. Therefore, they are limited to the source term equivalent to the top plus plenum region source term of a BPRA. However, *to be* conservative, the full total source term of BPRA is used in the shielding analysis to bound all CCs. The source term energy distribution is shown in M.5-12. Any CC to be stored in a 32PT DSC must be bounded by this source term.

Reconstituted fuel assembly is an intact fuel assembly in which one or more enriched fuel rods have been replaced with either stainless steel rods or zircaloy clad rods with depleted, natural or lower than original enriched uranium dioxide as fuel material referred to as "lower enrichment UO_2 rods" in this section.

The replacement rod is of similar outside dimensions as the original fuel rod, displacing the same amount of water in the fuel matrix. The lower enrichment UO_2 rods are of similar design and behavior as the standard fuel rods aside from the uranium enrichment.

Reconstituted fuel assemblies with up to 56 solid stainless steel rods or unlimited number of lower enrichment UO_2 rods that replace fuel rods are also acceptable for the 32PT DSC payload. The reconstituted rods may be placed at any location in the fuel assembly and the reconstituted assemblies may be placed anywhere in the basket. The cooling time required for reconstituted fuel assembly is increased *per Section M.5.2.5.*

CE 15x15 fuel assemblies with plugging clusters having a nominal mass of 2.3 kg 304L stainless steel (including 0.1 kg Inconel x-750) are evaluated. The material weights in the top, plenum and the incore region (including the weight of the heavy metals) used for the design basis source term calculation bound the CE 15x15 fuel assembly with plugging clusters.

The methodology, assumptions, and criteria used in this evaluation are summarized in the following subsections.

M.5.1 Discussion and Results

The maximum dose rates due to 32 design-basis PWR fuel assemblies with *CCs* in the NUHOMS®-32PT DSC loaded into the Standardized NUHOMS®-HSM are summarized in Table M.5-3 for both the 32PT-S100/32PT-L100 and 32PT-S125/32PT-L125 design configurations. Table M.5-4 provides maximum and surface average dose rates on the HSM loaded with the NUHOMS®-32PT DSC for both the 32PT-S100/32PT-L100 and 32PT-S125/32PT-L125 design configurations. Table M.5-5 provides a summary of the dose rates on and around the TC for canister transfer for 32PT-S100/32PT-L100 and 32PT-S125/32PT-L125 configurations. The dose rates in these tables are for *HLZC#2. These dose rates are scaled as indicated in Section M.5.4.16 to account for HLZC#4 and the FQTs (Technical Specifications Tables 1-3a through 1-3p).*

A basket with two alternate poison plate configurations is also considered. One is a 16 poison plate and the second is a 24 poison plate configuration. Since the total weight of the material in the basket is the same as the original poison plate configuration, the results calculated here are also applicable to baskets with these alternate poison plate configurations.

A discussion of the method used to determine the design-basis fuel and CC source terms is included in Section M.5.2. The model specification and shielding material densities are given in Section M.5.3. The method used to determine the dose rates due to 32 design-basis fuel assemblies with CCs in the NUHOMS®-32PT DSC design configurations is provided in Section M.5.4. Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The shielding evaluation is performed with the DORT [5.2] code with the CASK-81 cross section library [5.3]. Sample input files used for calculating neutron and gamma source terms and dose rates are included in Section M.5.5.1.

The shielding evaluations for storage configurations documented herein are based on the Standardized HSM design and are bounding for the HSM-HS design. This is due to the fact that the HSM-HS module contains at least one additional *foot* of concrete shielding. Therefore, no additional shielding calculations are necessary for the HSM-HS design.

The shielding evaluations for loading and transfer configurations documented herein are based on the OS197 TC and are bounding for the OS200 TC. This is due to the fact that the neutron and gamma shielding material thicknesses are slightly higher for the OS200 TC. Further, the aluminum sleeve employed to accommodate the smaller diameter 32PT DSC within the OS200 TC also provides for slightly enhanced gamma shielding. Therefore, no additional shielding calculations are necessary for the OS200 TC.

The NUHOMS®-32PT DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. [

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M.5.2 Source Specification

Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for *a* heavy metal weight of 0.475 MTU. SAS2H/ORIGEN-S *is* used to develop the design-basis source terms. The thermal and radiological source terms for the CCs are taken from Appendix J.

The B&W 15x15 assembly is the bounding fuel assembly design for shielding purposes because it has the highest initial heavy metal loading as compared to the 14x14, other 15x15, and 17x17 fuel assemblies which are also authorized contents of the NUHOMS®-32PT DSC. In addition, the maximum Co59 content of the hardware regions for each assembly type is less than that of the B&W 15x15 Mark B fuel assembly. The neutron flux during reactor operation is peaked in the active fuel or in-core region of the fuel assembly and drops off rapidly outside the active fuel region. Much of the fuel assembly hardware is outside of the active fuel region of the fuel assembly. To account for this reduction in neutron flux, the fuel assembly is divided into four exposure "regions." The four axial regions used in the source term calculation are: the bottom (nozzle) region, the active fuel region, the (gas) plenum region, and the top (nozzle) region. The B&W 15x15 fuel assembly masses for each irradiation region are listed in Table M.5-6. The light elements that make up the various materials for the various fuel assembly materials are taken from reference [5.4] and are listed in Table M.5-7. The design-basis heavy metal weight is 0.475 MTU. These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. To account for the reduction in neutron flux outside the active fuel regions neutron flux (fluence) correction factors are applied to light element composition for each region. The neutron flux correction factors are given in Table M.5-8.

The fuel qualification tables are generated based on the decay heat limits for the various heat load zoning configurations shown in Figure M.2-1 *through Figure M.2-3a. SCALE6.0/ORIGEN-ARP* is used to calculate the minimum required cooling time as a function of assembly initial enrichment and burnup for each decay heat limit. The total decay heat includes the contribution from the fuel as well as the hardware in the entire assembly. *The FQTs also account for the 8 watts per design basis CC.* Because the decay heat generally increases slightly with decreasing enrichment for a given burnup, it is conservative to assume that the required cooling time for a higher enrichment assembly is the same as that for a lower enrichment assembly with the same burnup. The required cooling time for initial enrichments that fall between any two *data points* are assumed to be that of the lower enrichment case results.

Reconstituted fuel assemblies containing up to 56 stainless steel rods that replace fuel rods are also acceptable for the 32PT DSC payload provided the cooling time requirements of the fuel qualification tables are met. Additional discussion of the methodology used to evaluate reconstituted fuel assemblies is provided in Section M.5.2.5.

The 1-D discrete ordinates code ANISN [5.5] and the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.3] is used to determine these design-basis source terms. Finding the burnup/initial enrichment/cooling time combinations from the fuel qualification tables and decay heat load zoning configurations that produce the maximum dose rate on the HSM roof determine the design-basis source term for the HSM shielding calculations. Similarly the design-basis source terms for the OS197/OS197H TC are determined by finding the maximum surface dose rates on the side of the cask. This approach, described in detail in Section M.5.2.4, is consistent with the method used to determine the fuel qualification tables for the Standardized NUHOMS® canister designs described in Section 7.2.3.

The radiological source terms generated in the SAS2H/ORIGEN-*S* runs are used in the ANISN evaluations to calculate the surface dose rates. The ANISN models are similar to the appropriate DORT models for the locations of interest. Heat load configuration 2 (Figure M.2-2) *is used in the DORT analysis.* The HSM design-basis source terms for the outer ring of assemblies (modeled as sixteen assemblies) are from fuel with 41 GWd/MTU burnup, an initial enrichment of 3.1 wt. % U-235 and 5-years cooling. The HSM design-basis source terms for the inner sixteen assemblies are from fuel with 30 GWd/MTU burnup, and initial enrichment of 2.5 wt. % U-235 and 8-years cooling. Note that using this approach in modeling the outer ring of sixteen assemblies with the 1.2 kW source terms for all of the shielding analyses results in fairly conservative dose rates because the shielding analysis is in reality based on a 28.8 kW heat load. The TC design-basis source terms for the outer ring of assemblies (conservatively modeled as sixteen assemblies) are from fuel with 41 GWd/MTU burnup, an initial enrichment of 3.1 wt. % U-235 and 5-years cooling. The TC design-basis source terms for the inner sixteen assemblies are from fuel with 45 GWd/MTU burnup, and initial enrichment of 3.3 wt. % U-235 and 23-years cooling.

A sample SAS2H/ORIGEN-S input file for the Active Fuel Region for the 41 GWd/MTU, 3.1 wt. % U-235 and 5-years cooling case is listed and commented in Section M.5.5.1.

M.5.2.1 Gamma Source

Four SAS2H/ORIGEN-S runs are required for each burnup/initial enrichment/cooling time combination to determine gamma source terms for the four regions of interest for each fuel assembly; the bottom, active fuel, plenum and top regions. The only difference between the runs is in Block #10 "Light Elements" of the SAS2H input and the 81$$ card in the ORIGEN-S input. Each run includes the appropriate Light Elements for the region being evaluated and the 81$$ card is adjusted to have ORIGEN-S output the total gamma source for the active fuel region and only the light element source for the plenum and top nozzle regions.

The design-basis source terms for the authorized CC designs are taken from Appendix J of the FSAR *and are provided in Table M.5-12*. The design-basis CC source terms from Appendix J of the FSAR are based on three bounding BPRA designs: 1) B&W 15x15 Burnable Absorber Assemblies with up to 2 cycles burnup and 5-year cooled, 2) WE 17x17 Pyrex Burnable Absorber, 2-24 Rodlets with up to 2 cycles burnup and 10-year cooled, and 3) WE 17x17 WABA Burnable Absorber, 3-24 Rodlets with up to 2 cycles burnup and 10-year cooled.

All other CCs, including BPRA types other than the three analyzed above, must be examined on a case-by-case basis to demonstrate that they are bounded by the design basis CC source. Specifically, the maximum allowed CC gamma source is 3.91E+13 γ/s/assembly for BPRAs, NSAs, CRAs, and BAs and 4.1E+12 γ /s/assembly for TPAs.

The SAS2H/ORIGEN-S gamma ray source is output in the CASK-81 energy group structure.

Gamma source terms for the active fuel region include contributions from actinides, fission products, and activation product. The bottom, plenum and top nozzle regions include the contribution from the activation products in the specified region only. These results for the 41 GWd/MTU, 3.1 wt. % U-235 and 5-years cooling case are shown in Table M.5-9. The results for the 30 GWd/MTU, 2.5 wt. % U-235 and 8-years cooling case are shown in Table M.5-10. Finally, the results for the 45 GWd/MTU, 3.3 wt. % U-235 and 23-years cooling case are shown in Table M.5-11.

Gamma source terms for use in the shielding models are calculated by multiplying the assembly sources by the number of assemblies in the region of interest (16) and dividing by the appropriate inner/outer heat load region volume. The appropriate assembly region volumes for both the inner and outer heat load zones are listed in Table M.5-13.

M.5.2.2 Neutron Source Term

One SAS2H/ORIGEN-S run is required for each burnup/initial enrichment/cooling time combination to determine the total neutron source terms for the active fuel regions. The results for each burnup/initial enrichment/cooling time combination of interest are summarized in Table M.5-14.

Neutron source terms for use in the shielding models are calculated by multiplying the assembly sources by the number of assemblies in the active fuel region of interest (16) and dividing by the appropriate active fuel inner/outer heat load region volume. The appropriate assembly region volumes for both the inner and outer heat load regions are listed in Table M.5-13.

M.5.2.3 Axial Peaking

Axial peaking factors for both neutron and gamma sources in PWR fuel are taken from Reference [5.6]. These peaking factors were derived from work performed by the Department of Energy in support of its Topical Report for burnup credit [5.7]. The neutron and gamma peaking factors are shown as a function of the core height in Table M.5-15. These factors are directly applied to each DORT interval in the fuel region. Neutron peaking factors in each zone are equal to the gamma factor raised to the fourth power to correctly account for the variation of neutron source with burnup. The axial source distribution defined in Table M.5-15 introduces some level of conservatism into this calculation because the length average peaking factor of 1.06 is greater than 1.

M.5.2.4 ANISN Evaluation for Bounding Source Terms

SAS2H is used to calculate the minimum required cooling time as a function of assembly initial enrichment and burnup for each decay heat limit. To determine which configuration and burnup, wt. % initial enrichment and cooling time combinations result in the bounding dose rates on the surface of the HSM and TC, the total source term, which includes the contribution from the fuel as well as the hardware in the entire assembly (including end fittings) is used to calculate its total ANISN dose rate on the HSM roof and TC radial using the ANISN code.

The CC contribution is fixed and is included in the design basis shielding evaluation as such and therefore is not included in this ANISN evaluation.

ANISN [5.5] determines the fluence of particles throughout one-dimensional geometric systems by solving the Boltzmann transport equation using the method of discrete ordinates. Particles can be generated by either particle interaction with the transport medium or extraneous sources incident upon the system. Anisotropic cross-sections can be expressed in a Legendre expansion of arbitrary order.

The ANISN code implements the discrete ordinates method as its primary mode of operation. Balance equations are solved for the flow of particles moving in a set of discrete directions in each cell of a space mesh and in each group of a multigroup energy structure. Iterations are performed until all implicitness in the coupling of cells, directions, groups, and source regeneration is resolved.

ANISN coupled with the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.3] and the ANSI/ANS-6.1.1-1977 flux-to-dose conversion factors [5.10] is chosen to generate the ANISN dose rates used to determine the relative strength of the various source terms from fuel assemblies to determine the design basis source terms for the HSM and TC. These design basis source terms are used with DORT to calculate the bounding system dose rates. ANISN provides an efficient method to calculate the design basis source terms.

The surface dose rates are calculated using individual ANISN models to perform the evaluation for the fuel assembly parameters in the fuel qualification table. The ANISN model used to calculate the relative dose rates on the HSM surface is similar to the cut through the center of the DORT HSM roof model used for the shielding evaluation (for each configuration). The ANISN model used to generate the relative dose rates on the TC is similar to the cut through the center of the DORT TC side model used for the shielding evaluation. Figure M.5-31 and Figure M.5-32 provide sketches for the ANISN models of the HSM roof and TC centerline, respectively. When modeling 0.63 kW or 0.60 kW source region in Region A (16 assemblies) of Figure M.5-31 and Figure M.5-32, the Region B does not include any source terms. Similarly, when modeling 0.87 kW or 1.2 kW source region in Region B (16 assemblies), of these figures, the Region A does not include any source terms.

For modeling 0.7 kW source region, both Region A and Region B (32 total assemblies) include source terms corresponding to 0.7 kW. An example ANISN input file is included in Section M.5.5.5.

The material densities used in the ANISN models for the various model regions are listed in Table M.5-27. These material densities are very similar to those used for the DORT and MCNP analysis, but are simplified to reduce the size of the ANISN input decks. Only the important elements of a give material are included and the gram density of the material is maintained. The density of NS-3 is used instead of water in ANISN models. This has no impact on the results of ANISN evaluation because ANISN is only used to compare the relative strength of the source terms for each entry in the fuel qualification table.

To determine the design basis source terms to be used in the HSM shielding calculations, the total roof surface dose rates for the configurations shown in Figure M.2-1, Figure M.2-2 and Figure M.2-3 are calculated. Configuration 1 consists of sixteen 0.63 kW/FA inner assemblies and sixteen 0.87 kW/FA outer assemblies, Configuration 2 is based on bounding assumption of sixteen 0.6 kW/FA inner assemblies and sixteen 1.2 kW/FA outer assemblies, and Configuration 3 consists of thirty-two 0.7 kW/FA assemblies.

For Configuration 1, the maximum total roof dose rate is 35.0 plus 0.19 or 35.2 mrem/hr. For Configuration 2, the total roof dose rate is 46.87 plus 0.16 or 47.03 mrem/hr, while for Configuration 3, the total roof dose rate is 28.4 mrem/hr. Based on these results, Configuration 2 results in the maximum dose rates for the HSM. *(Note that Configuration 4 is not evaluated here because Configuration 4 did not exist at the time the DORT analysis was performed. Configuration 4 is evaluated in Sections M.5.2.7 and M.5.4.16).* Design basis source terms for the outer ring of assemblies (conservatively modeled as sixteen assemblies) are from fuel with 41 GWd/MTU burnup, an initial enrichment of 3.1 wt. % U-235 and 5 years cooling. The design basis source terms for the inner sixteen assemblies are from fuel with 30 GWd/MTU burnup, and initial enrichment of 2.5 wt. % U-235 and 8 year cooling. Note that using this approach in modeling, the outer ring of sixteen assemblies with the 1.2 kW source terms, results in fairly conservative dose rates because the shielding analysis is in reality based on a 28.8 kW heat load.

Similarly, to determine the design basis source terms to be used in the Transfer Cask shielding calculations, the total side centerline surface dose rates for the configurations shown in Figure M.2-1, Figure M.2-2 and Figure M.2-3 are calculated using the results provided in Table M.5-28 through Table M.5-37. For Configuration 1, the total side dose rate is 404.86 plus 34.57 or 439.43 mrem/hr. For Configuration 2, the total side dose rate is 593.8 plus 32.08 or 625.88 mrem/hr, while for Configuration 3, the total roof dose rate is 331.4 mrem/hr. *(Note that Configuration 4 is not evaluated here because Configuration 4 did not exist at the time the DORT analysis was performed. Configuration 4 is evaluated in Sections M.5.2.7 and M.5.16.)* Based on these results, Configuration 2 results in the maximum dose rates for the TC. Design basis source terms for the outer ring of assemblies (conservatively modeled as sixteen assemblies) are from fuel with 41 GWd/MTU burnup, an initial enrichment of 3.1 wt. % U-235 and 5 years cooling. The design basis source terms for the inner sixteen assemblies are from fuel with 45 GWd/MTU burnup, and initial enrichment of 3.3 wt. % U-235 and 23 year cooling.

ANISN *may be* used to develop a "response function" to calculate the gamma and neutron dose rates. The ANISN response functions receive as input the neutron and gamma source terms as calculated by SAS2H. The response function methodology is numerically equivalent to modeling each individual case in ANISN.

Separate response functions are generated for the source in the inner 16 assemblies (for 0.63 and 0.6 kW/assembly heat loads), outer 16 assemblies (for 1.2 and 0.87 kW/assembly heat loads), and all 32 assemblies (for 0.7 kW/assembly heat load). These response functions are provided in Table M.5-38 through Table M.5-43.

To generate a gamma response function, a separate ANISN model is executed with a single gamma per assembly in each of the 18 CASK-81 gamma groups. Once the dose rate resulting from a single gamma per assembly is known for each energy group, the dose rate for a given gamma source can be determined simply by multiplying the source strength in each group by the dose rate contribution for that group and summing the results.

A neutron response function is generated in a similar fashion as a gamma response function, although only one ANISN neutron file is required because the neutron spectrum is adequately represented by the Cm-244 spectrum. Therefore, the ANISN model is executed with one neutron per assembly distributed throughout all 22 groups. The dose rate of secondary neutron capture gammas is calculated by ANISN in addition to the neutron dose rate. This method allows for the calculation of the neutron and capture gamma dose rate on the surface of the TC or HSM knowing only the magnitude of the neutron source.

M.5.2.5 Reconstituted Fuel

As explained in Section M.5.2, reconstituted fuel assemblies may contain up to 56 stainless steel rods that replace fuel rods. Because steel rods replace fuel rods, the decay heat of a reconstituted assembly is typically less than the decay heat of an equivalent standard assembly. Conversely, because steel contains Co-59 which activates to form Co-60, for low cooling times a reconstituted assembly typically generates higher dose rates than an equivalent standard assembly.

To quantify this statement, additional SAS2H runs are generated for reconstituted assemblies. The SAS2H input files for a reconstituted assembly with 56 stainless steel rods are very similar to the input files for a standard assembly except for the following changes: (1) The number of fuel rods is reduced from 208 to 152 to account for 56 stainless steel rods, (2) the POWER input variable is adjusted to maintain the correct burnup for the reduced fuel loading, and (3) the light elements change to reflect that 56 fuel rods have been replaced with steel rods.

Note that a reconstituted rod cannot be irradiated for more than two cycles because the first cycle will always contain fresh, undamaged fuel. To accurately model this behavior, two SAS2H models are generated for each transition point. The first SAS2H model is for only one cycle of irradiation of 56 reconstituted rods, while the second SAS2H model is for three cycles of irradiation of 56 reconstituted rods. By subtracting the single cycle source term of the reconstituted rods from the total source term (fuel and reconstituted rods) for three cycles, the source term for three cycle irradiation of fuel and two cycle irradiation of reconstituted rods is generated.

This source term is inserted into the HSM and OS197/OS197H TC response functions to determine the dose rates for comparison to the design basis source dose rates. If the reconstituted fuel dose rate for either the HSM or OS197/OS197H TC exceeded the dose rate with design basis fuel, cooling time is increased for the reconstituted fuel source term calculation. When the reconstituted fuel with 56 stainless steel rods is analyzed using this approach, no more than 6 additional years of cooling time is required for reconstituted fuel to be bounded by the design basis source.

Similarly, for reconstituted fuel assemblies with a maximum of 10 stainless steel rods, a maximum of 1.5 years of additional cooling time is required for reconstituted fuel to be bounded by the design basis source.

Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.

M.5.2.6 SCALE6.0/ORIGEN-ARP Fuel Qualification Tables

SCALE6.0/ORIGEN-ARP is used to develop unified fuel qualification tables (FQTs). These FQTs apply to the 32PT, 32PTH1, 24PTH, and 37PTH DSCs, as the FQTs are developed solely on heat load. These FQTs are provided in the Technical Specifications, Tables 1-3a through 1-3p.

FQTs are developed for three different uranium loadings: 0.492 MTU, 0.475 MTU, and 0.380 MTU. (Note that the 32PT DSC is limited to a maximum of 0.475 MTU.) Because cooling times are selected to target specific decay heat values and decay heat is proportional to the uranium loading, the FQT cooling times decrease with decreasing uranium loading to maintain the same heat load. In most cases, the uranium loading of a fuel assembly will fall between the 0.492 MTU, 0.475 MTU, and 0.380 MTU values. In such cases, the cooling time interpolation methodology described in Section M.5.6 may be employed.

The FQTs are developed to target decay heat values 10 watts below the licensing limits prescribed in the Heat Load Zone Configurations for each fuel assembly. This is to account for the presence of CCs, which have a maximum decay heat of 8 watts. Also, the minimum fuel assembly cooling time is 2 years.

Each FQT contains an unanalyzed zone marked in the FQTs as gray. Limited extrapolation of FQT cooling times into the unanalyzed regions is allowed. The extrapolation may be performed for a maximum difference of 4 GWd/MTU in burnup or 0.4 wt.% in enrichment. The extrapolation may be performed for either fixed enrichment (variable burnup, fixed FQT column) or fixed burnup (variable enrichment, fixed FQT row). The methodology is:

1. *Perform a regression analysis on the FQT cooling times and associated variable (either burnup or enrichment). Note: All FQT cooling times in either the row or column of data being extrapolated shall be used, even if many of the cooling times are the same.*
2. *Develop a fitting equation for the data. A fourth-order polynomial with parameters having at least six significant digits to avoid rounding errors is recommended.*
3. *Use the fitting equation to compute the extrapolated cooling time at the desired enrichment or burnup.*
4. *Add 0.2 years as additional margin.*

Because extrapolation may be performed on either an FQT row or column of data, in some cases extrapolation to the same FQT cell could be achieved using either data set. It is possible that the extrapolating equations with two alternative sets of parameters may result in slightly different predictions for the cooling times. However, either of the predicted values are legitimate to use because they are both conservative.

An example is provided for extrapolating the 475 kgU FQT for a 2.5 kW fuel assembly at a fixed burnup of 56 GWd/MTU. Note that only built-in capabilities of MS Excel® are utilized in this example. The cooling times are extracted from Technical Specifications Table 1-3o and summarized in the table below. The minimum enrichment in the analyzed region is 2.9 wt.%, and the cooling time for an enrichment of 2.5 wt.% is desired.

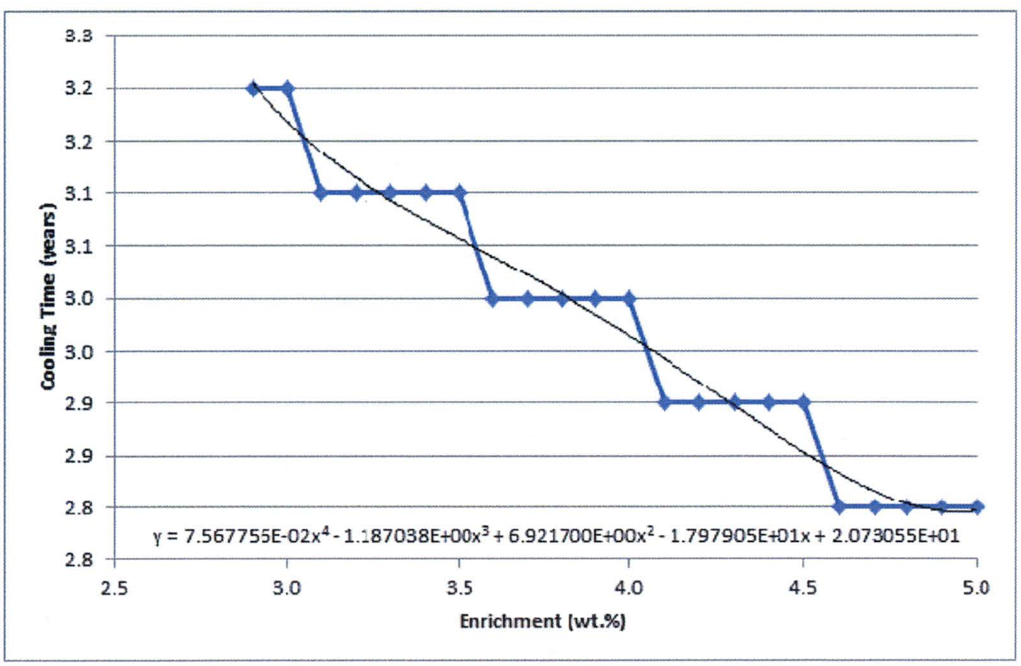
Example Extrapolation FQT Data

Enrichment (wt.%)	*Cooling time (years)*	*Enrichment (wt.%)*	*Cooling time (years)*
2.9	*3.2*	*4.0*	*3.0*
3.0	*3.2*	*4.1*	*2.9*
3.1	*3.1*	*4.2*	*2.9*
3.2	*3.1*	*4.3*	*2.9*
3.3	*3.1*	*4.4*	*2.9*
3.4	*3.1*	*4.5*	*2.9*
3.5	*3.1*	*4.6*	*2.8*
3.6	*3.0*	*4.7*	*2.8*
3.7	*3.0*	*4.8*	*2.8*
3.8	*3.0*	*4.9*	*2.8*
3.9	*3.0*	*5.0*	*2.8*

The plot of the data is shown in the figure below. The fitting equation from the figure is:

$$Y = 7.567756E\text{-}02x^4 - 1.187038x^3 + 6.921700x^2 - 17.97905x + 20.73055$$

For x = 2.5 wt.%, y = cooling time = 3.5 years. An additional 0.2 years is added to this value for a final extrapolated cooling time of 3.7 years.



Example Extrapolation Data Plot

M.5.2.7 SCALE6.0/ORIGEN-ARP Source Terms

Because the original FQTs have been replaced with unified FQTs (Technical Specifications Tables 1-3a through 1-3p), the design basis source terms developed in Section M.5.2 are obsolete because they are based upon burnup, enrichment, and cooling time combinations that are no longer applicable. Therefore, SCALE6.0/ORIGEN-ARP design basis source terms are developed based on the unified FQTs.

It is demonstrated in Section M.5.2 that Heat Load Zone Configuration 2 (HLZC#2) bounds Configurations 1 and 3. However, the analysis in Section M.5.2 does not consider HLZC#4, which was added at a later time. Comparing HLZC#2 and #4, it is observed that the 1.2 kW corner fuel assembles of HLZC#2 have been replaced with 2.2 kW fuel assemblies in HLZC#4. Because gamma source terms correlate with decay heat, HLZC#4 is the bounding configuration for dose rate analysis.

The methodology used to develop the SCALE6.0/ORIGEN-ARP design basis source terms is the same as described in Section M.5.2. However, because HLZC#4 is more non-uniform than HLZC#2, the three-zone 32PTH1 DSC transfer cask response functions from Chapter U.5, Table U.5-15, are used. The response functions are used to evaluate the source terms for each FQT burnup, enrichment, and cooling time (BECT) combination. The BECT combination that results in the maximum dose rate is selected as the design basis source. Because the response functions are only used to rank the BECT combinations, using the 32PTH1 DSC response functions for the 32PT DSC HLZC#4 is acceptable.

Based on the ANISN response function analysis, the SCALE6.0/ORIGEN-ARP design basis source terms for 32PT DSC HLZC#4 in the OS197 transfer cask are:

- *0.4 kW: 45 GWd/MTU, 1.1% enrichment, 36.4 years cooling (Table M.5-49)*
- *0.6 kW (inner): 45 GWd/MTU, 1.1% enrichment, 18.2 years cooling (Table M.5-50)*
- *0.6 kW (outer): 31 GWd/MTU, 1.1% enrichment, 6.6 years cooling (Table M.5-51)*
- *2.2 kW: 45 GWd/MTU, 1.1% enrichment, 2.7 years cooling (Table M.5-52)*

Source terms are provided only for the minimum uranium loading of 0.380 MTU. While 0.380 MTU and 0.475 MTU source terms are similar for equivalent decay heat, the 0.380 MTU source terms result in bounding dose rates compared to 0.475 MTU source terms because the self-shielding is reduced.

The MCNP5 neutron models are run with the NONU card to suppress subcritical neutron multiplication. Subcritical neutron multiplication is addressed by multiplying the neutron source computed by ORIGEN-ARP by $1/(1 - k_{eff})$. Values of k_{eff} appropriate for the burnups of the sources are provided in the source term tables (Table M.5-49 through M.5-52).

The gamma source in the active fuel region is modeled with the axial burnup profile appropriate for the burnup of the source and is obtained from Table 20 of ORNL/TM-12973 [5.16]. These profiles are summarized in Table M.5-53. The neutron profile is derived as the 4^{th} power of the gamma profile and is also summarized in Table M.5-53. The burnup peaking factor accounts for the increase in the neutron source magnitude due to the axial burnup profile. The burnup peaking factors used in the neutron calculations are provided in the source term tables (Table M.5-49 through M.5-52).

The CC source terms provided in Table M.5-12 are applicable and may be added to the fuel-only source terms provided in Table M.5-49 through Table M.5-52.

M.5.4 Shielding Evaluation

Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from 32 *0.490 MTU* fuel assemblies with CCs are calculated with the DORT Code [5.2] at various location*s* on and around the NUHOMS® -32PT DSCs, HSM, and OS197/OS197H TC. AMD 15 72.48

The radiation shielding evaluation for the Standardized NUHOMS® System during loading, transfer and storage for the 24P and 52B canisters is discussed in Sections 3.3.5, 7.0 and 8.0 of the FSAR. The following shielding evaluation discussion specifically addresses the NUHOMS®-32PT-S100/32PT-L100 and 32PT-S125/32PT-L125 DSCs in an HSM and TC using the *0.490 MTU* design-basis source terms determined in Section M.5.2.

Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from 32 0.380 MTU fuel assemblies with CCs are also calculated, but with the MCNP5 Code [5.17], at various locations on and around the NUHOMS® 32PT DSCs within the HSM and OS197/OS197H TC. AMD 15

The shielding evaluation that determines the effect of loading 0.380 MTU per assembly on the dose rates is described in Section M.5.4.16. AMD 15

M.5.4.1 Computer Programs

DORT [5.2] determines the fluence of particles throughout one-dimensional or two-dimensional geometric systems by solving the Boltzmann transport equation using either the method of discrete ordinates or a diffusion theory approximation. Particles can be generated by either particle interaction with the transport medium or extraneous sources incident upon the system. Anisotropic cross-sections can be expressed in a Legendre expansion of arbitrary order.

The DORT code implements the discrete ordinates method as its primary mode of operation. Balance equations are solved for the flow of particles moving in a set of discrete directions in each cell of a space mesh and in each group of a multigroup energy structure. Iterations are performed until all implicitness in the coupling of cells, directions, groups, and source regeneration is resolved.

DORT was chosen for *the 0.490 MTU analysis* because of its ability to solve two dimensional, cylindrical, deep penetration radiation transport problems similar to the NUHOMS® System. AMD 15

The MCNP code, MCNP5 [5.17] with the continuous energy ENDFB-VI cross section library is employed to determine the 0.380 MTU/FA dose rates for the shielding analysis described in Section M.5.4.16. MCNP5 has been employed to perform the shielding analysis of the 24PTH System (Appendix P.5), the 32PTH1 System (Appendix U.5), and the 37PTH System (Appendix Z.5). AMD 15

M.5.4.2 Spatial Source Distribution

The source components are:

- The neutron sources due to the active fuel regions of the inner sixteen and outer sixteen fuel assemblies,

- The gamma source due to the active fuel regions of the inner sixteen and outer sixteen fuel assemblies,
- The gamma source due to the plenum regions of the inner sixteen and outer sixteen fuel assemblies,
- The gamma source due to the top regions of the inner sixteen and outer sixteen fuel assemblies,
- The gamma source due to the bottom region of the inner sixteen and outer sixteen fuel assemblies,

M.5.4.13.4 Outer Cover Welding 32PT-S100/32PT-L100 DSC Configuration

The dose rates during outer cover welding for 32PT-S100/32PT-L100 DSC configuration are shown in Figure M.5-28.

M.5.4.14 Supporting 3-D Analysis

The purpose of this section is to provide supportive three-dimensional shielding analysis of the NUHOMS®-32PT System to demonstrate that the 2-D DORT analysis described above bounds a three dimensional analysis of the HSM with the bounding 32PT canister and source terms. As demonstrated in Table M.5-3 and Table M.5-4, the bounding 32PT canister is the 32PT-S100/32PT-L100 DSC Design Configuration with eight (modeled as 16) 1.2 kW (41 GWd/MTU, 3.1 wt. % U-235, 5-year cooled fuel) fuel assemblies and 24 (modeled as 16) 0.6 kW (30 GWd/MTU, 2.5 wt. % U-235, 8-year cooled fuel) fuel assemblies with 32 design basis *CCs*. The source terms (Section M.5.2) and material densities (Section M.5.3.1) used in the MCNP models are the same as those used in the DORT models including the axial peaking factors described in Section M.5.2.3. Figure M.5-29 and Figure M.5-30 are MCNP generated cuts through the 3-D model of the canister and HSM. The figures show the shielding materials and thicknesses as modeled with MCNP. An example input deck is included in Section M.5.5.4.

To demonstrate that the 2-D DORT analysis bounds a 3-D analysis the peak neutron and gamma dose rates on the roof and front vents; roof surface average; and peak neutron and gamma dose rates on the HSM door are compared. The results are provided in Table M.5-24. As shown by the results reported in Table M.5-24, the 2-D DORT analysis bounds the 3-D MCNP analysis.

In addition, this 3-D shielding analysis with MCNP has been validated by comparison to actual measured dose rate data from installed NUHOMS® Systems.

The actual measured dose rate data is from surveys taken at which utilizes the NUHOMS®-24P storage system. The HSM Model 80 and B&W 15x15 mark B fuel assembly type are identical to that evaluated with the 32PT System. MCNP models are developed for the stored canister and the HSM. The MCNP models were used to calculate dose rates at locations around the HSM that correspond to the various survey locations for which the data is reported.

The results of these benchmark runs are shown in Table M.5-25 and Table M.5-26.

The results show that MCNP predicts conservatively higher total (neutron plus gamma) dose rates compared to the measured data. For those two points (neutron dose on the roof bird screens for DSC 45 and gamma dose rate on HSM door) where the measured data is higher than the calculated data, it should be noted that the magnitude of these dose rates is small. However, at these two locations the calculated total dose rate is still higher than the measured dose rates. Therefore, a ratio of 0.8 for calculated/measured dose rates for the aforementioned two points is regarded as a fairly accurate estimate. Some conservatism still exists in the methodology used to calculate the source terms, and this is why the calculated dose rates are, in general, higher than the measured dose rates.

M.5.4.15 Impact on Dose Rates due to Reduced Density Concrete and Gaps between HSMs

A bounding analysis is performed by employing a minimum concrete density of 140 pounds per cubic foot (pcf) in the HSM-H MCNP model combined with a maximum gap of 1.5 inches between adjacent HSM-H modules and shield walls to determine the effect on maximum and average dose rates due to a fully loaded 32PTH1 DSC. These calculations are documented in Appendix U.5, Section U.5.4.10. The ratios shown in Appendix U.5, *Table U.5-18* and *Table U.5-19* can be used as scaling factors to increase the maximum and surface-average dose rates of the *32PT DSC* in the HSM-H to account for low density concrete and 1.5-inch gaps during HSM fabrication and installation. Note that the HSM-H concrete contains high density rebar, which is not credited in the MCNP models. Further, the modules are installed adjacent to each other such that there will not be a "uniform" gap of 1.5 inches. Ignoring the effect due to increased vent dose rates, the increase in the average dose rates caused by both the maximum postulated uniform gaps and the minimum postulated concrete density is expected to be less than 20% at the front and roof surfaces of the HSM-H module. Dose reduction hardware may be installed to further reduce these dose rates.

M.5.4.16 Shielding Analysis with a Loading of 0.380 MTU per Fuel Assembly, Updated Bounding Source Terms, and New HLZC 4

The 32PT/OS197TC and 32PT/HSM dose rates computed by DORT and reported in Tables M.5-3, M.5-4, M.5-5, and M.5-23 must be scaled up to account for the following effects: (1) HLZC 4, which bounds HLZC 2 used in the DORT runs, (2) the source terms developed in Section M.5.2.7, which are more penalizing than the design basis source terms used in the DORT analysis, and (3) 0.380 MTU fuel assemblies, which results in a self-shielding penalty compared to the 0.475 MTU fuel assemblies used in the DORT analysis.

Dose rate calculations are performed using 3-D MCNP5 models. The approach is performed as follows: First, MCNP models are created to supplement the DORT analysis contained in this appendix. The maximum gamma, neutron, and total side on contact dose rates for the 32PT-L125 DSC in the OS197 TC using MCNP5 and 0.475 MTU fuel with HLZC 4 are calculated for the normal transfer configuration.

*The MCNP5 model geometry is similar to Figure M.5-1, although the transition rails are modeled as solid aluminum, consistent with the licensing drawings. (The model depicted in Figure M.5-1 is used only to justify the DORT basket model homogenization assumptions, as detailed in Section M.5.4.7.) The HLZC 4 loading configuration is asymmetric, resulting in the heat load being higher in the upper hemisphere compared to the lower hemisphere. Only the upper right quadrant is modeled, with reflective boundaries modeled on the west and south surfaces. Modeling quarter symmetry is conservative because it results in a total basket heat load of (0.4+0.6*5+2.2*2)*4 = 31.2 kW rather than the HLZC 4 total of 24 kW.*

The models are then benchmarked by comparing the MCNP results to those found using DORT with 0.475 MTU fuel for the normal transfer configuration. This comparison verifies that the MCNP models produce results which are conservatively representative for the 32PT-L125 DSC in the OS197 TC.

The models are then used to generate reduced MTU dose rate results. The maximum gamma, neutron, and total side on contact dose rates for the 32PT-L125 DSC in the OS197 TC using MCNP5 and 0.380 MTU fuel using HLZC 4 are calculated for the normal transfer configuration.

With only 1 dose rate data point available from the MCNP models, it is necessary to use a scaling factor approach in order to update the other data contained in Tables M.5-3, M.5-4, M.5-5, and M.5-23. The shielding characteristics of the 32PT DSC in the OS197 TC are shown to be conservatively approximated by the 32PTH1 DSC in the OS200 TC by comparing the scaling factor of the change in dose rates from loading high MTU fuel to loading low MTU fuel for the two systems. This is done as follows:

- *The 32PT-L125/OS197 TC 0.380 MTU/HLZC 4 side on contact total dose rate results are compared to the 0.475 MTU/HLZC 2 side on contact total dose rate results to create a side on contact total dose rate scaling factor for the normal transfer configuration.*
- *This scaling factor is compared to the same scaling factor generated for the 32PTH1 DSC in the OS200 TC.*
  - *In Appendix U, Section U.5.4.12, the side total dose rate scaling factor for 0.380 MTU fuel VS. 0.490 MTU fuel for the 32PTH1 DSC in the OS200 TC in the normal transfer configuration is 1.24.*
  - *The same configuration scaling factor for the 32PT-L125 DSC in the OS197 TC is 1.08.*
- *Since the increase in the side on contact total dose rate for the 32PTH1 DSC in the OS200 TC is greater than that found for the 32PTL-125 DSC in the OS197 TC, it is conservative to use the dose rate scaling factors derived for the 32PTH1 System to approximate the dose rate impact of loading 0.380 MTU fuel into the 32PT System. The actual dose rates are not comparable between the two systems, but the change in dose rates from loading high MTU fuel to loading low MTU fuel for 32PT is conservatively approximated by 32PTH1.*

The maximum and average total dose rate scaling factors derived in the 32PTH1 System for both the HSM and TC in Appendix U, Section U.5.4.12 are also used here.

- *The dose rates for the HSM front and roof are to be scaled by 1.18.*
- *The dose rates for the HSM side and rear are to be scaled by 1.36.*
- *The site dose for the HSM is to be scaled by 1.18.*
- *The dose rates for the TC for normal, welding and decontamination are to be scaled as follows:*
  - *by 1.24 for the side,*
  - *by 1.63 for the top,*
  - *by 1.91 for the bottom.*
- *The dose rates for the TC for accidents are to be scaled by 1.13 based on an explicit MCNP5 32PT DSC model.*

- *The occupational exposure for the TC loading and storage operations is to be scaled by 1.25.*

These scaling factors are included as footnotes in the dose rate results summarized in Table M.5-3, M.5-4, M.5-5, and M.5-23.

These scaling factors are also employed to scale the occupational exposure and generic site dose (2X10 back-to-back and front-to-front arrays) results calculated for the 32PT System in Appendix M.10, and to scale the dose rate consequences of accidents for the 32PT System in Appendix M.11.

5.13 *Deleted.*

5.14 *ORNL/TM-2005/39, Version 6, SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Oak Ridge National Laboratory, January 2009.*

5.15 *LA-UR-03-1987, MCNP - A General Monte Carlo N-Particle Transport Code, Version 5, Los Alamos National Laboratory, April 2003.*

5.16 *ORNL/TM-12973, Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages, Oak Ridge National Laboratory, May 1996.*

5.17 *"Monte Carlo N-Particle Transport Code System," CCC-730, Oak Ridge National Laboratory, RSICC Computer Code Collection, August 2001.*

Table M.5-1
PWR Fuel Assembly Design Characteristics[3]

Assembly Class	**B&W 15x15**	**WE 17x17**	**CE 15x15**	**WE 15x15**	**CE 14x14**	**WE 14x14**
DSC Configuration	**Max Unirradiated Length (in)**					
32PT-S100	165.75	165.75	165.75	165.75	165.75	165.75
32PT-L100	171.71[1]	171.71[1]	171.71	171.71	171.71	171.71
32PT-S125	165.75	165.75	165.75	165.75	165.75	165.75
32PT-L125	171.71[1]	171.71[1]	171.71	171.71	171.71	171.71
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Maximum MTU/assembly[2]	0.475	0.475	0.475	0.475	0.475	0.475
Maximum Number of Fuel Rods	208	264	216	204	176	179
Maximum Number of Guide/ Instrument Tubes	17	25	9	21	5	17

(1) Maximum Assembly + CC Length (unirradiated)
(2) The maximum MTU/assembly is based on the shielding analysis.
(3) Maximum Co-59 content in the Top End Fitting Region is 15.6 grams per assembly.
Maximum Co-59 content in the Plenum Region is 5.0 grams per assembly.
Maximum Co-59 content in the Active Fuel Region is 24.7 grams per assembly.
Maximum Co-59 content in the Bottom Region is 12.8 grams per assembly.

Table M.5-3
Dose Rates Due to the 32 PWR Fuel Assemblies with CCs

Dose Rate Location	Configuration 2 32PT-S100/32PT-L100 DSC Design Configuration			Configuration 2 32PT-S125/32PT-L125 DSC Design Configurations		
	Gamma (mrem/hr)	Neutron (mrem/hr)	Total(1) (mrem/hr)	Gamma (mrem/hr)	Neutron (mrem/hr)	Total(1) (mrem/hr)
HSM Roof (centerline) (4)	38.6	0.6	39.2	38.6	0.6	39.2
HSM Roof Birdscreen(4)	1201	16.9	1218	1201	16.9	1218
HSM End Shield Wall Surface(5)	5.4	0.2	5.6	5.4	0.2	5.6
HSM Door Exterior Surface (centerline) (4)	158	36.3	185	77.5	28.1	99.3
HSM Front Birdscreen(4)	780	7.3	788	745	6.8	752
HSM Back Shield Wall(4)	1.45	0.05	1.50	1.37	0.05	1.41
Centerline Top DSC Cover Plate w/3"ns3+1" steel Dry Welding(7)	123	18.7	142	36.7	15.0	51.7
Outer Edge Centerline Top DSC (Peak Annulus) (7)	3834	132	3966	1458	111	1569
Cask Surface (Radial) Contact Normal Condition(6)	784	261	950	784	259	947
3 ft from Cask Surface (Radial) Normal Condition(6)	293	98.3	391	293	97.8	390
Cask Surface (Radial) Contact Accident Condition(9)	1070	3780	4640	1070	3770	4630
Cask Top Axial Surface(7)	94.8	32.6	107	37.7	27.5	48.7
Cask Bottom Axial Surface(8)	758(2)	957(2)	1707(2)	193(3)	770(3)	960(3)

Notes:

(1) Gamma and Neutron peaks do not always occur at same location therefore the total is not always the sum of the gamma plus neutron.

(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 170 mrem/hr gamma, 115 mrem/hr neutron for a total average dose rate of 285 mrem/hr.

(3) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 48.7 mrem/hr gamma, 92.3 mrem/hr neutron for a total average dose rate of 141 mrem/hr.

(4) These dose rates increase by 18% when loading 0.380 MTU FAs.
(5) These dose rates increase by 36% when loading 0.380 MTU FAs.
(6) The Side dose rates increase by 24% when loading 0.380 MTU FAs.
(7) The Top dose rates increase by 63% when loading 0.380 MTU FAs.
(8) The Bottom dose rates increase by 91% when loading 0.380 MTU FAs.
(9) The Accident dose rates increase by 13% when loading 0.380 MTU FAs.

Table M.5-4[2]
Summary of HSM Dose Rates

Surface	Dose Rate Component	Configuration 2 32PT-S100/32PT-L100 DSC Configuration		Configuration 2 32PT-S125/32PT-L125 DSC Configurations	
		Maximum Dose Rate (mrem/hr)	Surface Average Dose Rate (mrem/hr)	Maximum Dose Rate (mrem/hr)	Surface Average Dose Rate (mrem/hr)
Rear[1] *[4]*	Gamma	1.45	0.48	1.37	0.45
	Neutron	0.05	0.02	0.05	0.02
Front*[3]*	Gamma	780	86.4	745	56.2
	Neutron	7.3	9.1	6.8	7.2
Roof*[3]*	Gamma	1201	54.5	1201	54.2
	Neutron	16.9	0.75	16.9	0.75
Side[1] *[4]*	Gamma	5.4	1.7	5.4	1.7
	Neutron	0.2	0.05	0.2	0.05

Notes:
(1) Includes 24 inch shield wall.
(2) Dose rates can be higher by 6% to account for the use of grout during HSM fabrication and installation.
(3) These dose rates increase by 18% when loading 0.380 MTU FAs.
(4) These dose rates increase by 36% when loading 0.380 MTU FAs.

Table M.5-5
Summary of Onsite TC Dose Rates (Maximum[(1)])

	Configuration 2 32PT-S100/32PT-L100 DSC Configuration			Configuration 2 32PT-S125/32PT-L125 DSC Configuration		
	Cask Surface			**Cask Surface**		
	Side[(4)] (mrem/hr)	Top[(5)] (mrem/hr)	Bottom[(2) (6)] (mrem/hr)	Side[(4)] (mrem/hr)	Top[(5)] (mrem/hr)	Bottom[(3) (6)] (mrem/hr)
Neutron	2.61E+02	3.26E+01	9.57E+02	2.59E+02	2.75E+01	7.70E+02
Gamma	7.84E+02	9.48E+01	7.58E+02	7.84E+02	3.77E+01	1.93E+02
Total	9.50E+02	1.07E+02	1.71E+03	9.47E+02	4.87E+01	9.60E+02
	1-Meter from Cask Surface			**1-Meter from Cask Surface**		
	Side[(4)] (mrem/hr)	Top[(5)] (mrem/hr)	Bottom[(2) (6)] (mrem/hr)	Side[(4)] (mrem/hr)	Top[(5)] (mrem/hr)	Bottom[(3) (6)] (mrem/hr)
Neutron	9.83E+01	1.13E+01	8.95E+01	9.78E+01	9.58E+00	7.14E+01
Gamma	2.93E+02	1.69E+01	1.85E+02	2.93E+02	6.63E+00	4.92E+01
Total	3.91E+02	2.52E+01	2.74E+02	3.90E+02	1.45E+01	1.21E+02
	2-Meters from Cask Surface			**2-Meters from Cask Surface**		
	Side[(4)] (mrem/hr)	Top[(5)] (mrem/hr)	Bottom[(2) (6)] (mrem/hr)	Side[(4)] (mrem/hr)	Top[(5)] (mrem/hr)	Bottom[(3) (6)] (mrem/hr)
Neutron	5.26E+01	4.47E+00	2.83E+01	5.24E+01	3.80E+00	2.25E+01
Gamma	1.73E+02	8.87E+00	8.07E+01	1.73E+02	3.28E+00	2.15E+01
Total	2.26E+02	1.31E+01	1.09E+02	2.25E+02	6.82E+00	4.40E+01

Notes:

(1) Gamma and Neutron peaks do not always occur at same location therefore the total is not always the sum of the gamma plus neutron.

(2) This bottom surface dose rate is directly below the grapple ring cut out in the cask bottom. The bottom surface average dose rates are 115 neutron, 170 gamma or 285 total. These average dose rates include the area below grapple ring cutout in the cask.

(3) This bottom surface dose rate is directly below the grapple ring cut out in the cask bottom. The bottom surface average dose rates are 92.3 neutron, 48.7 gamma or 141 total. These average dose rates include the area below grapple ring cutout in the cask.

(4) The Side dose rates increase by 24% when loading 0.380 MTU FAs.

(5) The Top dose rates increase by 63% when loading 0.380 MTU FAs.

(6) The Bottom dose rates increase by 91% when loading 0.380 MTU FAs.

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Table M.5-45
Minimum Cooling Time in Years by Linear Interpolation versus Fitting Equation with PWR Fuel

Burnup / Enrichment	*kgU*	*Decay Heat*						*Method for Determining Cooling Time*
		1.2kW	*0.8kW*	*0.7kW*	*0.6kW*	*0.5kW*	*0.4kW*	
30 GWd/MTU / 1.5wt%	*400*	*3.2*	*4.6*	*5.3*	*6.4*	*8.5*	*14.2*	*Linear Interpolation*
		3.3	*4.6*	*5.3*	*6.5*	*8.6*	*14.4*	*Fitting Equation 3 for PWR*
	380	*3.1*	*4.4*	*5*	*6*	*7.6*	*12.4*	*FQT*
	492	*3.9*	*5.6*	*6.6*	*8.5*	*12.7*	*22.3*	*FQT*
48 GWd/MTU - 2.7wt%	*450*	*5.9*	*13.1*	*18.0*	*24.9*	*34.4*	*47.5*	*Linear Interpolation*
		5.9	*13.3*	*18.2*	*25.3*	*34.8*	*47.9*	*Fitting Equation 3 for PWR*
	380	*4.7*	*8.7*	*11.7*	*17.3*	*26*	*38.2*	*FQT*
	492	*6.6*	*15.7*	*21.7*	*29.5*	*39.5*	*53*	*FQT*
56 GWd/MTU - 3.22wt%	*430*	*7.5*	*18.1*	*24.1*	*31.7*	*41.5*	*55.4*	*Linear Interpolation*
		7.6	*18.5*	*24.5*	*32.1*	*41.9*	*55.9*	*Fitting Equation 3 for PWR*
	380	*5.9*	*13.3*	*18.5*	*25.8*	*35.2*	*48*	*FQT*
	492	*9.5*	*24.1*	*31*	*39*	*49.2*	*64.5*	*FQT*
62 GWd/MTU - 3.4wt%	*475*	*12.4*	*28.3*	*35.3*	*43.7*	*54.7*	*69.9*	*Linear Interpolation*
		12.5	*28.5*	*35.5*	*43.9*	*54.9*	*70.1*	*Fitting Equation 3 for PWR*
	380	*7.3*	*18.3*	*24.5*	*32*	*41.8*	*55.2*	*FQT*
	492	*13.3*	*30.1*	*37.2*	*45.8*	*57*	*72.5*	*FQT*
40 GWd/MTU - 1.5wt%	*410*	*4.4*	*7.6*	*9.9*	*13.8*	*20.6*	*31.9*	*Linear Interpolation*
		4.5	*7.7*	*10.0*	*14.0*	*20.9*	*32.3*	*Fitting Equation 3 for PWR*
	380	*4.1*	*6.6*	*8.2*	*11.4*	*17.5*	*28.4*	*FQT*
	492	*5.4*	*10.3*	*14.4*	*20.3*	*29.2*	*41.5*	*FQT*
16 GWd/MTU - 0.7wt%	*480*	-	-	*3.7*	-	-	-	*Linear Interpolation*
		-	-	*3.7*	-	-	-	*Fitting Equation 3 for PWR*
	380	-	-	*3.2*	-	-	-	*FQT*
	492	-	-	*3.8*	-	-	-	*FQT*
26 GWd/MTU - 1.5wt%	*480*	-	-	*5.3*	-	-	-	*Linear Interpolation*
		-	-	*5.3*	-	-	-	*Fitting Equation 3 for PWR*
	380	-	-	*4.3*	-	-	-	*FQT*
	492	-	-	*5.4*	-	-	-	*FQT*

Table M.5-46

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Proprietary Information on Pages M.5-99j through M.5-99k
Withheld Pursuant to 10 CFR 2.390.

Table M.5-49
HLZC#4 0.4 kW Design Basis Source Term

Bounding Radiological Source at 380 kgU/FA: 45 GWd/MTU, 1.1 wt. %, after 36.4 years of cooling						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle (g/s)	In-core (g/s)	Plenum (g/s)	Top Nozzle (g/s)
0.00e+00	to	5.00e-02	6.498E+09	3.054E+14	1.666E+10	4.571E+09
5.00e-02	to	1.00e-01	8.963E+08	9.062E+13	2.021E+09	6.245E+08
1.00e-01	to	2.00e-01	2.227E+08	4.896E+13	5.031E+08	1.527E+08
2.00e-01	to	3.00e-01	1.281E+07	1.510E+13	3.019E+07	8.792E+06
3.00e-01	to	4.00e-01	1.623E+07	9.904E+12	3.456E+07	1.058E+07
4.00e-01	to	6.00e-01	2.581E+07	7.212E+12	2.238E+07	2.479E+06
6.00e-01	to	8.00e-01	1.569E+09	6.913E+14	7.920E+09	1.320E+09
8.00e-01	to	1.00e+00	1.513E+09	3.701E+12	7.659E+09	1.281E+09
1.00e+00	to	1.33e+00	2.550E+11	7.817E+12	5.621E+11	1.771E+11
1.33e+00	to	1.66e+00	7.200E+10	1.227E+12	1.587E+11	5.001E+10
1.66e+00	to	2.00e+00	4.915E+01	2.508E+10	3.203E+01	6.144E-03
2.00e+00	to	2.50e+00	1.723E+06	1.353E+09	3.798E+06	1.197E+06
2.50e+00	to	3.00e+00	1.472E+03	1.270E+08	3.245E+03	1.022E+03
3.00e+00	to	4.00e+00	5.694E-06	3.727E+07	2.914E-05	4.690E-06
4.00e+00	to	5.00e+00	0.0	1.258E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	5.050E+06	0.0	0.0
6.50e+00	to	8.00e+00	0.0	9.906E+05	0.0	0.0
8.00e+00	to	1.00e+01	0.0	2.103E+05	0.0	0.0
Total Gamma, g/(sec*FA)			3.377E+11	1.181E+15	7.557E+11	2.351E+11
(1)Total Neutrons, n/(sec*FA)			3.650E+8			

(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for subcritical neutron multiplication and an axial variation of burn-up profile in the active fuel region, where k_{eff}= 0.25189 and bpf=1.152.

Table M.5-50
HLZC#4 0.6 kW (Inner Zone) Design Basis Source Term

Bounding Radiological Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 18.2 years of cooling						
E_{min}, MeV	***to***	***E_{max}, MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	*to*	*5.00e-02*	*5.740E+10*	*4.802E+14*	*1.251E+11*	*3.717E+10*
5.00e-02	*to*	*1.00e-01*	*1.002E+10*	*1.283E+14*	*2.209E+10*	*6.944E+09*
1.00e-01	*to*	*2.00e-01*	*2.877E+09*	*8.598E+13*	*5.638E+09*	*1.679E+09*
2.00e-01	*to*	*3.00e-01*	*1.542E+08*	*2.526E+13*	*2.915E+08*	*8.450E+07*
3.00e-01	*to*	*4.00e-01*	*2.665E+08*	*1.578E+13*	*4.195E+08*	*1.099E+08*
4.00e-01	*to*	*6.00e-01*	*2.452E+09*	*2.152E+13*	*1.615E+09*	*8.754E+06*
6.00e-01	*to*	*8.00e-01*	*2.834E+09*	*1.075E+15*	*8.752E+09*	*1.323E+09*
8.00e-01	*to*	*1.00e+00*	*1.635E+09*	*1.622E+13*	*7.925E+09*	*1.365E+09*
1.00e+00	*to*	*1.33e+00*	*2.910E+12*	*5.198E+13*	*6.416E+12*	*2.022E+12*
1.33e+00	*to*	*1.66e+00*	*8.218E+11*	*1.067E+13*	*1.812E+12*	*5.709E+11*
1.66e+00	*to*	*2.00e+00*	*7.753E+01*	*4.080E+10*	*5.053E+01*	*9.583E-03*
2.00e+00	*to*	*2.50e+00*	*1.966E+07*	*2.441E+09*	*4.335E+07*	*1.366E+07*
2.50e+00	*to*	*3.00e+00*	*1.680E+04*	*1.938E+08*	*3.704E+04*	*1.167E+04*
3.00e+00	*to*	*4.00e+00*	*8.411E-06*	*7.505E+07*	*4.305E-05*	*6.928E-06*
4.00e+00	*to*	*5.00e+00*	*0.0*	*2.498E+07*	*0.0*	*0.0*
5.00e+00	*to*	*6.50e+00*	*0.0*	*1.002E+07*	*0.0*	*0.0*
6.50e+00	*to*	*8.00e+00*	*0.0*	*1.967E+06*	*0.0*	*0.0*
8.00e+00	*to*	*1.00e+01*	*0.0*	*4.176E+05*	*0.0*	*0.0*
*Total Gamma, g/(sec*FA)*			*3.810E+12*	*1.911E+15*	*8.399E+12*	*2.641E+12*
*(1)Total Neutrons, n/(sec*FA)*			*7.219E+8*			
(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for a subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where k_{eff}= 0.25189 and bpf=1.152						

All changes on this page are Amd 15.

Table M.5-51
HLZC#4 0.6 kW (Outer Zone) Design Basis Source Term

Bounding Radiological Source at 380 kgU/FA: 31 GWD/MTU, 1.1 wt. %, after 6.6 years of cooling						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle (g/s)	In-core (g/s)	Plenum (g/s)	Top Nozzle (g/s)
0.00e+00	to	5.00e-02	2.545E+11	6.054E+14	4.619E+11	1.301E+11
5.00e-02	to	1.00e-01	3.709E+10	1.628E+14	8.114E+10	2.547E+10
1.00e-01	to	2.00e-01	1.572E+10	1.282E+14	2.397E+10	6.172E+09
2.00e-01	to	3.00e-01	9.254E+08	3.622E+13	1.289E+09	3.061E+08
3.00e-01	to	4.00e-01	2.181E+09	2.393E+13	2.317E+09	4.002E+08
4.00e-01	to	6.00e-01	3.619E+10	2.738E+14	2.361E+10	2.671E+07
6.00e-01	to	8.00e-01	1.997E+10	1.244E+15	1.791E+10	9.437E+08
8.00e-01	to	1.00e+00	2.524E+10	1.236E+14	1.046E+10	1.471E+10
1.00e+00	to	1.33e+00	1.068E+13	1.568E+14	2.359E+13	7.420E+12
1.33e+00	to	1.66e+00	3.015E+12	4.243E+13	6.660E+12	2.095E+12
1.66e+00	to	2.00e+00	5.068E+01	3.544E+11	3.546E+01	1.084E+00
2.00e+00	to	2.50e+00	7.214E+07	4.472E+11	1.594E+08	5.014E+07
2.50e+00	to	3.00e+00	6.163E+04	2.582E+10	1.362E+05	4.284E+04
3.00e+00	to	4.00e+00	6.589E-06	2.438E+09	3.373E-05	5.428E-06
4.00e+00	to	5.00e+00	0.0	1.257E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	5.047E+06	0.0	0.0
6.50e+00	to	8.00e+00	0.0	9.900E+05	0.0	0.0
8.00e+00	to	1.00e+01	0.0	2.102E+05	0.0	0.0
Total Gamma, g/(sec*FA)			1.408E+13	2.798E+15	3.087E+13	9.694E+12
[1]Total Neutrons, n/(sec*FA)			3.605E+8			
[1]This is a "raw" source calculated with ORIGEN-ARP. Multiply it by $bpf/(1-k_{eff})$ to account for a subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where k_{eff}= 0.28950 and bpf=1.265.						

Table M.5-52
HLZC#4 2.2 kW Design Basis Source Term

Bounding Radiological Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 2.7 years of cooling						
E_{min}, MeV	***to***	***E_{max}, MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	to	5.00e-02	7.908E+11	2.822E+15	1.131E+12	2.724E+11
5.00e-02	to	1.00e-01	7.882E+10	8.838E+14	1.707E+11	5.357E+10
1.00e-01	to	2.00e-01	4.259E+10	7.976E+14	5.648E+10	1.301E+10
2.00e-01	to	3.00e-01	2.667E+09	2.245E+14	3.172E+09	6.437E+08
3.00e-01	to	4.00e-01	8.140E+09	1.702E+14	7.179E+09	8.409E+08
4.00e-01	to	6.00e-01	1.261E+11	1.949E+15	8.240E+10	1.538E+08
6.00e-01	to	8.00e-01	6.892E+10	3.317E+15	5.180E+10	1.340E+09
8.00e-01	to	1.00e+00	6.479E+11	7.233E+14	1.197E+11	3.690E+11
1.00e+00	to	1.33e+00	2.245E+13	4.131E+14	4.948E+13	1.559E+13
1.33e+00	to	1.66e+00	6.339E+12	1.334E+14	1.397E+13	4.403E+12
1.66e+00	to	2.00e+00	2.341E+06	5.856E+12	4.927E+06	1.510E+06
2.00e+00	to	2.50e+00	1.517E+08	1.181E+13	3.344E+08	1.054E+08
2.50e+00	to	3.00e+00	1.296E+05	4.553E+11	2.857E+05	9.001E+04
3.00e+00	to	4.00e+00	1.169E-05	4.224E+10	5.981E-05	9.627E-06
4.00e+00	to	5.00e+00	0.0	4.539E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	1.822E+07	0.0	0.0
6.50e+00	to	8.00e+00	0.0	3.574E+06	0.0	0.0
8.00e+00	to	1.00e+01	0.0	7.589E+05	0.0	0.0
Total Gamma, g/(sec*FA)			3.055E+13	1.145E+16	6.508E+13	2.071E+13
[(1)]Total Neutrons, n/(sec*FA)			1.316E+9			
[(1)]This is a "raw" source calculated with ORIGEN-ARP. Multiply it by $bpf/(1-k_{eff})$ to account for a subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where k_{eff}= 0.25189 and bpf=1.152.						

All changes on this page are Amd 15.

Table M.5-53
Axial Profiles

	Gamma Profiles					*Neutron Profiles*				
Axial Zone	*8-16 GWd/MTU*	*16-24 GWd/MTU*	*24-36 GWd/MTU*	*36-44 GWd/MTU*	*44-55 GWd/MTU*	*8-16 GWd/MTU*	*16-24 GWd/MTU*	*24-36 GWd/MTU*	*36-44 GWd/MTU*	*44-55 GWd/MTU*
1	0.488	0.510	0.585	0.624	0.655	0.056	0.067	0.116	0.150	0.183
2	0.745	0.789	0.866	0.896	0.911	0.306	0.386	0.561	0.643	0.687
3	0.914	0.952	0.993	1.010	1.009	0.697	0.821	0.972	1.041	1.037
4	1.029	1.045	1.045	1.044	1.041	1.122	1.194	1.194	1.189	1.175
5	1.133	1.115	1.096	1.077	1.069	1.652	1.549	1.446	1.347	1.308
6	1.154	1.135	1.104	1.080	1.072	1.779	1.664	1.488	1.363	1.322
7	1.178	1.148	1.106	1.080	1.072	1.932	1.742	1.499	1.363	1.322
8	1.191	1.156	1.105	1.079	1.071	2.019	1.791	1.494	1.358	1.318
9	1.199	1.160	1.103	1.079	1.070	2.074	1.816	1.483	1.358	1.313
10	1.203	1.162	1.102	1.078	1.069	2.102	1.829	1.478	1.352	1.308
11	1.203	1.163	1.101	1.078	1.069	2.102	1.835	1.472	1.352	1.308
12	1.198	1.163	1.100	1.078	1.068	2.067	1.835	1.467	1.352	1.303
13	1.189	1.160	1.099	1.078	1.068	2.006	1.816	1.462	1.352	1.303
14	1.174	1.152	1.099	1.077	1.069	1.906	1.766	1.462	1.347	1.308
15	1.144	1.132	1.096	1.075	1.068	1.717	1.646	1.446	1.337	1.303
16	1.091	1.101	1.081	1.070	1.066	1.419	1.472	1.368	1.313	1.293
17	0.961	1.014	1.029	1.040	1.041	0.852	1.057	1.122	1.171	1.175
18	0.820	0.880	0.957	0.986	0.994	0.450	0.598	0.838	0.945	0.976
19	0.616	0.662	0.790	0.863	0.879	0.143	0.190	0.388	0.553	0.595
20	0.367	0.401	0.544	0.613	0.639	0.018	0.025	0.087	0.140	0.165

M.6 Criticality Evaluation

The design criteria for the NUHOMS®-32PT DSC requires that the fuel loaded in the DSC remain subcritical under normal, and accident conditions as defined in 10CFR Part 72.

The NUHOMS®-32PT DSC system's criticality safety is ensured by fixed neutron absorbers, soluble boron in the pool and favorable geometry. Burnup credit is not taken in this criticality evaluation. The fixed neutron absorbers are present in the form of borated metallic plates and *non-irradiated poison rod assemblies (PRAs) with B_4C absorber or Ag-In-Cd absorber,* which are inserted in the guide tubes of certain assemblies in the basket. *The PRA with Ag-In-Cd absorber is designated as AIC PRA while the PRAs with B_4C absorber is designated as B_4C PRA or PRA, as applicable.* These materials are ideal for long-term use in the radiation and thermal environments of a DSC. A basket may contain 0, 4, 8, or 16 PRAs and is designated a Type "A," Type "B," Type "C" or Type "D" basket, respectively. Each of these basket types has the same minimum Boron-10 content of 0.007 gm/cm^2, (90% credit taken in the criticality analysis or 0.0063 gm/cm^2). Two additional basket types, designated as Type A1 and A2, have a higher minimum Boron-10 content of 0.015 gm/cm^2 and 0.020 gm/cm^2, respectively, (90% credit taken in the criticality analysis or 0.0135 gm/cm^2 and 0.018 gm/cm^2, respectively). Type A1 and A2 basket configurations are qualified for the 24 *poison* plate basket design only with 0 PRAs. Metal Matrix Composites (MMCs) at a minimum areal density of 0.0070g/cm^2 have been qualified for use as a neutron absorber with 90% credit as justified in Section M.9.1.7.2. Similarly, Section M.9.1.7.1 provides the justification for the use of 90% credit for borated aluminum. In addition to the fixed neutron poison in the basket, PRAs may be required for the center four, eight or sixteen assemblies depending on fuel assembly design and initial enrichment. The minimum required B_4C content of the PRAs is 40% Theoretical Density (TD) with 75% credit taken in the criticality analysis or 30% TD. *The minimum required Ag content of the AIC PRAs includes a* **[**

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Based on Basket Type A1 and A2, two more basket types are added and designated as Type A1-32 and A2-32. These two new basket types are identical to Type A1 and A2 except that they have 32 poison plates. Based on Basket Type B, four more basket types are added and designated as Type B1, B2, B1-r, and B2-r. Type B1 and B2 are identical to Type B except that they have higher minimum Boron-10 content of 0.015 gm/cm^2 and 0.020 gm/cm^2. Type B1-r and B2-r are identical to Type B1 and B2 except that they contain 4 AIC PRAs instead of 4 B_4C PRAs. Based on Basket Type C, three more basket types are added and designated as Type C1, C1-r, and C2-r. Type C1 is identical to Type C except that it has higher minimum B-10 content of 0.015 gm/cm^2. Type C1-r is identical to Type C1 except that it contains 8 AIC PRAs instead of 8 B_4C PRAs. Type C2-r is identical to Type C1-r except that it has higher minimum B-10 content of 0.020 gm/cm^2.

Basket designations as a function of number of PRAs and required B-10 loading for metallic plates are as *specified in Table M.2-4b.*

M.6.1 Discussion and Results

Figure M.6-1 shows the cross section of the NUHOMS®-32PT DSC. The NUHOMS®-32PT DSC stainless steel basket consists of a welded plate or tube design. The welded plates or tubes form 32 compartments with sufficient space to accommodate aluminum or poison/aluminum inserts and a PWR fuel assembly. The fuel compartment structure is connected to perimeter transition rail assemblies as shown on the drawings in Section M.1.5. The poison/aluminum plates and aluminum plates are located inside the fuel compartments. The poison plates may be arranged in any of the following configurations: a 20 poison plate configuration (base configuration), as shown in Figure M.6-1; an alternate 16 poison plate configuration, as shown in Figure M.6-13; and another alternate 24 poison plate configuration, as shown in Figure M.6-14. Figure M.6-2 through Figure M.6-4 show the fuel compartments that must contain PRAs/*RCCAs* for loading configurations that require four, eight or sixteen PRAs. *Figure M.6-2 and Figure M.6-3 are also applied to the AIC PRAs.* The 20 poison plate basket configuration shown in Figure M.6-1 is analyzed as a Type A/B/C/D basket while the 24 poison plate basket configurations shown in Figure M.6-14 is analyzed as an Alternate Type A/B/C/D basket at a *B*-10 loading of 0.0070 g/cm^2. The 24 poison plate basket configuration is also analyzed as a Type A1 and Type A2 basket at *B*-10 loadings of 0.0150 g/cm^2 and 0.0200 g/cm^2 respectively. The 16 poison plate basket configuration shown in Figure M.6-13 is also analyzed as an Alternate Type A basket. *The 32 poison plate basket configuration is analyzed as a Type A1-32 and Type A2-32 basket at B-10 loadings of 0.0150 g/cm^2 and 0.020 g/cm^2, respectively.*

The analysis presented herein is performed for a NUHOMS®-32PT DSC in the NUHOMS® OS197/197H Transfer Casks (TCs) during normal and accident loading conditions. The NUHOMS® OS197/197H TCs consists of an inner stainless steel shell, lead gamma shield, a stainless steel structural shell and a hydrogenous (liquid) neutron shield. This analysis is applicable to any licensed cask of similar construction. The NUHOMS®-32PT DSC/TC configuration is shown to be sub-critical under normal and accident conditions of loading, transfer and storage.

The criticality analysis determines the most reactive configuration for the basket and assembly location. Then criticality calculations evaluate a variety of fuel assembly types, initial enrichments and PRA configurations. Finally, the maximum allowed initial enrichment for each assembly type/PRA configuration is determined. The maximum allowed initial enrichment for each assembly type/PRA configuration is listed in Table M.6-1. The calculations determine k_{eff} with the CSAS25 and CSAS5 control modules of SCALE-4.4 and SCALE 6.0, respectively, [6-1 and 6-4] for each assembly type/PRA configuration and initial enrichment, including all uncertainties to assure criticality safety under all credible conditions.

Note that the results of the 20-poison plate basket that specify the minimum allowable fuel assembly are not included in Table M.6-1. This implies that the 20-poison plate basket design which is exclusively modeled in most of the criticality evaluations in this chapter is not authorized to store PWR fuel assemblies. However, this basket design is employed in all the criticality sensitivity calculations and is referenced throughout the remainder of this chapter. The results of these sensitivity calculations are still applicable to the 32PT DSC.

The results of the evaluation presented include reconstituted fuel assemblies where fuel pins are replaced with up to 56 solid stainless steel rods or an unlimited number of lower enriched UO_2 rods of the same diameter as the fuel pins.

M.6.2 Package Fuel Loading

The NUHOMS®-32PT DSC is capable of transferring and storing *a maximum of 32* PWR fuel assemblies. *In addition, damaged fuels or failed fuels that remain intact (for a total of 32 PWR fuel assemblies) can also be stored within the NUHOMS®-32PT DSC,* Table M.6-2 lists the fuel assemblies considered as authorized contents of the NUHOMS®-32PT DSC.

Table M.6-3 lists the fuel parameters for the PWR fuel assemblies. Reload fuel from other manufacture's with the same parameters are also considered as authorized contents.

For the *intact* WE 17x17, BW 15x15, WE 15x15, CE 14x14, *CE 15x15,* and WE 14x14 class assemblies, CCs are also included as authorized contents. The only change to the package fuel loading to evaluate the addition of these CCs is replacing the borated water in the water holes with $^{11}B_4C$. Since these CCs displace borated moderator in the assembly guide and or instrument tubes, an evaluation is performed to determine the potential impact of storage of CCs that extend into the active fuel region on the system reactivity. For CCs such as CRAs and BPRAs no credit is taken for the cladding and absorbers; rather the CCs are modeled as $^{11}B_4C$ in the entire tube of the respective design. Thus, the highly borated moderator in the tube is modeled as $^{11}B_4C$. The inclusion of more Boron-11 and carbon enhances neutron scattering causing the neutron population in the fuel assembly to be slightly increased which increases reactivity. Therefore, these calculations bound any CC design that is compatible with WE 17x17, BW 15x15, WE 15x15, CE 14x14 and WE 14x14 class assemblies. CCs that do not extend into the active fuel region of the assembly do not have any effect on the reactivity of the system as evaluated because only the active fuel region is modeled in this evaluation with periodic boundary conditions making the model infinite in the axial direction. The fuel assembly dimensions reported in Table M.6-3 remain unchanged for the CC cases. The models that include CCs only differ in that the region inside the guide tubes and instrument tube are modeled as $^{11}B_4C$ instead of moderator of PRAs. Additionally, the presences of non-multiplying sources like the NSAs have no impact on criticality calculations.

Since the criticality analysis models simulate only the active fuel height, any CC that is inserted into the fuel assembly such that it does not extend into the active fuel region is considered as authorized for storage without additional calculations as required for control components that extend into the active fuel region. For example, TPAs or ORAs are permitted for storage from a criticality standpoint, without any further calculations that model these CCs, since TPAs or ORAs do not extend into the active fuel region.

Table M.6-4 lists the minimum B_4C contents for PRAs for the various assembly classes. The linear B_4C content per PRA rod used in the KENO V.a model is calculated by multiplying the B_4C density modeled by the cross-sectional area of the poison rod as modeled. Taking the modeled linear B_4C content and dividing by 0.75, to account for the fact that we only take credit for 75% of the B-10 in the analysis, calculates the minimum linear B_4C content per PRA rod specified in Table M.6-4. For example, the modeled B_4C density is 0.756 g/cm^3. For the B&W 15 x15 PRA, the poison in the PRA is modeled with a radius of 0.55 cm. The cross sectional area of the poison is therefore $\pi(0.55)^2$ or 0.950 cm^2. Therefore, the modeled linear B_4C density is 0.72 g/cm and the minimum specified B_4C density is 0.96 g/cm.

The AIC PRA absorber material is also allowed for the intact fuels, which is a very high thermal neutron absorber containing silver-indium-cadmium (AIC) alloy. No credit is taken for residual indium or cadmium isotopes in AIC PRAs. Table M.2-4a lists the minimum silver content employed for AIC PRAs for the WE 17x17 fuel class.

The CE 14x14, CE 15x15, WE 14x14 and WE 17x17 damaged fuel classes are qualified for transferring and storage in the NUHOMS®-32PT DSC. Several different configurations with varying number of damaged fuel assemblies (balanced by intact fuel assemblies) are allowed for loading in the NUHOMS®-32PT DSC. The CE 14x14, CE 15x15, WE 14x14 and WE 17x17 failed fuels are qualified for transferring and storage in the NUHOMS®-32PT DSC. Failed fuels are placed in failed fuel cans (FFCs) closed at top and bottom ends with an end cap. A maximum of eight failed fuel assemblies (balanced by intact fuel assemblies) are allowed for loading in the NUHOMS®-32PT DSC. The CC and PRA are not allowed for the fuel loading configurations containing either damaged fuels or the failed fuels. The results from the WE 17x17 fuel class for damaged and failed fuel configurations are applicable to the CE 15x15 fuel class.

M.6.3 Model Specification

The following subsections describe the physical models and materials of the NUHOMS®-32PT DSC as loaded and transferred in the NUHOMS® OS197 or OS197H TC used for input to the CSAS25 module of SCALE-4.4 [6-1] *and CSAS5 module of SCALE6 [6.4]* to perform the criticality evaluation. The reactivity of canister under storage conditions is bounded by the TC analysis with zero internal moderator density case. The TC analysis with zero internal moderator density case bounds the storage conditions in the HSM because (1) the canister internals are always dry (purged and backfilled with He) while in the HSM, and (2) the TC contains materials such as steel and lead which provide close reflection of fast neutrons back into the fueled basket while the HSM materials (concrete) are much further from the sides of the DSC and thereby tend to reflect thermalized neutrons back to the canister which are absorbed in the canister materials reducing the system reactivity.

M.6.3.1 Description of Calculational Model

The TC and canister are explicitly modeled using the appropriate geometry options in KENO V.a of the CSAS25 module in SCALE-4.4. Several models are developed to evaluate the fabrication tolerances of the canister, basket, fuel clad outer diameter, fuel assembly locations, fuel assembly type, initial enrichments, PRA locations, and storage of CCs with the *intact* B&W 15x15, WE 17x17, *CE 15x15,* WE 15x15, CE 14x14 and WE 14x14 assembly classes.

The first model is a full active-fuel height and full radial cross section of the canister and TC with reflective boundary conditions on the ends and sides. The model does not explicitly include the water neutron shield. However, the infinite array of TCs without the neutron shield does contain unborated water between the TCs. KENO plots of these models for each assembly class are included in Section M.6.6.2. This model is used to determine the most reactive fuel assembly for a given enrichment and without any PRAs, most reactive assembly-to-assembly pitch, and to determine the most reactive canister configuration accounting for manufacturing tolerances and fuel assembly clad outer diameter tolerances.

All calculations to determine the most reactive configuration are performed utilizing the configurations containing 20 poison plates. There is no change to the most reactive configuration due to a change in the number and orientation of the poison plates in the basket (16 poison plate, 24 poison plate, *and 32 poison plate* configurations).

The second model is of the most reactive configuration identified above. This model is used to determine the maximum enrichment allowed for each assembly type as a function of the number of PRAs (none, four, eight and sixteen), *the number of AIC PRAs,* and boron loading, as appropriate. In addition, the effect of CCs for the B&W 15x15, WE 17x17, WE 15x15, CE 14x14 and WE 14x14 class assemblies for the various configurations are evaluated.

Additionally, the maximum allowable enrichment is determined for the CE 15x15 and WE 17x17 fuel classes with A1-32 and A2-32 basket types, which is a 32 poison plates configuration.

The design basis intact assembly KENO models are utilized as starting KENO models for the damaged assembly calculations. These KENO models are modified to evaluate the various damaged fuel configurations like single shear, double shear, missing fuel rods, optimum pitch, and axial fuel shifting. These models are then analyzed to determine the most reactive damaged fuel configuration, which is used to determine the maximum allowable initial enrichment for each assembly type as a function of basket type and soluble boron concentration.

For CE 14x14 damaged fuel class, configurations with 16, 28, and up to 32 damaged fuel assemblies are evaluated; the balance fuel assemblies in configurations with 16 and 28 damaged fuels are intact.

For failed fuel, CE 14x14, WE 14x14, and WE 17x17 are analyzed with the NUHOMS®-32PT DSC loaded with 24 intact fuel assemblies and 08 failed fuel pin arrays enclosed in FFCs and end caps, which are placed in corner locations of the DSC as shown in Figure M.6-9. These failed fuel pin arrays comprise of de-cladded fuel pins arranged in arrays of varying sizes. These configurations are analyzed at different soluble boron concentrations and for various basket types (A1, A2, for CE 14x14 and WE14x14 fuels and A1, A2, A1-32 and A2-32 for WE 17x17 fuel).

Figure M.6-5 is a sketch of each KENO V.a unit showing all materials and dimensions for each Unit and an annotated cross section map showing the assembled geometry units in the radial direction of the most reactive configuration identified in this evaluation. The bounding k_{eff} is calculated with a Westinghouse 17x17 LOPAR/Standard assembly with an initial enrichment of 3.4 wt. % U-235, with no PRAs and 32 BPRAs.

Note that BPRAs are the most relevant CCs for criticality considerations and are utilized in the rest of this chapter to cover all CCs.

M.6.3.2 Package Regional Densities

The Oak Ridge National Laboratory (ORNL) SCALE code package [6-1] contains a standard material data library for common elements, compounds, and mixtures. All the materials used for the TC and canister analysis are available in this data library.

Table M.6-5 provides a complete list of all the relevant materials used for the criticality evaluation. The material density for the B-10 in the poison plates includes a 10% reduction, a 25% reduction for the PRAs, *and a* [

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M.6.4 Criticality Calculations

This section describes the analysis performed for the criticality analysis. The analyses are performed with the CSAS25 module of the SCALE system. A series of calculations are performed to determine the relative reactivity of the various fuel assembly designs evaluated and to determine the most reactive configuration without PRAs and BPRAs. The most reactive fuel for a given enrichment, as demonstrated by the analyses, is the B&W 15x15 Mark B assembly. The most reactive credible configuration is an infinite array of flooded TCs with minimum fuel compartment inner diameter, minimum basket structure thickness and minimum assembly-to-assembly pitch.

As mentioned in Section M.6.1, the NUHOMS®-32PT DSC (including the 16 poison plate, 24 poison plate, *and 32 poison plate* alternate configurations) is evaluated to determine the maximum initial enrichment authorized for each assembly class without PRAs, and with four, eight and 16 PRAs, as applicable.

M.6.4.1 Calculational Method

M.6.4.1.1 Computer Codes.

The CSAS25 control module of SCALE-4.4 [6-1] is used to calculate the effective multiplication factor (k_{eff}) of the fuel in the TC. The CSAS25 control module allows simplified data input to the functional modules BONAMI-S, NITAWL-S, and KENO V.a. These modules process the required cross sections and calculate the k_{eff} of the system. BONAMI-S performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL-S applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below 0.0015 for all calculations.

The CSAS5 control module of SCALE 6.0 [6-4] is used in the determination of the maximum allowable enrichment for the WE 17x17 assembly class, *Type A1 and A2 baskets (2800 ppm soluble boron), Type A1-32 and A2-32 baskets, configurations with B_4C PRAs or AIC PRAs, and for the CE 14x14, WE 14x14 and WE17x17 assembly classed in damaged and failed fuel configurations.* The CSAS5 control module allows simplified data input to the functional modules BONAMI, NITAWL, and KENO V.a. These modules process the required cross sections and calculate the k_{eff} of the system. BONAMI-S performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below 0.0010 for all calculations.

M.6.4.1.2 Physical and Nuclear Data.

The physical and nuclear data required for the criticality analysis include the fuel assembly data and cross-section data as described below.

Table M.6-3 provides the pertinent data for criticality analysis for each fuel assembly evaluated in the NUHOMS®-32PT DSC.

The criticality analysis used the 44-group cross-section library built into the SCALE system. ORNL used ENDF/B-V data to develop this broad-group library specifically for criticality analysis of a wide variety of thermal systems.

M.6.4.1.3 Bases and Assumptions.

The analytical results reported in Section M.3.7 demonstrate that the TC containment boundary and canister basket structure do not experience any significant distortion under hypothetical accident conditions. Therefore, for both normal and hypothetical accident conditions the TC geometry is identical except for the neutron shield and skin. As discussed above, the neutron shield and skin are conservatively removed and the interstitial space modeled as water.

The TC is modeled with KENO V.a using the available geometry input. This option allows a model to be constructed that uses regular geometric shapes to define the material boundaries. The following conservative assumptions are also incorporated into the criticality calculations:

1. No burnable poisons are accounted for in the fuel,

2. Water density is at optimum moderator density,

3. Unirradiated fuel – no credit taken for fissile depletion due to burnup or fission product poisoning,

4. The maximum lattice average fuel enrichment is modeled as uniform everywhere throughout the assembly. Natural uranium blankets and axial or radial enrichment zones are modeled as enriched uranium. It is assumed that the fuel assemblies are of uniform enrichment everywhere.

5. All fuel rods are filled with 100% moderator in the pellet/cladding gap,

6. Only the active fuel length of each assembly type is explicitly modeled with reflective boundary conditions on the ends; therefore the model is effectively infinitely long,

7. The neutron shield and stainless steel skin of the TC are stripped away and the infinite array of TCs are pushed close together with moderator in the interstitial spaces,

8. The least material condition (LMC) is assumed for the fuel compartment and fuel structure assemblies are pushed together toward the center maximizing reactivity; the reduction in steel thickness also reduces neutron absorption in the steel in the basket,

9. The transition rails between the basket and the canister shell are modeled as 100% aluminum. Steel and open space in the transition rails reduces reactivity because these materials have much higher absorption cross sections as compared to the aluminum,

10. Only 90% credit is taken for the B-10 in the poison plates and 75% credit for the B-10 in PRAs are credited in the KENO models,

11. Temperature is 20°C (293K),

12. Ninety six percent theoretical density for fuel which conservatively increases the total fuel content in the model, and

13. All stainless steel, including XM-19, is modeled as SS304; the small differences in the composition of the various stainless steels have no effect on results of the calculation.

14. The most reactive geometry remains unaffected for the 16 poison plate*, 24 and 32* poison plate configurations.

15. **[**

]

M.6.4.1.4 Determination of k_{eff}

The Monte Carlo calculations performed with CSAS25 *and CSAS5* (KENO V.a) use a flat neutron starting distribution. The total number of histories traced for each calculation is approximately 500,000. This number of histories is sufficient to converge the source and produce standard deviations of less than 0.15% in k_{eff}. The maximum k_{eff} for the calculation is determined with the following formula:

$$k_{eff} = k_{KENO} + 2\sigma_{KENO.}$$

M.6.4.2 Fuel Loading Optimization

Determination of the Most Reactive Fuel Type

All fuel lattices listed in Table M.6-3 are evaluated to determine the most reactive fuel assembly type with initial enrichments of 3.3 wt. % U-235. The fuel types are analyzed with water in the fuel pellet cladding annulus and are centered in the fuel compartments of the 20 poison plate configuration. Nominal basket dimensions are used in the KENO models. The effect of moderator density is also evaluated using the most reactive fuel type, B&W 15x15 Mark B.

- The canister/TC model for this evaluation differs from the actual design in the following ways:
- the *B*-10 content in the poison plates is 10% lower than the minimum required,
- the neutron shield and the skin of the TC are conservatively replaced with water between the TCs, and
- the stainless steel and aluminum transition rails provide support to the fuel compartment grid are modeled as solid aluminum.

In all other respects, the model is the same as that described in Sections M.6.3.1 and M.6.1.1. KENO plots of these models are included in Section M.6.6.2.

Two typical input files are included in Section M.6.6.3, one without PRAs and one with four (4) PRAs. The results of these calculations are listed in Table M.6-6. The most reactive fuel type evaluated for the canister design for a given initial enrichment, without PRAs and without BPRAs, is the B&W 15x15 Mark B assembly.

Determination of the Most Reactive Intact Configuration

The fuel-loading configuration of the canister/TC affects the reactivity of the package. Several series of analyses determine the most reactive configuration for the canister/TC.

For this analysis, the most reactive fuel type is used to determine the most reactive 20 poison plate configuration. The canister/TC is modeled over the active fuel height of the B&W 15x15 Mark B assembly with reflective boundary conditions on all sides. This represents and infinite array in the x-y direction of canisters/TCs that are conservatively infinite in length. The first model in this series of calculations is identical to the model used above. The canister/TC model for this evaluation differs from the actual design in the following ways:

The fifth set of analyses evaluates the effect of fuel compartment size on the system reactivity. The model starts with the minimum basket structure plate/tube thickness modeled as above because it represents the most reactive condition. For this evaluation the fuel compartment width is varied from 8.715 to 8.925 inches square. Note that these dimensions are the basket structure plate to plate distance or tube inner diameter. The actual space remaining for the fuel assembly is obtained by subtracting the poison/aluminum plate thickness. The results of the evaluation are given in Table M.6-11. The results show that the effect is within the statistical uncertainty of calculation. The most reactive configuration is with the minimum fuel compartment size because the assembly-to-assembly pitch is minimized. The balance of this evaluation use the minimum fuel compartment width because it represents the most reactive configuration.

The final series of sensitivity analyses determined the most reactive fuel assembly-to-assembly pitch. The most reactive configurations occur at the minimum assembly-to-assembly pitch. For assemblies with large cross sections such as the B&W 15x15 Mark B, however, there is little space for the assemblies to move with,in the minimum fuel compartment modeled. The results in Table M.6-12 demonstrate that for the B&W 15x15 Mark B assembly the effect is within the statistical uncertainty of the calculation. For smaller assemblies such as the WE 14x14, this effect would be statistically significant because the assemblies have more space in which to move inside the fuel compartment, therefore the minimum assembly-to-assembly pitch is used for the balance of this analysis.

Determination of the Most Reactive Damaged Configuration

The bent or bowed fuel rod cases assume that the fuel is intact but not in its nominal fuel rod pitch. It is possible that the fuel rods may be crushed inwards or bowed outwards to a certain degree. Therefore, this will be evaluated by varying the fuel rod pitch from a minimum pitch (based on clad OD) to a maximum based on the fuel compartment size for each fuel assembly class. All pitch variations assume a uniform rod pitch throughout the entire fuel matrix.

The single-ended fuel rod shear cases assume that a fuel rod shears in one place and is displaced to a new location. The fuel pellets are assumed to remain in the fuel rod.

The double shear cases assume the fuel rod shears in two places, and a part of the fuel is separated. If the fuel rod is displaced in the same plane, it is equivalent to the single shear described earlier. If displaced in a different plane, parts of the assembly will have either a fuel rod missing, or an additional fuel rod. To simulate this, one row of fuel rods will be removed from one section of the assembly and added to the other, so that the total amount of fuel in the DSC remains unchanged.

The poison shift analyzes the possibility of loose fuel rods shifting beyond the height of the poison plates. The maximum difference between the height of the basket and the cavity length of the DSC is 1.0 inch.

In the missing fuel rod analysis, a certain number of fuel rods are progressively removed as a way of changing the fuel-to-moderator ratio in the assembly, and its effect on the reactivity of the system. This accounts for the possibility of a reactivity increase due to softening of the neutron spectrum from the increased moderator.

For WE17, WE14 and CE14, above five damaged fuel configurations are analyzed to find the most reactive one which is used for the maximum initial enrichment determination. The methodology used in analysis of damaged fuel is similar to that employed for the NUHOMS®-32PTH1 DSC. The details of damaged fuel analysis methodology are given in Table M.6-72.

Determination of the Most Reactive Failed Configuration

The NUHOMS®-32PT DSC is designed to accommodate up to a maximum of eight failed fuel assemblies encapsulated in FFCs. Failed fuel rods are fuel rods that have been removed from another damaged or failed fuel assembly and placed in a secondary container. These include breached rods, grossly breached rods, other defective rods and fuel debris.

The FFCs are modeled as stainless steel containers containing failed fuel array. For conservatively simulating breached rods, the fuel rods are modeled without cladding, but the layer of full density fresh water in the fuel-cladding gap is retained. Rod pitch studies are performed on these failed fuel rods similar to that performed for damaged fuel to determine optimum fuel rod pitch.

Additional sensitivity analyses performed for 32 failed WE 17x17 FAs with full rod lattice (17x17 failed rods) and reduced lattice (16x16 failed rods) show the maximum reactivity occurs for the reduced lattice (16x16) similar to the nominal lattice (17x17) when empty rod locations are replaced with fuel rods, as shown in Table M.6-74.

For WE17, WE14, and CE14, the maximum allowable enrichment is determined for each fuel type as a function of soluble boron concentration and basket type.

M.6.4.3 Studies with Radial Variation of Enrichment

The next set of analyses is for the Exxon/ANF 15x15 CE fuel assembly. These calculations justify the adequacy of the uniform maximum lattice average fuel enrichment assumed in the analysis. These calculations were carried out using a soluble boron concentration of 2500 ppm and a poison loading of 6.30 mg B-10/cm^2 with moderator densities (MD) varying from 40% to 100% of full density. The 16 poison plate configuration was utilized for these analyses and, the results are applicable to the 20 and 24 poison plate configuration and all basket types.

The different variable enrichments used in these calculations are shown in Table M.6-25. A plot of the fuel assembly with the radial variation in enrichment is shown in Figure M.6-17 and Figure M.6-18. The results for this evaluation are listed in Table M.6-26. Due to the higher initial enrichment values utilized to model the loading patterns 2A, 2B and 2C, the k_{eff} values shown in Table M.6-26 are higher than USL. The purpose of these calculations is to determine the sensitivity of k_{eff} due to radial variation of enrichment only and not to USL; hence, the results shown in Table M.6-26 are acceptable for these sensitivities studies.

The results demonstrate that the assumption of the uniform maximum lattice average enrichment is both appropriate and conservative. The k_{eff} of the system with the uniform maximum lattice average enrichment assumption is greater than that of the non-uniform case at all MD.

M.6.4.4 Determination of the Maximum Initial Enrichment for each Fuel Class

Determination of the Maximum Initial Enrichment for Intact Fuels

The most reactive configuration as determined in Section M.6.3.1 above, is with solid aluminum in the transition rail region, nominal poison/aluminum plate thickness, minimum basket structure plate/tube thickness, minimum fuel compartment width, minimum assembly-to-assembly pitch and uniform maximum planar enrichment. The following analysis uses this configuration to determine the maximum allowed initial enrichment as a function of *basket type* and *soluble boron loading* for each assembly class. All *four* poison plate configurations (20 poison plate configuration, 16 poison plate configuration, 24 poison plate configuration, *and 32 poison plate configuration*) are evaluated in these calculations. Calculations at 2500 ppm soluble boron are performed for all assembly classes. For the CE 14x14, WE 14x14, and CE 15x15 assembly classes, calculations are also performed for a range of boron soluble loadings (1800-2500 ppm) for the 16 poison plate and the 24 poison plate configurations. *Calculations are performed for the CE 15x15 and WE 17x17 loaded in the 32 poison plate basket design.* The most reactive assembly type for each assembly class is used for each evaluation. In addition, for each case the internal moderator density is varied to determine the peak reactivity for the specific configuration. The maximum initial enrichment for each assembly class and PRA configuration are provided in Table M.6-1.

The canister/TC model for this evaluation differs from the actual design in the following ways:

- the *B*-10 content in the poison plates is 10% lower than the minimum required,
- the *B*-10 content in the PRAs is 25% lower than the minimum required,
- *the Ag content in the AIC-PRA is 25% lower than minimum required,*
- the stainless steel/aluminum transition rails that provide support to the fuel compartment grid are modeled as various solid materials to determine the most reactive condition,
- BPRAs, when modeled, are modeled as solid $^{11}B_4C$ in the guide tubes and instrument tubes,
- the neutron shield and the skin of the TC are conservatively replaced with water between the TCs, and
- the worst-case material conditions, as determined in Section M.6.4.2 above, are modeled.

The input file for the case with the highest calculated reactivity is included in Section M.6.6.4.

WE 17x17 Class Assemblies

The most reactive WE 17x17 class assembly is the WE 17x17 LOPAR/standard assembly as demonstrated in Table M.6-6. The results for the WE 17x17 class assembly calculations for the 20 poison plate configuration are listed in Table M.6-13 and Table M.6-14 for cases without and with BPRAs, respectively. The results for the WE 17x17 class assembly calculations for the 16 poison plate configuration are listed in Table M.6-27 and Table M.6-28 for cases without and with BPRAs, respectively. The results for the WE 17x17 class assembly calculations for the 24 poison plate Type A/B/C/D basket configuration are listed in Table M.6-29 and Table M.6-30 for cases without and with BPRAs, respectively. The 24 poison plate Type A1 and A2 basket configuration results are listed in Tables M.6-51 and M.6-57, respectively.

Additional configurations with the 24 poison plate design include B_4C PRAs (04 and 08 in number) and AIC PRAs (04 and 08 in number). The AIC PRA configurations are evaluated at 2800 ppm soluble boron concentration, and the maximum allowable enrichments are determined. The results for the AIC PRA configurations are given in Table M.6-63 (Parts 4 and 5).

For the AIC PRA configurations, the poison rod radius is modeled as 0.4902 cm (0.386" diameter) in the KENO base model. Considering the initial AIC composition is 80 wt. % Ag, 15 wt. % In and 5 wt. % Cd with a density of 10.17 g/cm^3, the initial density of Ag is 8.14 g/cm^3 (6.14 g/cm) per rod of AIC. [ ] *Table M.6-1 (Part 3 of 3) shows the maximum initial enrichments for each configuration with AIC PRA.* [ ]

A sensitivity evaluation is performed to demonstrate the applicability of the linear density requirement, [ ] *for a smaller AIC diameter. The sensitivity analysis utilizes the following model for Type B1-r case with 4 AIC PRAs presented in Table M.6-1 (Part 3 of 3):*

- *WE 17x17 fuel assembly without CC, 4.60 wt%, in presence of 2800 ppm boron. The maximum k_{eff} is 0.9370 (80% water density – 2800 ppm boron), Table M.6-63 (Part 4 of 5).*

The sensitivity analysis employs a smaller poison radius of 0.43307 cm (0.341-in. diameter or 0.86614 cm diameter). See AIC diameter Control Rod specification for a WE 17x17 fuel assembly in Table 4 of NUREG/CR-6759. In order to obtain a statistically equivalent k_{eff} to 0.9370, a linear Ag density of [ ] *is required.*

Based on the results of the base and sensitivity analyses, a minimum Ag linear density of 2.60 g/cm is specified for AIC PRAs, which corresponds to a [ ] *For a nominal absorber diameter of 0.341 inches, as indicated in Figure M.1-2 and Table M.2-4a,* [ ]

Criticality analysis for the 32 poison plate configuration is performed with and without CCs. This configuration is evaluated at 2500 ppm and 2800 ppm soluble boron concentrations and the maximum allowable enrichments are determined. The results for this configuration are given in Table M.6-63 (Parts 2 and 3 of 5).

B&W 15x15 Class Assemblies

The most reactive B&W 15x15 class assembly is the B&W 15x15 Mark B assembly as demonstrated in Table M.6-6. The results for the B&W 15x15 class assembly calculations for the 20 poison plate configuration are listed in Table M.6-15 and Table M.6-16 for cases without and with BPRAs, respectively. The results for the B&W 15x15 class assembly calculations for the 16 poison plate configuration are listed in Table M.6-31 and Table M.6-32 for cases without and with BPRAs, respectively. The results for the B&W 15x15 class assembly calculations for the 24 poison plate Type A/B/C/D basket configuration are listed in Table M.6-33 and Table M.6-34 for cases without and with BPRAs, respectively. The 24 poison plate Type A1 and A2 basket configuration results are listed in Tables M.6-49 and M.6-55, respectively.

CE 15x15 Class Assemblies

The most reactive CE 15x15 class assembly is the CE 15x15 Palisades assembly as demonstrated in Table M.6-6. The results for the CE 15x15 class assembly calculations for the 20 poison plate configuration are listed in Table M.6-17 for cases without BPRAs. BPRAs are not authorized to be stored with CE 15x15 class assemblies.

The addition of plugging cluster assemblies, i.e., steel rods, into each of the eight guide tubes of a CE 15x15 class assembly reduces the maximum reactivity of the payload. The introduction of the steel rods displaces both moderator and soluble boron within the assemblies. The plugging clusters are assumed to extend approximately 1 inch into the top of the assembly's active fuel region, and the resulting change in the maximum reactivity is less than the statistical uncertainties of the calculations. To demonstrate the *effect* of displacing the borated water on system reactivity, CE 15x15 Palisades cases with the highest fuel enrichments and highest soluble boron loadings (2300, 2400, and 2500 ppm boron) were reevaluated with steel in the guide tubes. Two scenarios were evaluated: full length steel rods and 1 inch long steel rods. The calculated reactivity of the models are shown in Table M.6-44 and Table M.6-45 (the k_{KENO} and 1σ values in columns 2 and 3 are from Table M.6-17). As shown therein, the addition of plugging clusters reduces reactivity of the system regardless of the length of the plugging cluster. These results also apply to the 16 and 24 poison plate configurations.

The results for the CE 15x15 class assembly calculations for the 16 and 24 poison plate configuration as well as the 24 position plate and Type A basket configuration as a function of boron loading are listed in Table M.6-41. The results for the 24 poison plate and Type A1 and A2 basket configuration are listed in Tables M.6-48 and M.6-54, respectively.

Additional configurations with the 24 poison plate design (at 2800 ppm) and the 32 poison plate design are analyzed. The maximum allowable enrichments are determined for a soluble boron concentration at 2500 ppm and 2800 ppm. The result for criticality analysis of these configurations is given in Table M.6-64.

WE 15x15 Class Assemblies

The most reactive WE 15x15 class assembly is the WE 15x15 Standard assembly as demonstrated in Table M.6-6. The results for the WE 15x15 class assembly calculations for the 20 poison plate configuration are listed in Table M.6-18 for cases without BPRAs. The results for the WE 15x15 class assembly calculations for the 16 poison plate configuration are listed in Table M.6-36 for cases without BPRAs. The results for the WE 15x15 class assembly calculations for the 24 poison plate Type A/B/C/D basket configuration are listed in Table M.6-37 for cases without BPRAs. The 24 poison plate Type A1 and A2 basket configuration results, with and without BPRAs, are listed in Tables M.6-50 and M.6-56, respectively.

CE 14x14 Class Assemblies

The most reactive CE 14x14 class assembly is the CE 14x14 Fort Calhoun assembly as demonstrated in Table M.6-6. The results for the CE 14x14 class assembly calculations for the 20 poison plate configuration as a function of boron loading are listed in Table M.6-19 for cases without BPRAs.

The results for the CE 14x14 class assembly calculations for the 16 poison plate configuration as well as the 24 position plate and Type A/B/C/D basket configuration as a function of boron loading are listed in Table M.6-42. The results for the 24 poison plate and Type A1 and A2 basket configuration results, with and without BPRAs, are listed in Tables M.6-47 and M.6-53, respectively.

WE 14x14 Class Assemblies

The most reactive WE 14x14 class assembly is the WE 14x14 ZCA/ZCB assembly as demonstrated in Table M.6-6. The results for the WE 14x14 class assembly calculations for the 20 poison plate configuration as a function of boron loading are listed in Table M.6-20 for cases without BPRAs.

The results for the WE 14x14 class assembly calculations for the 16 poison plate configuration as well as the 24 position plate and Type A/B/C/D basket configuration as a function of boron loading are listed in Table M.6-43. The results for the 24 poison plate and Type A1 and A2 basket configuration results, with and without BPRAs, are listed in Tables M.6-46 and M.6-52, respectively.

<u>Determination of the Maximum Initial Enrichment for Damaged fuels</u>

WE 17x17 Damaged Fuel Assemblies

WE 17x17 damaged fuel assemblies are allowed for loading in the NUHOMS®-32PT DSC with the 24 (A1 and A2 basket type) poison plates design, as well as the 32 poison plates design (A1-

32 and A2-32 basket types). The most reactive configuration (MRC) is ascertained by means of the criticality analysis of all damaged fuel scenarios previously discussed, and it was shown that rod pitch expansion yields the MRC. The MRC is then used to determine the maximum allowable enrichments for various basket types at different values of soluble boron concentration. The results for damaged fuel configurations analyzed with WE 17x17 fuel are given in Table M.6-65. The maximum allowable enrichments for the A1, A2, A1-32, and A2-32 basket types are shown in Table M.6-68.

The 28 damaged, 4 intact WE 17x17 FA loading configuration is examined in the 24-poison plate Type A1 DSC with 2500 ppm of soluble boron in the water for the sensitivity analysis with the CE 15x15 fuel. The maximum k_{eff} of 0.9386 occurs at a maximum enrichment of 4.05 wt. % with 70% water density in the model, as shown in Table M.6-68. The sensitivity analysis for the CE 15x15 fuel class is performed in the most reactive configuration, which is a maximum pitch within the 32PT compartment opening. The maximum k_{eff} of 0.9371 occurs at a maximum enrichment of 4.05 wt. % with 75% water density in the model.

The maximum planar average enrichments determined for the WE 17x17 fuel class using the 28 damaged, 4 intact loading configuration reported in Table M.6-61 are applicable to the CE 15x15 fuel class.

WE 14x14 Damaged Fuel Assemblies

For WE 14x14 assemblies, a similar procedure is followed to establish that the rod pitch study for a configuration with 32 damaged WE 14x14 assemblies is the most reactive out of all the damaged fuel scenarios discussed previously. The MRC is used to establish the maximum allowable enrichments for basket type A1, as well as A2, at different values of soluble boron concentration. The results for damaged fuel configurations analyzed with WE 14x14 fuel are given in Table M.6-66. The maximum allowable enrichments for the A1 and A2 basket types are given in Table M.6-69.

CE 14x14 Damaged Fuel Assemblies

For CE14 damaged fuel assemblies, the double shearing is the most reactive configuration, which is followed by the rod pitch expansion with a Δk_{eff} of 0.0010. Considering the small difference in k_{eff}, the rod pitch expansion is selected as the MRC for CE 14x14, same as for WE17x17 and WE14x14. The established MRC is used to determine the maximum allowable enrichment for basket type A1 and A2 with 16, 28 and 32 damaged fuels assemblies. The loading configurations for 16 and 28 damaged fuels are shown in Figure M.2-7 and M.2-8. The results for the damaged fuel configurations analyzed with CE 14x14 fuel are given in Table M.6-67. The results for loading curves are given in Table M.6-70.

<u>Determination of the Maximum Initial Enrichment for Failed fuels</u>

WE 17x17 Failed Fuel

As is previously described in Section M.6.4.2 for CE 14x14 and WE 14x14 classes, the most reactive configuration is determined using a 17x17 failed fuel pin array with the optimum pin pitch. Once the MRC is determined, the maximum allowable enrichment has been deduced for A1, A2, A1-32 and A2-32 basket types with a 14x14, 15x15, 16x16 and a 17x17 failed fuel array

size at 2500 and 2800 ppm. The results for the criticality analysis of WE 17x17 failed fuel are given in Table M.6-71.

The 8 failed, 24 intact WE 17x17 fuel (15x15 failed fuels array) loading configuration is examined in the 24-poison plate Type A1 DSC with 2500 ppm of soluble boron in the water for the sensitivity analysis with the CE 15x15 fuel. The maximum k_{eff} of 0.9345 occurs at a maximum enrichment of 4.10 wt. % as shown in Table M.6-71. The sensitivity analysis for the CE 15x15 fuel class is performed in the most reactive configuration, which is a maximum pitch within the 32PT compartment opening. The maximum k_{eff} of 0.9349 occurs at a maximum enrichment of 4.10 wt. %.

The maximum planar average enrichments determined for the WE 17x17 fuel class using the 8 failed, 24 intact loading configuration reported in Table M.6-62 are applicable to the CE 15x15 fuel class. Note that the 16x16 and 17x17 lattice results are not applicable to the CE 15x15 fuel class.

WE 14x14 Failed Fuel

The most reactive configuration is determined using a 14x14 failed fuel pin array with the optimum pin pitch as described in Section M.6.4.2. The MRC is subsequently used to determine the maximum allowable enrichment values for basket types A1 and A2 at 1800 ppm, 2100 ppm, 2300 ppm and 2500 ppm for WE 14x14 failed fuel loading curve. The results for the criticality analysis of WE 14x14 failed fuel in A1 and A2 baskets are given in Table M.6-72.

CE 14x14 Failed Fuel

The most reactive configuration is determined using the 14x14 failed fuel pin array with the optimum pin pitch as described in Section M.6.4.2. The MRC is subsequently used to determine the maximum allowable initial enrichment for the A1 and A2 basket types at 2600 ppm soluble boron concentration. The results for the criticality analysis of CE 14x14 failed fuel are given in Table M.6-73.

When intact, damaged, or failed fuel are loaded per Technical Specification Figure 1-4b, the maximum enrichments for the three fuel types are restricted with the lowest enrichment shown across Table M.6-1, Table M.6-61 and Table M.6-62 considering the same poison plate configuration and soluble boron loading.

M.6.4.5 Criticality Results

Intact Fuel by SCALE 4.4

Table M.6-21 lists the bounding results for all conditions of storage. The highest calculated k_{eff}, including 2σ uncertainty, is for the WE 14x14 assembly in the 20 poison plate configuration with an initial U-235 enrichment of 3.85 wt. %, no PRAs, 2000 ppm boron, and 60% moderator density. The maximum allowed initial enrichment for each assembly type/PRA configuration is listed in Table M.6-1.

These criticality calculations were performed with CSAS25 of SCALE-4.4. For each case, the result includes (1) the KENO-calculated k_{KENO}, (2) the one sigma uncertainty σ_{KENO}, and (3) the final k_{eff}, which is equal to $k_{KENO} + 2\sigma_{KENO}$.

The criterion for subcriticality is that

$$k_{KENO} + 2\sigma_{KENO} \leq USL,$$

where USL is the upper subcritical limit established by an analysis of benchmark criticality experiments. From Section M.6.5, the minimum USL over the parameter range is 0.9411. From Table M.6-21 for the most reactive case,

$$k_{KENO} + 2\sigma_{KENO} = 0.9388 + 2\ (0.0011) = 0.9410 \leq 0.9411.$$

For the criticality evaluations that were performed with CSAS5 and SCALE 6.0, the maximum k_{eff} is obtained for the WE 17x17 loaded in the Type A2-*32* basket at an enrichment level of *5.00* wt. % U-235, soluble boron concentration of 2800 ppm, no PRAs, and without CCs:

$$k_{eff} = k_{keno} + 2\sigma_{keno} = 0.9362 + 2*(0.0007) = 0.9380 \leq 0.9404$$

The criticality evaluation used the same cross section set, fuel materials and similar material/geometry options that were used in the 121 benchmark calculations as shown in Table M.6-22. The modeling techniques and the applicable parameters listed in Table M.6-24 for the actual criticality evaluations fall within or very close the range of those addressed by the benchmarks in Table M.6-22. *Note that six experimental cases in Table 2.1 of NUREG/CR-6361 [6-2] (designated as BW1810I, W3269A, W3269B1, W3269B2, W3269B3 and W3269C) include Ag-In-Cd rods.*

M.6.5.2 Results of the Benchmark Calculations

The results from the comparisons of physical parameters of each of the fuel assembly types to the applicable USL value are presented in Table M.6-24. The minimum value of the USL is determined to be 0.9411 based on comparisons to the limiting assembly parameters as shown in Table M.6-24.

M.6.5.3 Benchmarking of SCALE 6.0

For the criticality evaluations performed using CSAS5 of SCALE 6.0 [6-4], the 92 experimental problems used to perform the benchmarking along with pertinent parameters are listed in Table M.6-58.

The USL is dependent on the set of evaluated critical experiments where the models in the experiments must have features similar to the system evaluated. The experiments in general have similar features or parameters in common such that a trend of how these affect the final k_{eff} due to the limitations associated with modeling, calculation methodology, and nuclear cross-section data, can be evaluated. The features or parameters considered are U-235 enrichment, fuel pitch (cm), average energy group causing fission (AEG), soluble boron (ppm), assembly separation (cm), and moderator-to-fuel volume ratio. Using the relevant parameters, the correlation (r-value) of the parameters to the k_{eff} of the experiments must be evaluated to assess the level of influence of the parameter on the system reactivity. The USLSTATS code [6-7] provides the means to obtain the USL functions that can be used to obtain the final USL value if it can be shown that the parameters are closely correlated with k_{eff}, that is, $|r|$, is nearly 1.0. As demonstrated in Section M.6.5.3.2, there is no close correlation between the parameters and k_{eff}. In cases where no closely correlated parameters exist, the single-sided tolerance limit methodology described in NUREG/CR-6698 [6-8] is used, it can be shown that the k_{eff} values are normally distributed, which is demonstrated in Section M.6.5.3.2.

M.6.5.3.1 Benchmark Experiments and Applicability

The criticality benchmark experiments are obtained from the International Handbook of Evaluated Criticality Safety Benchmark Experiments [6-6]. A brief description of the experiments is provided as follows:

8 configurations (cases 1-8) from LEU-COMP-THERM-001:

These configurations are water moderated U(2.35 wt. % U-235)O_2 fuel rods in 2.032 cm square-pitched arrays. These experiments use open-top carbon steel tank configuration with acrylic support plate, polyethylene lattice plates and aluminum plates as support structure.

Table M.6-1
Maximum Initial Enrichment For Each Configuration, wt. % U-235
(Part 1 of 3)

Assembly Class	Soluble Boron Loading (ppm)	No PRAs (Type A) Poison Plate Configuration		4 PRAs (Type B) Poison Plate Configuration	8 PRAs (Type C) Poison Plate Configuration	16 PRAs (Type D) Poison Plate Configuration
		16	24	24	24	24
WE 17x17 Fuel Assembly(1)	2500	3.40	3.40	4.00	4.50	5.00
B&W 15x15 Mark B Fuel Assembly(1)	2500	3.30	3.30	3.90	NE	5.00
WE 15x15 Fuel Assembly (without CC)	2500	3.40	3.40	4.00	4.60	5.00
WE 15x15 Fuel Assembly (with CC)	2500	3.40	3.40	4.00	4.55	5.00
CE 14x14 Fuel Assembly (without CC)	1800	3.35	3.50	4.00	4.35	NE
	2000	3.50	3.70	4.20	4.55	NE
	2100	3.60	3.80	4.30	4.70	NE
	2200	3.70	3.90	4.40	4.80	NE
	2300	3.75	4.00	4.50	4.90	NE
	2400	3.80	4.05	4.60	5.00	NE
	2500	3.90	4.15	4.70	-	NE
CE 14x14 Fuel Assembly (with CC)	1800	3.30	3.45	3.90	4.25	NE
	2000	3.45	3.65	4.10	4.50	NE
	2100	3.55	3.75	4.20	4.60	NE
	2200	3.60	3.80	4.30	4.70	NE
	2300	3.65	3.90	4.40	4.80	5.00
	2400	3.80	4.00	4.50	4.90	5.00
	2500	3.90	4.05	4.60	5.00	5.00
WE 14x14 Fuel Assembly (with and without CC)	1800	3.55	3.75	4.40	NE	5.00
	2000	3.75	3.90	4.60	NE	NE
	2100	3.80	4.00	4.75	NE	NE
	2200	3.90	4.10	4.85	NE	NE
	2300	4.00	4.20	5.00	NE	NE
	2400	4.10	4.30	-	NE	NE
	2500	4.15	4.40	-	-	NE
CE 15x15 Fuel Assembly	1800	3.00	3.15	NE	NE	NE
	2000	3.15	3.30	NE	NE	NE
	2100	3.20	3.40	NE	NE	NE
	2200	3.30	3.50	NE	NE	NE
	2300	3.35	3.55	NE	NE	NE
	2400	3.40	3.60	NE	NE	NE
	2500	3.50	3.70	NE	NE	NE

NOTES: (1) With or without CCs. CCs shall not be stored in basket location where a PRA is required.
NE–Not Evaluated

Table M.6-1
Maximum Initial Enrichment For Each Configuration, wt. % U-235
(Part 2 of *3*)

Assembly Class and Type	Soluble Boron Loading (ppm)	0 PRAs (Type A1 and Type A2 24 Poison Plate Configuration)	
		Type A1	Type A2
WE 17x17 fuel assembly (without CC)	2500	4.05	4.20
	2800	*4.30*	4.50
WE 17x17 fuel assembly (with CC)	2500	4.00	4.15
	2800	*4.25*	4.45
B&W 15x15 Mark B fuel assembly (without CC)	2500	4.00	4.10
B&W 15x15 Mark B fuel assembly (with CC)	2500	3.90	4.10
WE 15x15 fuel assembly (without CC)	2500	4.10	4.20
WE 15x15 fuel assembly (with CC)	2500	4.10	4.20
CE 14x14 fuel assembly (without CC)	1800	3.95	4.10
	2100	4.30	4.45
	2300	4.50	4.70
	2500	4.70	4.90
CE 14x14 fuel assembly (with CC)	1800	3.80	3.95
	2100	4.10	4.25
	2300	4.30	4.50
	2500	4.50	4.70
WE 14x14 fuel assembly (without CC)	1800	4.20	4.20
	2100	4.55	4.60
	2300	4.80	5.00
	2500	5.00	5.00
WE 14x14 fuel assembly (with CC)	1800	4.20	4.35
	2100	4.60	4.75
	2300	4.80	5.00
	2500	5.00	5.00
CE 15x15 fuel assembly	1800	3.50	3.60
	2100	3.75	3.90
	2300	3.95	4.10
	2500	4.10	4.30
	2800	*4.45*	*4.55*

Table M.6-1
Maximum Initial Enrichment for Each Configuration, wt. % U-235
(Part 3 of 3)

Assembly Class and Type	***Soluble Boron Loading (ppm)***	***No PRA***	
		32 Poison Plate Configuration	
		Type A1-32	***Type A2-32***
WE 17x17 fuel assembly (without CC)	*2500*	*4.45*	*4.65*
	2800	*4.75*	*5.00*
WE 17x17 fuel assembly (with CC)	*2500*	*4.40*	*4.60*
	2800	*4.70*	*4.90*
CE 15x15 fuel assembly (CC not allowed)	*2500*	*4.55*	*4.75*
	2800	*4.85*	*5.00*
Assembly Class and Type	***Soluble Boron Loading (ppm)***	***4 B_4C PRA***	
		24 Poison Plate Configuration	
		Type B1	***Type B2***
WE 17x17 fuel assembly (without CC)	*2800*	*4.85*	*4.95*
Assembly Class and Type	***Soluble Boron Loading (ppm)***	***4 AIC PRA***	
		24 Poison Plate Configuration	
		Type B1-r	***Type B2-r***
WE 17x17 fuel assembly (without CC)	*2800*	*4.60*	*4.75*
Assembly Class and Type	***Soluble Boron Loading (ppm)***	***8 B_4C PRA***	
		24 Poison Plate Configuration	
		Type C1	
WE 17x17 fuel assembly (without CC)	*2800*	*5.00*	*NE*
Assembly Class and Type	***Soluble Boron Loading (ppm)***	***8 AIC PRA***	
		24 Poison Plate Configuration	
		Type C1-r	***Type C2-r***
WE 17x17 fuel assembly (without CC)	*2800*	*4.85*	*5.00*

Note: This table is for intact fuel. NE stands for Not Evaluated.

Table M.6-5
Material Property Data

Material	Density g/cm³	Element	Weight %	Atom Density (atoms/b-cm)
UO_2 (Enrichment - 3.4 wt%)	10.52	U-235	3.00	8.0797E-04
		U-238	85.15	2.2666E-02
		O	11.85	4.6948E-02
UO_2 (Enrichment - 5.0 wt%)	10.52	U-235	4.41	1.1882E-03
		U-238	83.73	2.2290E-02
		O	11.86	4.6956E-02
Zircaloy-4	6.56	Zr	98.23	4.2541E-02
		Sn	1.45	4.8254E-04
		Fe	0.21	1.4856E-04
		Cr	0.10	7.5978E-05
		Hf	0.01	2.2133E-06
Borated Water (2500 ppm Boron)	1.000	H	0.11	6.6769E-02
		O	0.89	3.3385E-02
		B-10	4.602E-04	2.7713E-05
		B-11	2.038E-03	1.1155E-04
Water	0.998	H	11.1	6.6769E-02
		O	88.9	3.3385E-02
Stainless Steel (SS304)	7.94	C	0.080	3.1877E-04
		Si	1.000	1.7025E-03
		P	0.045	6.9468E-05
		Cr	19.000	1.7473E-02
		Mn	2.000	1.7407E-03
		Fe	68.375	5.8545E-02
		Ni	9.500	7.7402E-03
Aluminum	2.70	Al	100.0	6.0307E-02
Lead	11.34	Pb	100.0	3.2969E-02
Aluminum - Boron Poison Plate (0.0063 g/cm² B-10 Type A/B/C/D)	2.465(1)	B-10	1.34	1.9847E-03
		Al	98.66	5.4276E-02
Aluminum-Boron Poison Plate (0.0135 g/cm² B-10 Type A1)	2.465(1)	B-10	2.88	4.2606E-03
		Al	98.60	5.3203E-02
Aluminum-Boron Poison Plate (0.0200 g/cm² B-10 Type A2)	2.465(1)	B-10	3.84	5.6809E-03
		Al	95.74	5.2618E-02
B_4C in PRA	0.756	B-10	14.42	6.5599E-03
		B-11	63.83	2.6405E-02
		C	21.75	8.2411E-03
$^{11}B_4C$ in BPRA	2.555	B-11	78.56	1.0988E-01
		C	21.44	2.7470E-02
AIC PRA(2)	*N/A*	*Ag-107*	*N/A*	[ ]
		Ag-109	*N/A*	[ ]

Note:

(1) Note in some models the aluminum–boron poison is modeled with a density of 2.693 g/cm³ (see the sample input in Section M.6.6.5), although the number density of B-10 is equivalent.

(2) Ag composition credited in KENO base model.

Table M.6-61
Maximum Initial Enrichment for Each Configuration, wt. % U-235 – Damaged Fuel

Assembly Class and Type	*Soluble Boron Loading (ppm)*	*28 Damaged Fuels*	
		24 Poison Plate Configuration	
		Type A1	*Type A2*
WE 17x17 fuel assembly (without CC)	*2500*	*4.05*	*4.20*
CE 15x15 fuel assembly (without CC)	*2800*	*4.30*	*4.45*
Assembly Class and Type	*Soluble Boron Loading (ppm)*	*32 Damaged Fuels*	
		24 Poison Plate Configuration	*24 Poison Plate Configuration*
		Type A1	*Type A1*
WE 14x14 fuel assembly (without CC)	*1800*	*3.80*	*3.95*
	2100	*4.10*	*4.25*
	2300	*4.30*	*4.45*
	2500	*4.50*	*4.65*
CE 14x14 fuel assembly (without CC)	*1800*	*3.60*	*3.75*
	2100	*3.90*	*4.05*
	2300	*4.10*	*4.25*
	2500	*4.30*	*4.45*
	2600	*4.35*	*4.55*
Assembly Class and Type	*Soluble Boron Loading (ppm)*	*28 Damaged Fuels*	
		32 Poison Plate Configuration	
		Type A1-32	*Type A2-32*
WE 17x17 fuel assembly (without CC)	*2500*	*4.45*	*4.65*
CE 15x15 fuel assembly (without CC)	*2800*	*4.70*	*4.95*
Assembly Class and Type	*Soluble Boron Loading (ppm)*	*28 Damaged Fuels*	
		24 Poison Plate Configuration	
		Type A1	*Type A2*
CE 14x14 fuel assembly (without CC)	*1800*	*3.70*	*3.85*
	2100	*4.00*	*4.15*
	2300	*4.20*	*4.35*
	2500	*4.40*	*4.50*
	2600	*4.45*	*4.65*
Assembly Class and Type	*Soluble Boron Loading (ppm)*	*16 Damaged Fuels*	
		24 Poison Plate Configuration	
		Type A1	*Type A2*
CE 14x14 fuel assembly (without CC)	*1800*	*3.80*	*3.95*
	2100	*4.10*	*4.25*
	2300	*4.30*	*4.45*
	2500	*4.50*	*4.70*
	2600	*4.60*	*4.80*

All changes on this page are Amd 15.

Table M.6-62
Maximum Initial Enrichment for Each Configuration, wt. % U-235 – Failed Fuel

Assembly Class and Type	*FFC Array Size*	*Soluble Boron Loading (ppm)*	*8 Failed Fuels*	
			24 Poison Plate Configuration	
			Type A1	*Type A2*
WE 17x17 fuel assembly (without CC) *CE 15x15 fuel assembly (without CC)*	*14x14*	*2500*	*4.15*	*4.30*
		2800	*4.40*	*4.60*
	15x15	*2500*	*4.10*	*4.25*
		2800	*4.40*	*4.55*
	16x16(*)	*2500*	*4.05*	*4.20*
		2800	*4.35*	*4.50*
	17x17(*)	*2500*	*4.00*	*4.15*
		2800	*4.30*	*4.45*
WE 14x14 fuel assembly (without CC)	*14x14*	*1800*	*4.15*	*4.30*
		2100	*4.50*	*4.70*
		2300	*4.75*	*4.95*
		2500	*4.95*	*5.00*
CE 14x14 fuel assembly (without CC)	*14x14*	*2600*	*4.70*	*4.90*
Assembly Class and Type	*FFC Array Size*	*Soluble Boron Loading (ppm)*	*8 Failed Fuels*	
			32 Poison Plate Configuration	
			Type A1-32	*Type A2-32*
WE 17x17 fuel assembly (without CC) *CE 15x15 fuel assembly (without CC)*	*14x14*	*2500*	*4.55*	*4.75*
		2800	*4.85*	*5.00*
	15x15	*2500*	*4.50*	*4.70*
		2800	*4.80*	*5.00*
	16x16(*)	*2500*	*4.45*	*4.65*
		2800	*4.75*	*4.95*
	17x17(*)	*2500*	*4.40*	*4.60*
		2800	*4.65*	*4.90*

(*) Not applicable to CE 15x15 fuel assembly

Table M.6-63
Intact Fuel Criticality Evaluation - WE17
(Part 1 of 5)

Basket Type A1			
2800 ppm	**_no CC_**	**_4.30 wt. % U-235_**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
_60_	_0.9280_	_0.0007_	_0.9294_
_70_	_0.9333_	_0.0007_	_0.9346_
_80_	_0.9325_	_0.0007_	_0.9338_
_90_	_0.9278_	_0.0007_	_0.9291_
_100_	_0.9189_	_0.0006_	_0.9202_
2800 ppm	**_CC_**	**_4.25 wt. % U-235_**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
_60_	_0.9209_	_0.0007_	_0.9223_
_70_	_0.9285_	_0.0007_	_0.9299_
_80_	_0.9339_	_0.0007_	_0.9353_
_90_	_0.9320_	_0.0007_	_0.9333_
_100_	_0.9288_	_0.0006_	_0.9300_

Table M.6-63
Intact Fuel Criticality Evaluation - WE17
(Part 2 of 5)

Basket Type A1-32			
2500 PPM	*no CC*	*4.45 wt. % U-235*	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9191	0.0006	0.9204
70	0.9300	0.0007	0.9314
80	0.9342	0.0006	0.9354
90	0.9331	0.0006	0.9343
100	0.9286	0.0007	0.9300
2500 PPM	*CC*	*4.40 wt. % U-235*	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9094	0.0007	0.9107
70	0.9235	0.0007	0.9249
80	0.9315	0.0007	0.9328
90	0.9344	0.0006	0.9357
100	0.9338	0.0006	0.9350
2800 PPM	*no CC*	*4.75 wt. % U-235*	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9260	0.0006	0.9273
70	0.9349	0.0007	0.9363
80	0.9352	0.0009	0.9369
90	0.9322	0.0007	0.9336
100	0.9276	0.0007	0.9290
2800 PPM	*CC*	*4.70 wt. % U-235*	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9159	0.0008	0.9174
70	0.9295	0.0008	0.9310
80	0.9346	0.0007	0.9360
90	0.9355	0.0007	0.9369
100	0.9338	0.0006	0.9351

Table M.6-63
Intact Fuel Criticality Evaluation - WE17
(Part 3 of 5)

Basket Type A2-32			
2500 PPM	**no CC**	**4.65 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9155	0.0007	0.9169
70	0.9276	0.0007	0.9291
80	0.9343	0.0007	0.9357
90	0.9342	0.0007	0.9356
100	0.9313	0.0007	0.9327
2500 PPM	**CC**	**4.60 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9032	0.0006	0.9043
70	0.9195	0.0007	0.9209
80	0.9286	0.0007	0.9300
90	0.9342	0.0007	0.9356
100	0.9351	0.0007	0.9365
2800 PPM	**no CC**	**5.00 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9227	0.0006	0.9239
70	0.9345	0.0007	0.9358
80	0.9366	0.0007	0.9380
90	0.9360	0.0006	0.9372
100	0.9305	0.0006	0.9317
2800 PPM	**CC**	**4.90 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9090	0.0007	0.9104
70	0.9235	0.0007	0.9249
80	0.9322	0.0007	0.9337
90	0.9351	0.0007	0.9366
100	0.9332	0.0007	0.9346

Table M.6-63
Intact Fuel Criticality Evaluation - WE17
(Part 4 of 5)

Basket Type B1			
2800 PPM	no CC	4.85 wt. % U-235	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9236	0.0007	0.9250
70	0.9347	0.0006	0.9360
80	0.9367	0.0006	0.9380
90	0.9349	0.0007	0.9363
100	0.9318	0.0007	0.9332
Basket Type B2			
2800 PPM	no CC	4.95 wt. % U-235	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9166	0.0007	0.9180
70	0.9280	0.0006	0.9291
80	0.9340	0.0007	0.9355
90	0.9353	0.0006	0.9366
100	0.9308	0.0008	0.9324
Basket Type B1-r			
2800 PPM	no CC	4.60 wt. % U-235	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9252	0.0006	0.9265
70	0.9321	0.0007	0.9334
80	0.9355	0.0007	0.9370
90	0.9317	0.0006	0.9330
100	0.9243	0.0006	0.9256
Basket Type B2-r			
2800 PPM	no CC	4.75 wt. % U-235	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9218	0.0006	0.9230
70	0.9327	0.0007	0.9341
80	0.9354	0.0006	0.9366
90	0.9328	0.0006	0.9340
100	0.9269	0.0006	0.9282

Table M.6-63
Intact Fuel Criticality Evaluation - WE17
(Part 5 of 5)

Basket Type C1			
2800 PPM	**no CC**	**5.00 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.8968	0.0007	0.8981
70	0.9115	0.0007	0.9129
80	0.9172	0.0007	0.9186
90	0.9194	0.0006	0.9206
100	0.9159	0.0007	0.9174
Basket Type C1-r			
2800 PPM	**no CC**	**4.85 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9238	0.0007	0.9251
70	0.9338	0.0007	0.9351
80	0.9364	0.0007	0.9378
90	0.9336	0.0007	0.9350
100	0.9292	0.0006	0.9305
Basket Type C2-r			
2800 PPM	**no CC**	**5.00 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9209	0.0006	0.9221
70	0.9313	0.0007	0.9326
80	0.9367	0.0007	0.9380
90	0.9350	0.0006	0.9363
100	0.9303	0.0007	0.9317

Table M.6-64
Intact Fuel Criticality Evaluation - CE15
(Part 1 of 2)

Basket Type A1			
2800 PPM	***no CC***	***4.45 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9280*	*0.0006*	*0.9293*
70	*0.9340*	*0.0007*	*0.9354*
80	*0.9364*	*0.0007*	*0.9377*
90	*0.9340*	*0.0006*	*0.9352*
100	*0.9296*	*0.0006*	*0.9308*
Basket Type A2			
2800 PPM	***no CC***	***4.55 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9221*	*0.0007*	*0.9234*
70	*0.9309*	*0.0007*	*0.9322*
80	*0.9331*	*0.0007*	*0.9344*
90	*0.9316*	*0.0008*	*0.9331*
100	*0.9266*	*0.0007*	*0.9279*

Table M.6-64
Intact Fuel Criticality Evaluation - CE15
(Part 2 of 2)

Basket Type A1-32			
2500 PPM	**no CC**	**4.55 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9152	0.0006	0.9165
70	0.9250	0.0007	0.9264
80	0.9320	0.0007	0.9334
90	0.9351	0.0007	0.9365
100	0.9325	0.0007	0.9338
2800 PPM	**no CC**	**4.85 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9193	0.0008	0.9209
70	0.9299	0.0006	0.9311
80	0.9337	0.0007	0.9350
90	0.9320	0.0009	0.9338
100	0.9302	0.0008	0.9318
Basket Type A2-32			
2500 PPM	**no CC**	**4.75 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9092	0.0007	0.9106
70	0.9221	0.0007	0.9235
80	0.9310	0.0007	0.9324
90	0.9344	0.0007	0.9358
100	0.9357	0.0007	0.9370
2800 PPM	**no CC**	**5.00 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9100	0.0008	0.9116
70	0.9234	0.0007	0.9248
80	0.9292	0.0007	0.9306
90	0.9299	0.0007	0.9312
100	0.9283	0.0007	0.9298

Table M.6-65
Damaged Fuel Sensitivity Study - WE17

Optimum Rod Pitch				
% of Nominal Fuel Rod Pitch	Most Reactive IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
98	80	0.9256	0.0007	0.9270
99	80	0.9280	0.0007	0.9293
100	80	0.9306	0.0006	0.9318
101.8	75	0.9325	0.0006	0.9337
103.5	70	0.9324	0.0007	0.9338
Single Shear				
Shear Distance (cm)	**Most Reactive IMD (%)**	k_{KENO}	σ_{KENO}	k_{EFF}
0.15	80	0.9318	0.0007	0.9332
0.30	75	0.9307	0.0007	0.9320
0.45	75	0.9308	0.0007	0.9321
0.55	75	0.9299	0.0008	0.9315
Poison Plate Shift				
Shift Distance (inch)	**Most Reactive IMD (%)**	k_{KENO}	σ_{KENO}	k_{EFF}
0.50	80	0.9309	0.0007	0.9322
1.00	75	0.9318	0.0007	0.9331
Missing Fuel Rod				
Number of Missing Fuel Rods	**Most Reactive IMD (%)**	k_{KENO}	σ_{KENO}	k_{EFF}
2	70	0.9294	0.0007	0.9308
4	75	0.9279	0.0006	0.9291
8	70	0.9219	0.0006	0.9232
16	65	0.9128	0.0006	0.9140

Notes:

The sensitivity study is performed for Basket Type A1 loaded with 32 damaged fuels of 4.00 wt. % U-235 enrichment. The soluble boron concentration is 2500 ppm.

The double shear configuration is not analyzed for WE17 since the fuel compartment is not wide enough to accommodate an additional rod of fuel rods.

Table M.6-66
Damaged Fuel Sensitivity Study – WE14

Optimum Rod Pitch				
% of Nominal Fuel Rod Pitch	**Most Reactive IMD (%)**	k_{KENO}	σ_{KENO}	k_{EFF}
98	70	0.8925	0.0007	0.8940
100	65	0.9066	0.0007	0.9080
104	65	0.9245	0.0007	0.9258
110	60	0.9362	0.0007	0.9377
113	65	0.9362	0.0007	0.9376
Single Shear				
Shear Distance (cm)	**Most Reactive IMD (%)**	k_{KENO}	σ_{KENO}	k_{EFF}
0.50	70	0.9119	0.0007	0.9134
1.00	70	0.9126	0.0007	0.9139
1.50	70	0.9112	0.0007	0.9125
2.00	65	0.9083	0.0008	0.9099
Double Shear				
Shear Distance (cm)	**Most Reactive IMD (%)**	k_{KENO}	σ_{KENO}	k_{EFF}
0.53	75	0.9289	0.0007	0.9302
0.70	70	0.9285	0.0008	0.9300
1.00	75	0.9306	0.0008	0.9322
1.51	70	0.9301	0.0007	0.9315
Poison Plate Shift				
Shift Distance (inch)	**Most Reactive IMD (%)**	k_{KENO}	σ_{KENO}	k_{EFF}
0.50	65	0.9072	0.0007	0.9086
1.00	70	0.9065	0.0006	0.9078
Missing Fuel Rod				
Number of Missing Fuel Rods	**Most Reactive IMD (%)**	k_{KENO}	σ_{KENO}	k_{EFF}
2	75	0.9043	0.0007	0.9056
4	65	0.9030	0.0008	0.9046
8	65	0.8977	0.0008	0.8993
16	60	0.8891	0.0008	0.8907

Note:
The sensitivity study is performed for Basket Type A1 loaded with 32 damaged fuels of 4.50 wt. % U-235 enrichment. The soluble boron concentration is 2500 ppm.

Table M.6-67
Damaged Fuel Sensitivity Study – CE14

Optimum Rod Pitch				
% of Nominal Fuel Rod Pitch	***Most Reactive IMD (%)***	k_{KENO}	σ_{KENO}	k_{EFF}
98	*75*	*0.9097*	*0.0007*	*0.9110*
100	*75*	*0.9196*	*0.0007*	*0.9211*
103	*70*	*0.9310*	*0.0008*	*0.9326*
105	*70*	*0.9336*	*0.0006*	*0.9348*
108	*65*	*0.9348*	*0.0006*	*0.9360*
Single Shear				
Shear Distance (cm)	***Most Reactive IMD (%)***	k_{KENO}	σ_{KENO}	k_{EFF}
0.30	*75*	*0.9223*	*0.0006*	*0.9236*
0.60	*75*	*0.9229*	*0.0007*	*0.9243*
0.90	*75*	*0.9226*	*0.0007*	*0.9240*
1.05	*70*	*0.9221*	*0.0007*	*0.9235*
Double Shear				
Shear Distance (cm)	***Most Reactive IMD (%)***	k_{KENO}	σ_{KENO}	k_{EFF}
0.56	*80*	*0.9349*	*0.0007*	*0.9363*
0.63	*75*	*0.9357*	*0.0007*	*0.9370*
Poison Plate Shift				
Shift Distance (inch)	***Most Reactive IMD (%)***	k_{KENO}	σ_{KENO}	k_{EFF}
0.50	*70*	*0.9202*	*0.0008*	*0.9217*
1.00	*70*	*0.9203*	*0.0008*	*0.9219*
Missing Fuel Rod				
Number of Missing Fuel Rods	***Most Reactive IMD (%)***	k_{KENO}	σ_{KENO}	k_{EFF}
2	*75*	*0.9180*	*0.0007*	*0.9194*
4	*75*	*0.9158*	*0.0006*	*0.9171*
8	*70*	*0.9132*	*0.0008*	*0.9148*
12	*60*	*0.9087*	*0.0006*	*0.9100*

Notes:

The sensitivity study is performed for Basket Type A1 loaded with 32 damaged fuels of 4.35 wt. % U-235 enrichment. The soluble boron concentration is 2600 ppm.

For the double shear, the maximum possible shear distance is 0.63 cm as shown in the table.

Table M.6-68
Damaged Fuel Criticality Evaluation - WE17
(Part 1 of 2)

Basket Type A1: 28-Damaged Fuel Assemblies			
2500 ppm	*Optimum Rod Pitch*	*4.05 wt. % U-235*	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9289*	*0.0007*	*0.9302*
65	*0.9339*	*0.0007*	*0.9353*
70	*0.9373*	*0.0007*	*0.9387*
75	*0.9344*	*0.0007*	*0.9358*
100	*0.9197*	*0.0006*	*0.9209*
2800 ppm	*Optimum Rod Pitch*	*4.30 wt. % U-235*	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9320*	*0.0008*	*0.9336*
65	*0.9351*	*0.0007*	*0.9365*
70	*0.9358*	*0.0006*	*0.9370*
75	*0.9350*	*0.0006*	*0.9362*
100	*0.9149*	*0.0007*	*0.9163*
Basket Type A2: 28-Damaged Fuel Assemblies			
2500 PPM	*Optimum Rod Pitch*	*4.20 wt. % U-235*	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
70	*0.9341*	*0.0006*	*0.9354*
75	*0.9361*	*0.0006*	*0.9373*
80	*0.9349*	*0.0007*	*0.9362*
90	*0.9309*	*0.0006*	*0.9322*
100	*0.9227*	*0.0007*	*0.9240*
2800 PPM	*Optimum Rod Pitch – 28-Damaged model*	*28- Damaged fuels: 4.45 wt. % U-235*	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
70	*0.9332*	*0.0007*	*0.9347*
75	*0.9349*	*0.0006*	*0.9361*
80	*0.9331*	*0.0007*	*0.9345*
90	*0.9277*	*0.0006*	*0.9289*
100	*0.9183*	*0.0006*	*0.9195*

Table M.6-68
Damaged Fuel Criticality Evaluation - WE17
(Part 2 of 2)

Basket Type A1-32: 32 Damaged Fuel Assemblies			
2500 ppm	**Optimum Rod Pitch**	**4.45 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
70	0.9345	0.0006	0.9358
75	0.9350	0.0007	0.9364
80	0.9361	0.0006	0.9373
90	0.9341	0.0006	0.9354
100	0.9258	0.0006	0.9271
2800 ppm	**Optimum Rod Pitch**	**4.70 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
70	0.9331	0.0007	0.9345
75	0.9344	0.0007	0.9357
80	0.9342	0.0006	0.9354
90	0.9283	0.0007	0.9297
100	0.9196	0.0008	0.9211
Basket Type A2-32: 32 Damaged Fuel Assemblies			
2500 PPM	**Optimum Rod Pitch**	**4.65 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
70	0.9329	0.0008	0.9344
75	0.9342	0.0007	0.9356
80	0.9348	0.0007	0.9362
90	0.9334	0.0007	0.9349
100	0.9280	0.0007	0.9293
2800 PPM	**Optimum Rod Pitch**	**4.95 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
70	0.9343	0.0007	0.9357
75	0.9360	0.0007	0.9373
80	0.9356	0.0007	0.9370
90	0.9316	0.0007	0.9329
100	0.9253	0.0007	0.9267

Table M.6-69
Damaged Fuel Criticality Evaluation – WE14
(Part 1 of 2)

Basket Type A1 - 32 Damaged Fuel Assemblies			
1800 ppm	**Optimum Rod Pitch**	**3.80 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9338	0.0007	0.9352
70	0.9357	0.0006	0.9369
80	0.9338	0.0008	0.9354
90	0.9271	0.0006	0.9283
100	0.9161	0.0006	0.9173
2100 ppm	**Optimum Rod Pitch**	**4.10 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9351	0.0006	0.9364
70	0.9335	0.0006	0.9348
80	0.9266	0.0007	0.9279
90	0.9120	0.0007	0.9135
100	0.8984	0.0006	0.8996
2300 PPM	**Optimum Rod Pitch**	**4.30 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9352	0.0006	0.9364
70	0.9333	0.0008	0.9348
80	0.9243	0.0007	0.9256
90	0.9118	0.0007	0.9131
100	0.8943	0.0007	0.8956
2500 PPM	**Optimum Rod Pitch**	**4.50 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9362	0.0007	0.9377
70	0.9345	0.0007	0.9358
80	0.9260	0.0007	0.9273
90	0.9151	0.0006	0.9163
100	0.8990	0.0007	0.9003

Table M.6-69
Damaged Fuel Criticality Evaluation – WE14
(Part 2 of 2)

Basket Type A2 - 32 Damaged Fuel Assemblies			
1800 ppm	***Optimum Rod Pitch***	***3.95 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9270*	*0.0007*	*0.9285*
70	*0.9341*	*0.0007*	*0.9354*
80	*0.9352*	*0.0008*	*0.9368*
90	*0.9304*	*0.0007*	*0.9317*
100	*0.9231*	*0.0006*	*0.9244*
2100 ppm	***Optimum Rod Pitch***	***4.25 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9323*	*0.0007*	*0.9338*
70	*0.9339*	*0.0007*	*0.9353*
80	*0.9281*	*0.0006*	*0.9294*
90	*0.9168*	*0.0006*	*0.9181*
100	*0.9033*	*0.0007*	*0.9048*
2300 PPM	***Optimum Rod Pitch***	***4.45 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9317*	*0.0007*	*0.9330*
70	*0.9325*	*0.0007*	*0.9339*
80	*0.9294*	*0.0007*	*0.9309*
90	*0.9214*	*0.0007*	*0.9228*
100	*0.9070*	*0.0007*	*0.9085*
2500 PPM	***Optimum Rod Pitch***	***4.65 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9338*	*0.0007*	*0.9352*
70	*0.9304*	*0.0006*	*0.9316*
80	*0.9214*	*0.0006*	*0.9226*
90	*0.9094*	*0.0007*	*0.9108*
100	*0.8931*	*0.0007*	*0.8944*

Table M.6-70
Damaged Fuel Criticality Evaluation – CE14
(Part 1 of 6)

Basket Type A1 : 32 Damaged Fuel Assemblies			
1800 ppm	**Optimum Rod Pitch**	**3.60 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9270	0.0007	0.9283
70	0.9339	0.0006	0.9351
80	0.9326	0.0007	0.9339
90	0.9262	0.0006	0.9274
100	0.9194	0.0006	0.9207
2100 PPM	**Optimum Rod Pitch**	**3.90 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9268	0.0006	0.9281
70	0.9333	0.0007	0.9347
80	0.9332	0.0006	0.9344
90	0.9274	0.0006	0.9286
100	0.9208	0.0007	0.9222
2300 ppm	**Optimum Rod Pitch**	**4.10 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9336	0.0006	0.9349
70	0.9348	0.0007	0.9362
80	0.9324	0.0006	0.9337
90	0.9244	0.0007	0.9257
100	0.9124	0.0007	0.9137
2500 ppm	**Optimum Rod Pitch**	**4.30 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9362	0.0007	0.9375
70	0.9375	0.0007	0.9388
80	0.9325	0.0007	0.9338
90	0.9219	0.0006	0.9231
100	0.9084	0.0006	0.9097
2600 ppm	**Optimum Rod Pitch**	**4.35 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9326	0.0006	0.9338
70	0.9331	0.0006	0.9344
80	0.9275	0.0007	0.9288
90	0.9188	0.0006	0.9199
100	0.9038	0.0006	0.9051

Table M.6-70
Damaged Fuel Criticality Evaluation – CE14
(Part 2 of 6)

Basket Type A1 : 28 Damaged Fuel Assemblies			
1800 ppm	**Optimum Rod Pitch**	**3.70 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9273	0.0007	0.9286
70	0.9354	0.0007	0.9367
80	0.9337	0.0006	0.9350
90	0.9295	0.0006	0.9308
100	0.9207	0.0006	0.9220
2100 PPM	**Optimum Rod Pitch**	**4.00 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9310	0.0007	0.9323
70	0.9364	0.0007	0.9378
80	0.9336	0.0006	0.9348
90	0.9260	0.0007	0.9273
100	0.9164	0.0006	0.9176
2300 ppm	**Optimum Rod Pitch**	**4.20 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9322	0.0007	0.9335
70	0.9364	0.0007	0.9378
80	0.9327	0.0007	0.9340
90	0.9244	0.0007	0.9258
100	0.9119	0.0007	0.9132
2500 ppm	**Optimum Rod Pitch**	**4.40 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9346	0.0007	0.9360
70	0.9361	0.0006	0.9373
80	0.9313	0.0006	0.9325
90	0.9229	0.0007	0.9243
100	0.9093	0.0006	0.9105
2600 ppm	**Optimum Rod Pitch**	**4.45 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9338	0.0006	0.9350
70	0.9334	0.0007	0.9347
80	0.9270	0.0006	0.9282
90	0.9182	0.0007	0.9195
100	0.9069	0.0006	0.9081

Table M.6-70
Damaged Fuel Criticality Evaluation – CE14
(Part 3 of 6)

Basket Type A1 : 16 Damaged Fuel Assemblies			
1800 ppm	**Optimum Rod Pitch**	**3.80 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9229	0.0007	0.9243
70	0.9324	0.0006	0.9336
80	0.9335	0.0007	0.9348
90	0.9304	0.0007	0.9318
100	0.9252	0.0007	0.9267
2100 PPM	**Optimum Rod Pitch**	**4.10 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9297	0.0008	0.9312
70	0.9344	0.0007	0.9358
80	0.9327	0.0007	0.9341
90	0.9275	0.0006	0.9287
100	0.9173	0.0006	0.9186
2300 ppm	**Optimum Rod Pitch**	**4.30 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9322	0.0007	0.9335
70	0.9344	0.0006	0.9357
80	0.9311	0.0007	0.9325
90	0.9238	0.0007	0.9251
100	0.9135	0.0006	0.9148
2500 ppm	Optimum Rod Pitch	4.50 wt. % U-235	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9329	0.0007	0.9342
70	0.9350	0.0007	0.9363
80	0.9308	0.0007	0.9321
90	0.9215	0.0008	0.9230
100	0.9099	0.0008	0.9115
2600 ppm	**Optimum Rod Pitch**	**4.60 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9346	0.0007	0.9359
70	0.9350	0.0007	0.9364
80	0.9308	0.0006	0.9321
90	0.9214	0.0008	0.9230
100	0.9097	0.0007	0.9110

Table M.6-70
Damaged Fuel Criticality Evaluation – CE14
(Part 4 of 6)

Basket Type A2 : 32 Damaged Fuel Assemblies			
1800 ppm	**Optimum Rod Pitch**	**3.75 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9252	0.0007	0.9265
70	0.9333	0.0007	0.9347
80	0.9357	0.0006	0.9370
90	0.9319	0.0007	0.9332
100	0.9226	0.0006	0.9239
2100 PPM	**Optimum Rod Pitch**	**4.05 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9293	0.0007	0.9307
70	0.9353	0.0007	0.9366
80	0.9334	0.0006	0.9347
90	0.9274	0.0006	0.9287
100	0.9166	0.0007	0.9179
2300 ppm	**Optimum Rod Pitch**	**4.25 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9306	0.0007	0.9320
70	0.9363	0.0007	0.9377
80	0.9331	0.0007	0.9344
90	0.9270	0.0006	0.9282
100	0.9141	0.0006	0.9153
2500 ppm	**Optimum Rod Pitch**	**4.45 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9286	0.0007	0.9300
70	0.9342	0.0007	0.9355
80	0.9331	0.0007	0.9345
90	0.9262	0.0006	0.9275
100	0.9177	0.0008	0.9192
2600 ppm	**Optimum Rod Pitch**	**4.55 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9328	0.0006	0.9340
70	0.9362	0.0007	0.9376
80	0.9308	0.0006	0.9321
90	0.9211	0.0007	0.9224
100	0.9110	0.0006	0.9123

Table M.6-70
Damaged Fuel Criticality Evaluation – CE14
(Part 5 of 6)

Basket Type A2 : 28 Damaged Fuel Assemblies			
1800 ppm	***Optimum Rod Pitch***	***3.85 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9259	0.0008	0.9275
70	0.9343	0.0007	0.9357
80	0.9362	0.0007	0.9376
90	0.9324	0.0007	0.9338
100	0.9247	0.0007	0.9261
2100 PPM	***Optimum Rod Pitch***	***4.15 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9283	0.0008	0.9298
70	0.9350	0.0006	0.9362
80	0.9344	0.0007	0.9358
90	0.9289	0.0006	0.9301
100	0.9198	0.0007	0.9211
2300 ppm	***Optimum Rod Pitch***	***4.35 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9306	0.0007	0.9320
70	0.9351	0.0007	0.9364
80	0.9324	0.0007	0.9338
90	0.9270	0.0007	0.9283
100	0.9163	0.0006	0.9175
2500 ppm	***Optimum Rod Pitch***	***4.50 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9295	0.0007	0.9309
70	0.9311	0.0007	0.9325
80	0.9283	0.0007	0.9297
90	0.9200	0.0006	0.9212
100	0.9089	0.0007	0.9103
2600 ppm	***Optimum Rod Pitch***	***4.65 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9330	0.0006	0.9342
70	0.9355	0.0007	0.9369
80	0.9319	0.0007	0.9333
90	0.9217	0.0007	0.9230
100	0.9103	0.0007	0.9116

Table M.6-70
Damaged Fuel Criticality Evaluation – CE14
(Part 6 of 6)

Basket Type A2 : 16 Damaged Fuel Assemblies			
1800 ppm	**Optimum Rod Pitch**	**3.95 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9234	0.0007	0.9247
70	0.9340	0.0007	0.9353
80	0.9353	0.0008	0.9368
90	0.9335	0.0006	0.9347
100	0.9268	0.0007	0.9282
2100 PPM	**Optimum Rod Pitch**	**4.25 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9245	0.0007	0.9259
70	0.9317	0.0006	0.9330
80	0.9322	0.0007	0.9337
90	0.9293	0.0007	0.9307
100	0.9219	0.0008	0.9234
2300 ppm	**Optimum Rod Pitch**	**4.45 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9285	0.0007	0.9299
70	0.9323	0.0007	0.9338
80	0.9315	0.0008	0.9331
90	0.9260	0.0007	0.9275
100	0.9164	0.0007	0.9178
2500 ppm	**Optimum Rod Pitch**	**4.70 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9329	0.0007	0.9342
70	0.9365	0.0007	0.9379
80	0.9330	0.0007	0.9343
90	0.9265	0.0007	0.9279
100	0.9160	0.0007	0.9174
2600 ppm	**Optimum Rod Pitch**	**4.80 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9308	0.0007	0.9321
70	0.9351	0.0009	0.9368
80	0.9318	0.0007	0.9331
90	0.9263	0.0007	0.9278
100	0.9154	0.0007	0.9167

Table M.6-71
Failed Fuel Criticality Evaluation - WE17
(Part 1 of 8)

Basket Type A1 : 8 Failed Fuel Cans			
2500 ppm	**FFC 14x14 Array**	**4.15 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9267	0.0007	0.9280
70	0.9322	0.0006	0.9335
80	0.9337	0.0007	0.9351
90	0.9304	0.0007	0.9318
100	0.9230	0.0006	0.9242
2500 ppm	**FFC 15x15 Array**	**4.10 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9272	0.0008	0.9287
70	0.9325	0.0006	0.9337
80	0.9330	0.0008	0.9345
90	0.9295	0.0007	0.9308
100	0.9233	0.0007	0.9247
2500 PPM	**FFC 16x16 Array**	**4.05 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9259	0.0007	0.9272
70	0.9335	0.0007	0.9348
80	0.9336	0.0007	0.9350
90	0.9301	0.0007	0.9315
100	0.9219	0.0007	0.9232
2500 PPM	**FFC 17x17 Array**	**4.00 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9239	0.0008	0.9255
70	0.9308	0.0007	0.9322
80	0.9331	0.0007	0.9345
90	0.9286	0.0006	0.9298
100	0.9218	0.0007	0.9231

Note:
The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-71
Failed Fuel Criticality Evaluation - WE17
(Part 2 of 8)

Basket Type A1 : 8 Failed Fuel Cans			
2800 ppm	**FFC 14x14 Array**	**4.40 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9289	0.0008	0.9304
70	0.9317	0.0007	0.9331
80	0.9328	0.0008	0.9343
90	0.9268	0.0006	0.9281
100	0.9182	0.0006	0.9195
2800 ppm	**FFC 15x15 Array**	**4.40 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9312	0.0006	0.9324
70	0.9359	0.0008	0.9374
80	0.9349	0.0009	0.9366
90	0.9301	0.0007	0.9314
100	0.9201	0.0006	0.9214
2800 PPM	**FFC 16x16 Array**	**4.35 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9309	0.0006	0.9321
70	0.9362	0.0007	0.9375
80	0.9349	0.0007	0.9363
90	0.9297	0.0007	0.9310
100	0.9208	0.0006	0.9221
2800 PPM	**FFC 17x17 Array**	**4.30 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9289	0.0007	0.9302
70	0.9343	0.0007	0.9357
80	0.9351	0.0007	0.9365
90	0.9281	0.0006	0.9293
100	0.9204	0.0007	0.9218

Note:
The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-71
Failed Fuel Criticality Evaluation - WE17
(Part 3 of 8)

Basket Type A2 : 8 Failed Fuel Cans			
2500 ppm	**FFC 14x14 Array**	**4.30 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9247	0.0007	0.9261
70	0.9312	0.0006	0.9325
80	0.9345	0.0007	0.9359
90	0.9318	0.0007	0.9331
100	0.9251	0.0007	0.9264
2500 ppm	**FFC 15x15 Array**	**4.25 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9234	0.0007	0.9248
70	0.9314	0.0007	0.9328
80	0.9324	0.0006	0.9337
90	0.9306	0.0006	0.9318
100	0.9244	0.0007	0.9258
2500 PPM	**FFC 16x16 Array**	**4.20 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9234	0.0008	0.9250
70	0.9323	0.0008	0.9338
80	0.9342	0.0006	0.9354
90	0.9314	0.0006	0.9327
100	0.9254	0.0006	0.9267
2500 PPM	**FFC 17x17 Array**	**4.15 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9225	0.0007	0.9238
70	0.9305	0.0006	0.9317
80	0.9337	0.0007	0.9350
90	0.9316	0.0006	0.9327
100	0.9253	0.0007	0.9267

Note:
The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-71
Failed Fuel Criticality Evaluation - WE17
(Part 4 of 8)

Basket Type A2 : 8 Failed Fuel Cans			
2800 ppm	***FFC 14x14 Array***	***4.60 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9277*	*0.0005*	*0.9287*
70	*0.9350*	*0.0007*	*0.9364*
80	*0.9342*	*0.0007*	*0.9356*
90	*0.9303*	*0.0007*	*0.9317*
100	*0.9238*	*0.0007*	*0.9252*
2800 ppm	***FFC 15x15 Array***	***4.55 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9283*	*0.0007*	*0.9297*
70	*0.9363*	*0.0007*	*0.9376*
80	*0.9348*	*0.0007*	*0.9362*
90	*0.9299*	*0.0008*	*0.9314*
100	*0.9236*	*0.0007*	*0.9250*
2800 PPM	***FFC 16x16 Array***	***4.50 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9296*	*0.0008*	*0.9312*
70	*0.9342*	*0.0007*	*0.9357*
80	*0.9360*	*0.0007*	*0.9373*
90	*0.9323*	*0.0008*	*0.9339*
100	*0.9233*	*0.0007*	*0.9246*
2800 PPM	***FFC 17x17 Array***	***4.45 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9280*	*0.0006*	*0.9292*
70	*0.9349*	*0.0007*	*0.9363*
80	*0.9351*	*0.0007*	*0.9364*
90	*0.9315*	*0.0007*	*0.9328*
100	*0.9237*	*0.0006*	*0.9249*

Note:

The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-71
Failed Fuel Criticality Evaluation - WE17
(Part 5 of 8)

Basket Type A1-32 : 8 Failed Fuel Cans			
2500 ppm	**FFC 14x14 Array**	**4.55 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9218	0.0007	0.9231
70	0.9305	0.0006	0.9317
80	0.9350	0.0007	0.9364
90	0.9334	0.0007	0.9347
100	0.9282	0.0007	0.9295
2500 ppm	**FFC 15x15 Array**	**4.50 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9216	0.0007	0.9230
70	0.9319	0.0007	0.9332
80	0.9339	0.0008	0.9355
90	0.9337	0.0007	0.9351
100	0.9296	0.0006	0.9309
2500 PPM	**FFC 16x16 Array**	**4.45 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9217	0.0008	0.9233
70	0.9327	0.0006	0.9340
80	0.9349	0.0007	0.9363
90	0.9346	0.0007	0.9360
100	0.9275	0.0007	0.9290
2500 PPM	**FFC 17x17 Array**	**4.40 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9203	0.0006	0.9215
70	0.9311	0.0008	0.9326
80	0.9349	0.0008	0.9365
90	0.9334	0.0007	0.9347
100	0.9289	0.0007	0.9303

Note:
The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-71
Failed Fuel Criticality Evaluation - WE17
(Part 6 of 8)

Basket Type A1-32 : 8 Failed Fuel Cans			
2800 ppm	***FFC 14x14 Array***	***4.85 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9264	0.0006	0.9276
70	0.9328	0.0008	0.9344
80	0.9353	0.0007	0.9367
90	0.9337	0.0007	0.9351
100	0.9265	0.0007	0.9280
2800 ppm	***FFC 15x15 Array***	***4.8i0 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9256	0.0007	0.9270
70	0.9352	0.0007	0.9366
80	0.9353	0.0006	0.9365
90	0.9335	0.0007	0.9350
100	0.9264	0.0006	0.9276
2800 PPM	***FFC 16x16 Array***	***4.75 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9265	0.0007	0.9278
70	0.9334	0.0006	0.9346
80	0.9362	0.0006	0.9374
90	0.9333	0.0007	0.9346
100	0.9277	0.0006	0.9289
2800 PPM	***FFC 17x17 Array***	***4.65 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9225	0.0006	0.9238
70	0.9316	0.0007	0.9329
80	0.9335	0.0008	0.9350
90	0.9302	0.0006	0.9315
100	0.9250	0.0006	0.9263

Note:

The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-71
Failed Fuel Criticality Evaluation - WE17
(Part 7 of 8)

Basket Type A2-32 : 8 Failed Fuel Cans			
2500 ppm	**FFC 14x14 Array**	**4.75 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9176	0.0007	0.9189
70	0.9291	0.0007	0.9305
80	0.9339	0.0007	0.9354
90	0.9342	0.0007	0.9357
100	0.9305	0.0007	0.9318
2500 ppm	**FFC 15x15 Array**	**4.70 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9171	0.0007	0.9184
70	0.9298	0.0007	0.9312
80	0.9338	0.0008	0.9354
90	0.9331	0.0006	0.9343
100	0.9304	0.0007	0.9318
2500 PPM	**FFC 16x16 Array**	**4.65 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9167	0.0008	0.9183
70	0.9294	0.0007	0.9308
80	0.9346	0.0008	0.9361
90	0.9339	0.0007	0.9352
100	0.9319	0.0007	0.9333
2500 PPM	**FFC 17x17 Array**	**4.60 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9176	0.0007	0.9189
70	0.9283	0.0007	0.9296
80	0.9358	0.0008	0.9373
90	0.9353	0.0007	0.9367
100	0.9333	0.0007	0.9347

Note:

The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-71
Failed Fuel Criticality Evaluation - WE17
(Part 8 of 8)

Basket Type A2-32 : 8 Failed Fuel Cans			
2800 ppm	**FFC 14x14 Array**	**5.00 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9182	0.0008	0.9198
70	0.9269	0.0007	0.9282
80	0.9304	0.0007	0.9317
90	0.9287	0.0007	0.9302
100	0.9253	0.0007	0.9266
2800 ppm	**FFC 15x15 Array**	**5.00 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9214	0.0007	0.9227
70	0.9297	0.0006	0.9310
80	0.9353	0.0007	0.9367
90	0.9323	0.0006	0.9335
100	0.9281	0.0008	0.9296
2800 PPM	**FFC 16x16 Array**	**4.95 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9223	0.0007	0.9236
70	0.9337	0.0007	0.9350
80	0.9354	0.0006	0.9366
90	0.9345	0.0008	0.9361
100	0.9286	0.0006	0.9299
2800 PPM	**FFC 17x17 Array**	**4.90 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9210	0.0007	0.9224
70	0.9318	0.0006	0.9331
80	0.9352	0.0007	0.9365
90	0.9338	0.0006	0.9350
100	0.9298	0.0007	0.9311

Note:

The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-72
Failed Fuel Criticality Evaluation – WE14
(Part 1 of 2)

Basket Type A1 : 8 Failed Fuel Cans			
1800 ppm	***FFC 14x14 Array***	***4.15 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9260*	*0.0007*	*0.9274*
70	*0.9333*	*0.0008*	*0.9348*
80	*0.9335*	*0.0007*	*0.9349*
90	*0.9314*	*0.0007*	*0.9328*
100	*0.9252*	*0.0008*	*0.9268*
2100 ppm	***FFC 14x14 Array***	***4.50 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9278*	*0.0007*	*0.9293*
70	*0.9330*	*0.0007*	*0.9345*
80	*0.9331*	*0.0008*	*0.9346*
90	*0.9289*	*0.0008*	*0.9304*
100	*0.9218*	*0.0008*	*0.9233*
2300 PPM	***FFC 14x14 Array***	***4.75 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9341*	*0.0007*	*0.9355*
70	*0.9357*	*0.0007*	*0.9371*
80	*0.9348*	*0.0007*	*0.9362*
90	*0.9289*	*0.0007*	*0.9303*
100	*0.9209*	*0.0009*	*0.9226*
2500 PPM	***FFC 14x14 Array***	***4.95 wt. % U-235***	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	*0.9327*	*0.0007*	*0.9340*
70	*0.9344*	*0.0008*	*0.9360*
80	*0.9329*	*0.0007*	*0.9342*
90	*0.9259*	*0.0007*	*0.9273*
100	*0.9157*	*0.0008*	*0.9173*

Note:

The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-72
Failed Fuel Criticality Evaluation – WE14
(Part 2 of 2)

Basket Type A2 : 8 Failed Fuel Cans			
1800 ppm	**FFC 14x14 Array**	**4.30 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9214	0.0007	0.9227
70	0.9306	0.0007	0.9321
80	0.9330	0.0007	0.9344
90	0.9327	0.0008	0.9343
100	0.9300	0.0007	0.9313
2100 ppm	**FFC 14x14 Array**	**4.70 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9261	0.0007	0.9275
70	0.9337	0.0008	0.9353
80	0.9349	0.0007	0.9364
90	0.9333	0.0007	0.9348
100	0.9274	0.0007	0.9287
2300 PPM	**FFC 14x14 Array**	**4.95 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9303	0.0007	0.9316
70	0.9349	0.0008	0.9364
80	0.9364	0.0008	0.9379
90	0.9302	0.0007	0.9316
100	0.9254	0.0010	0.9274
2500 PPM	**FFC 14x14 Array**	**5.00 wt. % U-235**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
60	0.9216	0.0007	0.9231
70	0.9258	0.0007	0.9272
80	0.9255	0.0009	0.9272
90	0.9209	0.0007	0.9223
100	0.9137	0.0007	0.9151

Note:

The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-73
Failed Fuel Criticality Evaluation – CE14

Basket Type A1 : 8 Failed Fuel Cans			
2600 ppm	**_FFC 14x14 Array_**	**_4.70 wt. % U-235_**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
_60_	_0.9321_	_0.0007_	_0.9335_
_70_	_0.9347_	_0.0007_	_0.9360_
_80_	_0.9297_	_0.0007_	_0.9311_
_90_	_0.9215_	_0.0007_	_0.9228_
_100_	_0.9083_	_0.0007_	_0.9097_
Basket Type A2 : 8 Failed Fuel Cans			
2600 ppm	**_FFC 14x14 Array_**	**_4.90 wt. % U-235_**	
IMD (%)	k_{KENO}	σ_{KENO}	k_{EFF}
_60_	_0.9320_	_0.0008_	_0.9336_
_70_	_0.9343_	_0.0007_	_0.9356_
_80_	_0.9322_	_0.0007_	_0.9336_
_90_	_0.9229_	_0.0008_	_0.9245_
_100_	_0.9151_	_0.0007_	_0.9165_

Note:
The KENO results are for the limiting fuel rod pitch of FFC failed fuel rod array. The 8 failed fuel locations are shown in Figure M.2-9.

Table M.6-74
Sensitivity with WE 17x17 fuels – Full Lattice and Reduced Lattice with Deletion of Rods

Model Description	k_{KENO}	1σ	$k_{eff} = k_{KENO} + 2\sigma$
WE 17x17 Fuel Assembly, 4.40 wt. %, 32 poison plates, Type A1 (15 mg B-10/cm²), 2500 ppm			
17x17 failed rod lattice			
17x17, Pitch = 0.496", IMD=60%	*0.9568*	*0.0007*	*0.9582*
17x17, Pitch = 0.496", IMD=60%	*0.9616*	*0.0006*	*0.9629*
17x17, Pitch = 0.496", IMD=70%	*0.9663*	*0.0006*	*0.9675*
17x17, Pitch = 0.496", IMD=75%	*0.9705*	*0.0007*	*0.9719*
17x17, Pitch = 0.496", IMD=80%	***0.9740***	***0.0007***	***0.9753***
17x17, Pitch = 0.496", IMD=90%	*0.9725*	*0.0007*	*0.9739*
17x17, Pitch = 0.496", IMD=100%	*0.9682*	*0.0007*	*0.9696*
17x17, 0.496", Remove 02 Rods, IMD=80%	*0.9700*	*0.0007*	*0.9714*
17x17, 0.496", Remove 04 Rods, IMD=80%	*0.9689*	*0.0008*	*0.9705*
17x17, 0.496", Remove 06 Rods, IMD=80%	*0.9663*	*0.0006*	*0.9676*
17x17, 0.496", Remove 08 Rods, IMD=80%	*0.9637*	*0.0006*	*0.9649*
17x17, 0.496", Remove 10 Rods, IMD=80%	*0.9625*	*0.0007*	*0.9639*
16x16 failed rod lattice			
16x16, Pitch = 0.527", IMD=60%	*0.9384*	*0.0009*	*0.9402*
16x16, Pitch = 0.527", IMD=60%	*0.9435*	*0.0007*	*0.9448*
16x16, Pitch = 0.527", IMD=70%	*0.9459*	*0.0006*	*0.9471*
16x16, Pitch = 0.527", IMD=75%	*0.9441*	*0.0006*	*0.9453*
16x16, Pitch = 0.527", IMD=80%	***0.9444***	***0.0007***	***0.9457***
16x16, Pitch = 0.527", IMD=90%	*0.9383*	*0.0007*	*0.9397*
16x16, Pitch = 0.527", IMD=100%	*0.9301*	*0.0006*	*0.9313*
16x16, 0.527", Remove 02 Rods, IMD=80%	*0.9425*	*0.0006*	*0.9437*
16x16, 0.527", Remove 04 Rods, IMD=80%	*0.9389*	*0.0007*	*0.9404*
16x16, 0.527", Remove 06 Rods, IMD=80%	*0.9367*	*0.0007*	*0.9380*
16x16, 0.527", Remove 08 Rods, IMD=80%	*0.9326*	*0.0007*	*0.9340*
16x16, 0.527", Remove 10 Rods, IMD=80%	*0.9281*	*0.0006*	*0.9294*

M.8.1 Procedures for Loading the Cask

Process flow diagrams for the NUHOMS® System operation are presented in Figure M.8-1 and Figure M.8-2. The location of the various operations may vary with individual plant requirements. The following steps describe the recommended generic operating procedures for the standardized NUHOMS® System.

M.8.1.1 Preparation of the TC and DSC

NOTE: If using the OS200 TC for transfer of the NUHOMS®-32PT DSC, verify that it has been fitted with an internal aluminum sleeve (refer to Drawing NUH-08-8004-SAR provided in Appendix U, Section U.1.5). This step, if required, can be performed at any time prior to placing the DSC in the TC.

1. Prior to placement in dry storage, the candidate intact *and damaged and failed* fuel assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification 2.1. Depending on the length of the fuel assemblies to be loaded, fuel spacers may be placed within the DSC to reduce the fuel assembly/DSC cavity gap in consideration of Part 71requirements. There are no requirements for fuel spacers under Part 72. Fuel spacers, if used, may be placed below the assembly, above the assembly, or both, and shall be evaluated for any adverse impact.

2. Prior to being placed in service, the TC is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification 5.4.2.d. Prior to being placed in service, the DSC should have the top shield plug, inner top cover and outer top cover test fitted and removed.

3. Place the TC in the vertical position in the cask decon area using the cask handling crane and the TC lifting yoke.

4. Place scaffolding around the cask so that the top cover plate and surface of the cask are easily accessible to personnel.

5. Remove the TC top cover plate and examine the cask cavity for any physical damage and ready the cask for service.

5a. If using the OS200 TC to load, verify that a cask spacer of appropriate height (refer to Drawing NUH-08-8005-SAR provided in Appendix U, Section U.1.5) is placed at the bottom of the TC.

6. Examine the DSC for any physical damage which might have occurred since the receipt inspection was performed. The DSC is to be cleaned and any loose debris removed.

7. Using a crane, lower the DSC into the cask cavity by the internal lifting lugs and rotate the DSC to match the cask and DSC alignment marks.

8. Fill the cask-DSC annulus with clean, demineralized water. Place the inflatable seal into the upper cask liner recess and seal the cask-DSC annulus by pressurizing the seal with compressed air.

8a. Place and verify that the bottom fuel assembly spacers, if required, are present in the fuel cells. Optionally, this step may be performed at any prior time.

9. *If damaged fuel assemblies are included in a specific campaign, place the required number of bottom end caps into the cell locations per Technical Specifications. Place and verify that the bottom fuel assembly spacers, if required, are present in the fuel cells. Optionally, this step may be performed at any prior time.*

10. *If failed fuel is to be loaded in the DSC, place the empty failed fuel can in the appropriate locations in the DSC in the appropriate locations in the DSC per Technical Specifications.*

11. Fill the DSC cavity with water from the fuel pool or an equivalent source which meets the requirements of Technical Specification 3.2.1. If loading 32PT-S100 or 32PT-L100 DSC (qualified for 100-ton crane capacity), drain neutron shield water from the TC.

    NOTE: A TC/DSC annulus pressurization tank filled with demineralized water as described above is connected to the top vent port of the TC via a hose to provide a positive head above the level of water in the TC/DSC annulus. This is an optional arrangement, which provides additional assurance that contaminated water from the fuel pool will not enter the TC/DSC annulus, provided a positive head is maintained at all times.

12. Position the cask lifting yoke and engage the cask lifting trunnions.

13. Visually inspect the yoke lifting hooks to insure that they are properly positioned and engaged on the cask lifting trunnions.

14. Connect the vacuum drying system (VDS) or optional liquid pump to the siphon port of the DSC and position the connecting hose such that the hose will not interfere with loading (yoke, fuel, shield plug, rigging, etc.). A flowmeter must be installed at a suitable location as part of this connection.

15. Move the scaffolding away from the cask as necessary.

16. Lift the cask just far enough to allow the weight of the cask to be distributed onto the yoke lifting hooks. Reinspect the lifting hooks to insure that they are properly positioned on the cask trunnions.

17. Optionally, secure a sheet of suitable material to the bottom of the TC to minimize the potential for ground-in contamination. This may also be done prior to initial placement of the cask in the decon area.

18. Prior to the cask being lifted into the fuel pool, the water level in the pool should be adjusted as necessary to accommodate the cask/DSC volume. If the water placed in the DSC cavity was obtained from the fuel pool, a level adjustment may not be necessary.

M.8.1.2 DSC Fuel Loading

1. Lift the cask/DSC and position it over the cask loading area of the spent fuel pool in accordance with the plant's 10CFR50 cask handling procedures.

2. Lower the cask into the fuel pool until the bottom of the cask is at the height of the fuel pool surface. As the cask is lowered into the pool, spray the exterior surface of the cask with demineralized water.

3. Place the cask in the location of the fuel pool designated as the cask loading area.

4. Disengage the lifting yoke from the cask lifting trunnions and move the yoke clear of the cask. Spray the lifting yoke with clean demineralized water as it is raised out of the fuel pool.

5. The potential for fuel misloading is essentially eliminated through the implementation of procedural and administrative controls. The controls instituted to ensure that *failed/damaged and/ or intact* fuel assemblies, Control Components (CCs) and Poison Rod Assemblies (PRAs), if required, are placed into a known cell location within a DSC, will typically consist of the following:

    - A cask/DSC loading plan is developed to verify that the *failed/damaged and/ or intact* fuel assemblies, and CCs, if applicable, meet the burnup, enrichment and cooling time parameters of Technical Specification 2.1. If PRAs are determined to be needed by Technical Specification 2.1, record the number required and the DSC cell location for each of the PRAs on the loading plan

    - The loading plan is independently verified and approved before the fuel load.

- A fuel movement schedule is then written, verified and approved based upon the loading plan. All fuel movements from any rack location are performed under strict compliance of the fuel movement schedule.
- *If loading damaged fuel assemblies, verify that the required number of bottom end caps are installed in appropriate fuel compartment tube locations.*
- *If loading failed fuel, verify that the required number of failed fuel cans are installed in the appropriate locations, or, once loaded with fuel, are installed in the appropriate locations in the basket.*

6. Prior to insertion of a spent fuel assembly (and BPRAs, if applicable) into the DSC, the identity of the assembly (and BPRAs, if applicable) is to be verified by two individuals using an underwater video camera or other means. Read and record the identification number from the fuel assembly (and BPRAs, if applicable) and check this identification number against the DSC loading plan which indicates which fuel assemblies (and BPRAs, if applicable) are acceptable for dry storage.

7. Position the fuel assembly for insertion into the selected DSC storage cell and load the fuel assembly. Repeat Step 6 for each SFA loaded into the DSC. *Damaged fuel assemblies or failed fuel assemblies may be loaded into the basket per Technical Specifications.* If applicable, insert the required number of PRAs at specific locations called out in the loading plan. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly and BPRAs, if applicable, in the DSC. Also record the location of each PRA inserted in the DSC (if applicable). *If loading damaged fuel assemblies, place top end caps over each damaged fuel assembly placed into the basket. If loading failed fuel, ensure that the failed fuel can lids are installed.*

8. After all the SFAs, BPRAs, and PRAs, if applicable, have been placed into the DSC and their identities verified, position the lifting yoke with the rigging cables connected to the top shield plug, adjust the rigging cables as needed to level the top shield plug and lower the shield plug onto the DSC.

   CAUTION: Verify that all the lifting height restrictions as a function of temperature specified in Technical Specification 5.3.1.A can be met in the following steps which involve lifting of the TC.

9. Visually verify that the top shield plug is properly seated onto the DSC.

10. Position the lifting yoke with the TC trunnions and verify that it is properly engaged.

11. Raise the TC to the pool surface. Prior to raising the top of the cask above the water surface, stop vertical movement.

12. Inspect the top shield plug to verify that it is properly seated onto the DSC. If not, lower the cask and reposition the top shield plug. Repeat Steps 11 and 12 as necessary.

13. Continue to raise the TC from the pool and spray the exposed portion of the cask with demineralized water until the top region of the cask is accessible.

14. Drain any excess water from the top of the DSC shield plug back to the fuel pool.

15. Check the radiation levels at the center of the top shield plug and around the perimeter of the cask.

16. If loading 32PT-S100 or 32PT-L100 DSC (qualified for 100-ton crane capacity), drain approximately 750 gallons of water (as indicated on the flowmeter) from the DSC back into the fuel pool or other suitable location using the VDS or optional liquid pump. Consistent with ISG-22 [8.7] guidance and Technical Specification 3.1.1, helium at 1-3 psig is used to backfill the DSC with an inert gas as water is being removed from the DSC.

38. Disengage the lifting yoke from the cask and lift the top shield plug from the DSC.

39. *If the DSC contains damaged or failed fuel, remove the top end cap or failed fuel can lid. Then remove the fuel or rod storage basket from the DSC and place the fuel or rod storage basket into the racks.* Remove the fuel from the DSC and place the fuel into the spent fuel racks.

40. Lower the top shield plug onto the DSC.

41. Visually verify that the top shield plug is properly positioned onto the DSC.

42. Engage the lifting yoke onto the cask trunnions.

43. Visually verify that the yoke lifting hooks are properly engaged with the cask trunnions.

44. Lift the cask by a small amount and verify that the lifting hooks are properly engaged with the trunnions.

45. Lift the cask to the pool surface. Prior to raising the top of the cask to the water surface, stop vertical movement and inspect the top shield plug to ensure that it is properly positioned.

46. Spray the exposed portion of the cask with demineralized water.

47. Visually inspect the top shield plug of the DSC to insure that it is properly seated onto the cask. If the top shield plug is not properly seated, lower the cask back to the fuel pool and reposition the plug.

48. Drain any excess water from the top of the top shield plug into the fuel pool.

49. Lift the cask from the pool. As the cask is rising out of the pool, spray the cask with demineralized water.

50. Move the cask to the cask decon area.

51. Check radiation levels around the perimeter of the cask. The cask exterior surface should be decontaminated if necessary.

52. Place scaffolding around the cask so that any point along the surface of the cask is easily accessible to personnel.

53. Ready the DSC vacuum drying system (VDS).

54. Connect the VDS to the vent port with the system open to atmosphere. Also connect the VDS to the siphon port and connect the other end of the system to the liquid pump. The pump discharge should be routed to the plant radwaste system or the spent fuel pool.

55. Open the valves on the vent port and siphon port of the VDS.

56. Activate the liquid pump.

57. Once the water stops flowing out of the DSC, deactivate the pump.

M.9 Acceptance Tests and Maintenance Program

Background for this particular UFSAR chapter:

Beginning with CoC 1004 Amendment 10, which was incorporated into UFSAR Revision 11, Chapter M.9, "Acceptance Tests and Maintenance Program," contained information which was incorporated by reference into the Technical Specifications (TS) associated with a particular amendment. It is known that certain general licensees reconcile the CoC 1004 UFSAR revisions provided to them to their loaded systems, pursuant to 10 CFR 72.48 and 10 CFR 72.212. In doing so they sometimes find the changed UFSAR portions incorporated by reference into the TS to be impossible to reconcile because the 10 CFR 72.48 regulation does not allow proposed activities which involve changes to the TS.

In order to facilitate this reconciliation process by general licensees, the following statements are provided, addressing the licensing basis for certain amendments, as they relate to certain UFSAR chapters which contain TS incorporated by reference. Additionally, so that the actual information is contained in the current CoC 1004 UFSAR, to facilitate the reconciliation by general licensees, the UFSAR Revision 11, 12, 13, and 14 versions of Chapter M.9 are inserted and annotated in this part of the UFSAR. For clarity, this includes annotating the version of Chapter M.9 directly associated with the latest UFSAR revision in which a change to Chapter M.9 occurred.

- Systems loaded to CoC 1004 Amendment 10 have Technical Specifications incorporated by reference from UFSAR Revisions 11 and 12 in Chapter M.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to systems loaded to Amendment 10.
- Systems loaded to CoC 1004 Amendment 11 have Technical Specifications incorporated by reference from UFSAR Revision 13 Chapter M.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 11.
- Note that CoC 1004 Amendment 12 was submitted and docketed, associated with a U.S. Department of Energy project, but due to a lack of review funding the NRC returned it without a review.
- Systems loaded to CoC 1004 Amendment 13 have Technical Specifications incorporated by reference from UFSAR Revisions 14 and 15 Chapter M.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 13.
- Systems loaded to CoC 1004 Amendment 14 have Technical Specifications incorporated by reference by FCN 721004-1575, which has been incorporated into UFSAR Revisions 16 and 17 Chapter M.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 14.
- *Systems loaded to CoC 1004 Amendment 15 have Technical Specifications incorporated by reference from UFSAR Revision 18 Chapter M.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 15.*

All changes on this page are AMD 11

M.9 Acceptance Tests and Maintenance Program

M.9.1 Acceptance Tests

The acceptance requirements for the NUHOMS®-32PT system are given in the *U*FSAR except as described in the following sections. The NUHOMS®-32PT DSC has been enhanced to provide leaktight confinement and the basket includes an updated poison plate design. Additional acceptance testing of the NUHOMS®-32PT DSC welds and poison plates are described.

M.9.1.1 Visual Inspection

Visual examinations are performed at the fabricator's facility to ensure that the NUHOMS®-32PT system components conform to the fabrication specifications and drawings.

Visual examination of all finished absorber plates and rods are done to ensure that they are free of cracks, porosity, blisters, or foreign substances. Dimensional inspections of the plates and rods are done to ensure that their functional requirements listed in M.9.17.1 are met.

M.9.1.2 Structural Tests

The NUHOMS®-32PT DSC confinement welds are designed, fabricated, tested and inspected in accordance with ASME B&PV Code Section III, Subsection NB [9.1] with exceptions as listed in Section M.3.1. The following requirements are unique to the NUHOMS®-32PT DSC:

- The inner bottom cover weld is inspected in accordance with Article NB-5231,
- The outer bottom cover weld root and cover are penetrant tested, and
- The outer top cover plate weld root and cover are penetrant tested.

The NUHOMS®-32PT DSC basket is designed, fabricated, and inspected in accordance with ASME B&PV Code Section III, Subsection NG [9.1] with exceptions as listed in Section M.3.1. The following requirement is unique to the NUHOMS®-32PT DSC basket:

- The fuel compartment welds are inspected in accordance with Article NG-5260.

M.9.1.3 Leak Tests

The NUHOMS®-32PT DSC confinement boundary is leak tested to verify that it is leaktight in accordance with ANSI N14.5 [9.2]. The personnel performing the leak test are qualified in accordance with SNT-TC-1A [9.14].

The leak tests are typically performed using the helium mass spectrometer method. Alternative methods are acceptable, provided that the required sensitivity is achieved.

M.9.1.4 Component Tests

The Standardized NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. No other components of the Standardized NUHOMS® system require testing, except as discussed in this Appendix.

M.9.1.5 Shielding Integrity Tests

No changes to Section 4.3.9 and Appendix U, Section U.9.1.5.

M.9.1.6 Thermal Acceptance Tests

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron absorbing materials, as specified in Section M.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section M.9.1.7.6.

M.9.1.7 Poison Acceptance

CAUTION

Sections M.9.1.7.1 through M.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 2) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated aluminum

(b) Boron carbide/aluminum metal matrix composite (MMC)

The 32PT DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content for these materials is given in Table M.9-1.

M.9.1.7.1 **Borated Aluminum**

See the Caution in Section M.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating primarily as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the

matrix of aluminum or aluminum alloy (other boron compounds, such as AlB_{12}, can also occur). For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section M.9.1.7.7. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

M.9.1.7.2 <u>Boron Carbide/Aluminum Metal Matrix Composites (MMC*s*)</u>

See the Caution in Section M.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, *molten metal infiltration,* or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding *or produced by molten metal infiltration* shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B_4C particles in MMCs shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 32PT DSC, MMCs shall pass the qualification testing specified in Section M.9.1.7.8, and shall subsequently be subject to the process controls specified in Section M.9.1.7.9.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section M.9.1.7.7. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

M.9.1.7.3 **Not Used**

M.9.1.7.4 **Visual Inspections of Neutron Absorbers**

See the Caution in Section M.9.1.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. Blisters shall be treated as non-conforming. For *clad MMCs*, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet. *Material that does not meet the following acceptance criteria shall be reworked, repaired, or scrapped.*

M.9.1.7.5 Other Visual Inspections Criteria (non-Technical Specifications)

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

M.9.1.7.6 Thermal Conductivity Testing of Poison Plates

Acceptance testing shall conform to ASTM E1225[1], ASTM E1461[2], or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. *Initial sampling shall be one test per lot, and may be reduced if the first five tests meet the specified minimum thermal conductivity. For cast products, the lot shall be defined by the heat or ingot. For other products, the lot shall be defined as material produced in a single production campaign using the same heat or lots of aluminum and boron carbide feed materials.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After *twenty five* tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The measured thermal conductivity values shall satisfy the minimum required conductivities as specified in Section M.4.3

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1 ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique."

2 ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method."

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section M.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

M.9.1.7.7 Specification for Acceptance Testing of Neutron Absorber *Content*

Acceptance testing for neutron absorber content shall be performed by either neutron transmission or by B-10 volume density measurement.

CAUTION
Portions of Section M.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 2) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

M.9.1.7.7.1 *Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission*

a) Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.1 inch diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard. Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.12].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than the statistically derived minimum thickness from M.9.1.7.7a) or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

M.9.1.7.7.2 *Specification for Acceptance Testing of Neutron Absorbers by B-10 Volume Density Measurement*

a) B-10 volume density measurement acceptance testing procedures shall be subject to approval by the certificate holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, as long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for B-10 volume density measurements shall be such that there is at least one density measurement for each 2000 square inches of final product in each lot.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

Areal density is determined by measuring the B-10 volume density in test samples and converting the measured values to areal density. The method of measurement of B-10 volume density shall be subject to approval by the certificate holder. The method of measurement of B-10 volume density shall be qualified against neutron transmission testing. Results of the two test methods shall be compared and a penalty shall be derived to account for the performance based results of neutron transmission testing.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B-10 areal densities are determined by volume density as described above. The lower tolerance limit of B-10 volume density is then determined, defined as the mean value of B-10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.12]. Finally, the minimum specified value of B-10 areal density is divided by the lower tolerance limit of B-10 volume density to arrive at the minimum plate thickness that provides the specified B-10 areal density.

Any plate that is thinner than the statistically derived minimum thickness from M.9.1.7.7.2 a) or the minimum design thickness, whichever is greater, shall be treated as nonconforming, with the following exception. Local depressions are acceptable, as long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

M.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

CAUTION

Portions of Section M.9.1.7.8.4 and Section M.9.1.7.8.5 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 2) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

M.9.1.7.8.1 Applicability and Scope

MMCs acceptable for use in the 32PT DSC are described in Section M.9.1.7.2.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section M.9.1.7.9 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the *C*ertificate *H*older.

M.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in Section M.9.1.7.8.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section M.9.1.7.8.5.

M.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for unclad MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for full density MMCs (clad or unclad) consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4]

[3] Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998

[4] Boralyn testing submitted to the NRC under docket 71-1027, 1998

M.9.1.7.8.4 **Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity**

At least three samples, one each from approximately the two ends and middle of the qualification material run shall be subject to:

a) **room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:**

- **Minimum yield strength, 0.2% offset: 1.5 ksi**
- **Minimum ultimate strength: 5 ksi**
- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

b) **Testing to verify more than 98% of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a means to be approved by the Certificate Holder. The maximum interconnected porosity is 0.5 volume %.**

c) ***Delamination Testing of Clad MMC***
Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage. *An example of such a test would be: (1) immerse a specimen at least 6 x 6 inches in water under pressure ≥30 psig for at least 24 hours, (2) place the specimen in a vacuum furnace preheated to at least 300°F and evacuate the furnace. Acceptance criterion: no blistering or delamination of the cladding.*

M.9.1.7.8.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

Uniformity of the boron distribution shall be verified either by:

a) **Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or**

5 ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products
6 ASTM E290, Standard Methods for Bend Testing of Materials for Ductility
7 ASTM E94, Recommended Practice for Radiographic Testing
8 ASTM E142, Controlling Quality of Radiographic Testing
9 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

b) **Quantitative testing for the B10 areal density, B10 density, the boron carbide weight fraction, *or the boron weight fraction,* on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section M.9.1.7.7, or by chemical analysis for boron carbide *or boron* content in the composite.**

M.9.1.7.8.6 Qualification Report

Qualification report shall be prepared by, or subject to approval by the Certificate Holder.

M.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION

Sections M.9.1.7.9.1 and M.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 2) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

M.9.1.7.9.1 **Applicability and Scope**

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section M.9.1.7.8 is required, depending on the characteristics of the material that could be affected by the process change.

M.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, reduce corrosion resistance, reduce the mechanical strength or ductility of the MMC.

M.9.1.7.9.3 **Identification and Control of Key Process Changes**

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section M.9.1.7.9.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that are established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average (d50) particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a magnesium-alloyed aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, and

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

M.9.1.7.10 <u>B_4C Linear Density Testing for Poison Rod Assemblies (PRAs)</u>

The PRAs are shown in Figure M.1-2, and additional physical requirements are listed in Table M.2-4. The B_4C poison is inserted into the stainless steel tubes shown in Figure M.1-2. Table M.2-4 specifies the minimum B_4C content per unit length in the axial direction of the rods for the various PRA designs. The minimum B_4C content per unit length is consistent with the criticality analysis (Section M.6) with an additional 25% margin.

Pellets or powder representing each powder lot shall be tested per ASTM C751 [9.6] or ASTM C750 (Type 2) [9.7] (or equivalent). Density and diameter shall be measured to verify conformance to the specification requirements.

Deviations from the specified dimensions or density may be accepted, so long as the resulting minimum B_4C mass per unit length is maintained.

<u>Justification for Durability of B_4C Pellets:</u>

B_4C is essentially inert and will not be attacked even by hot hydrofluoric or nitric acids[9.8]. It is insoluble in water [9.9], resistant to steam at temperatures of 200 to 300°C [9.10] and has a melting point of 2450°C [9.10]. Mechanically, B_4C is extremely hard (Mohs hardness of 9.3 vs. 10 for diamond) and is used in abrasion- and wear-resistant applications and in bullet-proof tiles. It has a compressive strength of 398,000 psi. In the PRAs, the B_4C pellets are sealed within stainless steel. With this configuration there is nothing that could cause the material to degrade

In the unlikely event that a pellet were to crack or break, the total mass would be confined by the steel to the same dimensions.

The irradiation-induced swelling is due to neutron capture by the ^{10}B isotope. Using data from [9.11] and by determining the neutron absorption in the B_4C (^{10}B capture) from the shielding analyses, the swelling is determined to be negligible ~ 0.00002%. Finally, according to [9.11], the first intergranular cracks do not start to appear until fluences are 5.5 orders of magnitude greater than those calculated for 50 years of operation. (1)

M.9.1.7.11 Linear Density Testing for AIC PRAs

The PRAs based on Silver-Indium-Cadmium neutron absorber material are also specified for criticality control and are designated as AIC PRAs. The Ag-In-Cd poison is inserted into the stainless steel tubes shown in Figure M.1-2. Table M.2-4a specifies the minimum silver content per unit length in the axial direction of the rods for the various AIC PRA designs. The minimum silver content per unit length is consistent with the criticality analysis (Section M.6) with an additional 25% margin.

The AIC PRAs are similar to the rod cluster control assemblies (RCCAs) that are primarily employed as control rods for reactor operations. The acceptance testing associated with the qualification of AIC PRAs for reactor use is sufficient for this purpose. The test methods include chemical and spectrochemical analysis of nuclear grade silver-indium-cadmium alloys to determine compliance with specifications per Table M.2-4a. The material shall be tested per ASTM C760-90 [9.13]. Density and diameter shall be measured to verify conformance to the specification requirements as non-irradiated metallic alloy.

The following illustrates one acceptable method and is intended to be utilized solely as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for AIC rod, minimum required linear Ag content per rod, is determined from a statistical analysis of the test results of their lot. The lower tolerance limit of silver is determined, defined as the mean value for the sample, minus k times the standard deviation where k is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.12]. Additionally, since the credit is less than 75%, only a chemical composition testing is sufficient.

Deviations from the specified dimensions or density may be accepted, as long as the resulting minimum silver mass per unit length is maintained.

AIC PRA materials are qualified for long term operations in the reactor core, which ensures long term performance as PRAs.

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1 With NRC approval of the CoC 1004 renewal application, the license is extended to a total of 60 years. The TLAAs identified as #1, #4, and #5 in Section 12.2 performed for the renewal of this design conservatively used a service life of 100 years.

All changes on this page are AMD 11

M.9.2 Maintenance Program

NUHOMS®-32PT system is a totally passive system and therefore requires little, if any, maintenance over the lifetime of the ISFSI. Typical NUHOMS®-32PT system maintenance tasks are performed in accordance with the *U*FSAR.

M.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, 2004 Edition through 2006 addenda.

9.2 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.3 Deleted.

9.4 Deleted.

9.5 "Aluminum Standards and Data, 2003," The Aluminum Association.

9.6 ASTM C751, "Standard Specification for Nuclear-Grade Boron Carbide Pellets."

9.7 ASTM C750, "Standard Specification for Nuclear-Grade Boron Carbide Powder."

9.8 The Merck Index, 9th edition, Merck & Co., 1976.

9.9 Grant (ed.), Hackh's Chemical Dictionary, 4th edition, McGraw-Hill, 1969.

9.10 Lipp, A., "Boron Carbide: Production, Properties, Application," Reprint from Technische Rundschau, Nos. 14, 28, 33 (1995) and 7 (1966).

9.11 Stoto, T. et al., "Swelling and Microcracking of Boron Carbide Subjected to Fast Neutron Irradiations," Journal of Applied Physics, Vol. 68, No.7, October 1, 1990, pp. 3198-3206.

9.12 Natrella, "Experimental Statistics," Dover, 2005.

9.13 *ASTM C760-90, "Standard Test Methods for Chemical and Spectrochemical Analysis of Nuclear-Grade Silver-Indium-Cadmium Alloys," ASTM International, West Conshohocken, PA, 2015.*

9.14 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

This version of Chapter M.9 is associated with CoC 1004 Amendment 15 and is from UFSAR Revision 18. Please see Page M.9 Introduction - 1 for a discussion as to why certain versions of Chapter M.9 are being maintained in the UFSAR.

All changes on this page are AMD 11

Table M.9-1
B10 Specification for the NUHOMS® - 32PT Poison Plates

Poison Type	*32PT Basket Type*	*Minimum Poison Loading (B10 g/cm²)*	*% Credit Used in Criticality Analysis*
Borated Aluminum /MMC	*A/B/C/D*	*0.007*	*90*
Borated Aluminum /MMC	*A1*	*0.015*	*90*
Borated Aluminum /MMC	*A2*	*0.020*	*90*

M.10 Radiation Protection

Section 7.4.1 discusses the anticipated cumulative dose exposure to site personnel during the fuel handling and transfer activities associated with utilizing one NUHOMS® HSM for storage of one DSC. Chapter 5 describes in detail the NUHOMS® operational procedures, several of which involve potential exposure to personnel.

M.10.1 Occupational Exposure

The occupational exposure results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section M.5.4.16. Loading 0.380 MTU fuel results in an increase in occupational exposure of 25%.

The expected occupational dose for placing a canister of spent fuel into dry storage is based on the operational steps outlined in Table 7.4-1. The total exposure for the occupational dose due to placing a single NUHOMS®-32PT DSC into storage is conservatively estimated to be 1.8 person-rem (32PT-S125/32PT-L125 DSC configuration) and 3.8 person-rem (32PT-S100/32PT-L100 DSC configuration). This is a very conservative estimate because the dose rates on and around the 32PT DSC used in these calculations are based on very conservative assumptions for the design-basis source terms (32PT-S100/32PT-L100 DSC with Heat Loading Zoning Configuration 2 from Chapter M.2). As in Section M.5, no credit is taken for the thicker door described in Section 8.1.1.6 or for any steel liners around the vent openings for the occupational exposure analysis. The calculated exposures for both configurations are due mainly to the expected gamma dose rate during preparation for welding. The increased calculated exposure for the 32PT-S100/32PT- L100 configuration is due to the thinner shield plug and due to draining the NUHOMS®-32PT DSC to meet a 32PT-S100/32PT-L100 DSC weight limit as described in Section M.8.

The NUHOMS®-32PT System loading operations, the number of workers required for each operation, and the amount of time required for each operation are presented in Table M.10-1 and Table M.10-2 for the 32PT-S125/32PT-L125 DSC and 32PT-S100/32PT-L100 DSC configurations respectively. This information is used as the basis for estimating the total occupational exposure associated with one fuel load. This evaluation is performed for the storage of one design-basis NUHOMS®-32PT DSC in an HSM. The dose rates applicable for each operation are based on the results presented in Section M.5.4 for loading operations. Engineering judgment and operational experience are used to estimate dose rates that were not explicitly evaluated. This evaluation assumes that a transfer trailer/skid with an integral ram is used for the DSC transfer operations. Licensees may elect to use different equipment and/or different procedures. Unique steps are sometimes necessary at the individual site to load the canister, complete closure operations and place the canister in the HSM. Specifically, the licensee may choose to modify the sequence of operations in order to achieve reduced dose rates for a larger number of steps, with the end result of reduced total exposure. The only requirement is that the licensee practice ALARA with respect to the total exposure received for a loading campaign. These estimated durations, manloading and dose rates are not limits.

The amount of time required to complete some operations is sometimes far greater than the actual amount of time spent in a radiation field. The process of vacuum drying the DSC includes setting up the vacuum drying system (VDS), verifying that the VDS is operating correctly, evacuating the DSC cavity, monitoring the DSC pressure, and disconnecting the VDS from the DSC. Of these tasks, only setup and removal of the VDS require a worker to spend time near the DSC. The most time consuming task, evacuating the DSC, does not require anyone to be present at all. The total exposure calculated for each task is therefore not necessarily equal to the

of the MCNP tallies to convert the tally results to fluxes (particles per second per square centimeter).

Gamma-ray spectrum calculations for the HSM are shown in Table M.10-4. The group fluxes on the HSM roof are taken from the ANISN run. The dose rate contribution from each group is the product of the flux and the flux-to-dose factor. The "Input Flux" column in Table M.10-4 is simply the roof flux in each group, divided by the total dose rate and represents the roof flux normalized to one mrem per hour. Similar calculations for neutrons are shown in Table M.10-5.

M.10.2.1 Activity Calculations

2x10 Back-to-Back Array

A box that envelops the HSM array and shield walls, as modeled in MCNP, approximates the 2x10 back-to-back array of HSMs. The dimensions of the box also include the width of the HSM end shield walls. As discussed above, the total activity of each face of the box is calculated by multiplying the flux per mrem/hr by the average dose rate of the face and by the area of the face.

Two 1x10 Front-to-Front Arrays

A box that envelops the HSM array and shield walls, as modeled in MCNP, approximates the two 1x10 arrays of HSMs. The dimensions of the box also include the width of the HSM end and back shield walls. As discussed above, the total activity of each face of the box is calculated by multiplying the flux per mrem/hr by the average dose rate of the face and by the area of the face.

The HSM surface activities are summarized in Table M.10-6.

M.10.2.2 Dose Rates

Dose rates are calculated for distances of 6.1 meters (20 feet) to 600 meters from the edges of the two ISFSI designs. The HSM is modeled in MCNP as a box, representing the HSM arrays.

The dose rate and annual dose results shown herein do not account for loading of 0.380 MTU fuel which is described in Section M.5.4.16. Loading 0.380 MTU fuel results in an increase in annual dose of 18%.

Neutron and gamma-ray sources are placed on each HSM, with shield walls, surface using the spectra and activities determined above. The angular distribution of source particles is modeled as a cosine distribution. The contribution of capture gamma-rays has been neglected, as has the contribution of bremsstrahlung electrons. The inclusion of coherent scattering greatly increases the variance in a problem with point detector tallies without improving the accuracy of the calculation. Thus, coherent scattering of photons is ignored.

The MCNP models of the two ISFSI layouts are described herein. For the 2x10 back-to-back array of HSMs with end shield walls the "box", dimensions are as follows. The total width is 1158 cm. The length of the "box" is 3220 cm and the height of the "box" is 457 cm.

For the two 1x10 front-to-front arrays of HSMs with end and back shield walls the "box", dimensions for each array are as follows. The total width is 640 cm. The length of the "box" is 3220 cm and the height of the "box" is 457 cm. The two 1x10 arrays are 1066 cm (35 feet) apart.

Table M.10-1
Occupational Exposure Summary
(32PT-S125/32PT-L125 DSC configuration)

Task	Number of Workers	Completion Time (hours)	Dose Rate (mrem/hr)	Exposure (mrem)
Location: Auxiliary Building and Fuel Pool				
Ready the DSC and Transfer Cask for Service(1)	2	4	0	-
Place the DSC into the Transfer Cask(1)	3	1	2	6
Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal	2	2	2	8
Fill the DSC Cavity with Water(2)	1	6	2	1
Install Lifting Yoke and Connect VDS	2	0.5	2	2
Place the Cask Containing the DSC in the Fuel Pool	5	0.5	2	5
Verify and Load the Candidate Fuel Assemblies into the DSC	3	8	2	48
Place the Top Shield Plug on the DSC	2	1	2	4
Raise the Cask/DSC to the Fuel Pool Surface	5	0.5	2	5
Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area	2	0.5	20	20
Location: Cask Decon Area				
Decontaminate the Outer Surface of the Cask (on the hook)(3)	3	1	varies	65
Cask Decontamination (in the decon area)(3)	3	1	varies	110
Remove the Cask/DSC Annulus Seal and Set-up Welder(3)	2	1.5	varies	88
Drain the DSC Cavity(3)	2	0.5	varies	33
Weld the Inner Top Cover to the DSC Shell and Perform NDE(3)	2	6	varies	71
Vacuum Dry and Backfill the DSC with Helium(3)	2	16	varies	33
Helium Leak Test the Shield Plug Weld	2	1	2	4
Seal Weld the Prefabricated Plugs to the Vent and Siphon Port and Perform NDE	1	1	241	241
Drain Cask/DSC Annulus(3)	1	0.25	varies	23
Install DSC Outer Top Cover Plate(3)	2	1	varies	209
Weld the Outer Top Cover Plate to DSC Shell and Perform NDE(3)	2	16	varies	112
Install the Cask Lid	2	1	15	30
Location: Reactor /Fuel Building Bay				
Ready the Cask Support Skid and Transfer Trailer for Service(1)	2	2	0	-
Place the Cask Onto the Skid and Secure(2)	3	0.5	225	225
Location: ISFSI Site				
Ready the HSM and Hydraulic Ram System for Service(1)	2	2	0	-
Transfer the Cask to the ISFSI(1)	6	1	0	-
Position the Cask in Close Proximity with the HSM(1)	3	1	0	-
Remove the Cask Lid	3	1	25	75
Align and Dock the Cask with the HSM	2	0.25	225	113
Position and Align Ram with Cask(3)	2	0.5	varies	28
Remove the RAM Access Cover Plate	2	0.25	121	61
Transfer the DSC from the Cask to the HSM(1)	3	0.5	0	-
Un-Dock the Cask from the HSM	2	0.083	752	125
Install the HSM Access Door	2	0.5	63	63
Total		**80**		**1,808**

Total estimated dose is 1.8 person-rem per canister load

Total estimated completion time is 80 hrs

(1) Performed away from any significant radiation sources.
(2) Personnel are not present throughout this activity.
(3) Dose rates and locations vary during this task.
Total estimated dose increases by 25% when loading 0.380 MTU/FA fuel.

Table M.10-2
Occupational Exposure Summary
(32PT-S100/32PT-L100 DSC configuration)

Task	Number of Workers	Completion Time (hours)	Dose Rate (mrem/hr)	Exposure (mrem)
Location: Auxiliary Building and Fuel Pool				
Ready the DSC and Transfer Cask for Service(1)	2	4	0	-
Place the DSC into the Transfer Cask(1)	3	1	2	6
Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal	2	2	2	8
Fill the DSC Cavity with Water(2)	1	6	2	1
Install Shield Plug and Connect VDS	2	0.5	2	2
Place the Cask Containing the DSC in the Fuel Pool	5	0.5	2	5
Verify and Load the Candidate Fuel Assemblies into the DSC	3	8	2	48
Place the Top Shield Plug on the DSC	2	1	2	4
Raise the Cask/DSC to the Fuel Pool Surface	5	0.5	2	5
Drain Water from DSC Cavity	1	1	100	100
Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area	2	0.5	20	20
Location: Cask Decon Area				
Decontaminate the Outer Surface of the Cask (on the hook)(3)	3	1	varies	680
Fill Cask Neutron Shield and DSC Cavity	1	0.1	57	6
Cask Decontamination (in the decon area)(3)	3	1	varies	112
Remove the Cask/DSC Annulus Seal and Set-up Welder(3)	2	1.5	varies	191
Drain the DSC Cavity(3)	2	0.5	varies	84
Weld the Inner Top Cover to the DSC Shell and Perform NDE(3)	2	6	varies	139
Vacuum Dry and Backfill the DSC with Helium(3)	2	16	varies	84
Helium Leak Test the Shield Plug Weld	2	1	2	4
Seal Weld the Prefabricated Plugs to the Vent and Siphon Port and Perform NDE	1	1	650	650
Drain Cask/DSC Annulus(3)	1	0.25	varies	58
Install DSC Outer Top Cover Plate(3)	2	1	varies	527
Weld the Outer Top Cover Plate to DSC Shell and Perform NDE(3)	2	16	varies	214
Install the Cask Lid	2	1	25	50
Location: Reactor /Fuel Building Bay				
Ready the Cask Support Skid and Transfer Trailer for Service(1)	2	2	0	-
Place the Cask Onto the Skid and Secure(2)	3	0.5	226	226
Location: ISFSI Site				
Ready the HSM and Hydraulic Ram System for Service(1)	2	2	0	-
Transfer the Cask to the ISFSI(1)	6	1	0	-
Position the Cask in Close Proximity with the HSM(1)	3	1	0	-
Remove the Cask Lid	3	1	27	81
Align and Dock the Cask with the HSM	2	0.25	226	113
Position and Align Ram with Cask(3)	2	0.5	varies	50
Remove the RAM Access Cover Plate	2	0.25	274	137
Transfer the DSC from the Cask to the HSM(1)	3	0.5	0	-
Un-Dock the Cask from the HSM	2	0.083	788	131
Install the HSM Access Door	2	0.5	96	96
Total		**81**		**3,831**

Total estimated dose is 3.8 person-rem per canister load
Total estimated completion time is 80 hrs

(1) Performed away from any significant radiation sources.
(2) Personnel are not present throughout this activity.
(3) Dose rates and locations vary during this task.
Total estimated dose increases by 25% when loading 0.380 MTU/FA fuel.

M.11.2 Postulated Accidents

M.11.2.1 Reduced HSM Air Inlet and Outlet Shielding

For the HSM-HS, this accident is not credible since the HSM-HS array is designed with the elimination of 6-inch gaps between the adjacent HSM-HS modules. The HSM-HS modules are placed next to each other and even in the unlikely event of large settlement of the ISFSI foundation, shifting of an adjacent HSM-HS occurring and causing the HSM-HS modules to separate is not credible.

This event is described in UFSAR Section 8.2.3 and Chapter U.2, Section U.2.2.3.

M.11.2.1.1 Cause of Accident

No change to Section 8.2.1.1.

M.11.2.1.2 Accident Analysis

There are no structural consequences that affect the safe operation of the NUHOMS®-32PT system resulting from the separation of the HSMs. The thermal effects of this accident results from the blockage of HSM air inlet and outlet openings on the HSM side walls in contact with each other. This would block the ventilation air flow provided to the HSMs in contact from these inlet and outlet openings. The increase in spacing between the HSM on the opposite side from 6 inches to 12 inches, will reduce the ventilation air flow resistance through the air inlet and outlet openings on these side walls, which will partially compensate the ventilation reduction from the blocked side. However, the effect on the NUHOMS®-32PT DSC, HSM and fuel temperatures is bounded by the complete blockage of air inlet and outlet openings described in Section M.11.2.7. The radiological consequences of this accident are described in the paragraph below.

M.11.2.1.3 Accident Dose Calculations

The off-site radiological effects that result from a partial loss of adjacent HSM shielding is an increase in the air scattered (skyshine) and direct doses from the 12 inch gap between the separated HSMs. The air scattered (skyshine) and direct doses are reduced from the gap between the HSMs that are in contact with each other. On-site radiological effects result from an increase in the direct radiation during recovery operations and increased skyshine radiation. Chapter 8, Table 8.2-2 shows the comparisons of the increased dose rate as a function of distance due to the reduced shielding effects of the adjacent HSM for the 24P DSC with 5-year cooled design basis fuel.

Table M.11-1 provides a similar table for 32PT-S100/32PT-L100 DSC with Configuration 2, from Appendix M.2, of the NUHOMS®-32PT System. For the NUHOMS®-32PT System, the dose received by a person located 100 meters away from the NUHOMS® installation for eight hours a day for five days (estimated recovery time) would be *less than 45* mrem. The increased dose to an

off-site person for 24 hours a day for five days located 2000 feet away would be *less than 0.25* mrem. Thus, the 10CFR72 requirements for this postulated event are met.

M.11.2.1.4 Corrective Actions

No change to Chapter 8, Section 8.2.1.4.

M.11.2.2 Earthquake

M.11.2.2.1 Cause of Accident

No change to Chapter 8, Section 8.2.3.1.

In addition, an alternate higher seismic design loading is postulated. The alternate higher seismic loading consists of an "enhanced" NRC Regulatory Guide 1.60 response spectra anchored at 1.0g maximum horizontal acceleration and 1.0g maximum vertical acceleration. The enhanced R.G. 1.60 response spectra is enriched in the frequency range above 9 Hz as shown in Appendix U, Figure U.2-4. This earthquake level is evaluated against Level D allowable criteria for those components evaluated in accordance with the ASME Code. There are no changes to the evaluations of the OS200 transfer cask and HSM-HS presented in Appendix U.

M.11.2.2.2 Accident Analysis

Chapter 8, Section 8.2.3.2 describes the analyses performed to demonstrate that the NUHOMS® System withstands the design basis seismic event. Section M.3.7.3 presents the changes to this analysis resulting from the addition of NUHOMS®-32PT DSC. As documented in Chapter 8 and Appendix U, the HSM/HSM-HS and the OS197/OS197H or OS200 TC have been evaluated for a payload that bounds the 32PT DSC payload, and thus these two components are not affected by the 32PT DSC. Therefore, only those analyses documented in Chapter 8 that are affected by the increased weight of the 32PT DSC are addressed in Section M.3.7.3. The results of this analysis show that seismic stresses are well below allowables and, thus, the leak-tight integrity of the canister is not compromised. The basket stresses are also low and do not result in deformations that would prevent fuel from being unloaded from the canister.

For the OS200 transfer cask and HSM/HSM-H, the assessment of earthquake, presented in Appendix U, Section U.11.2.2.2, is not changed.

M.11.2.2.3 Accident Dose Calculations

The design earthquake (including high seismic loading) does not damage the NUHOMS®-32PT system. Hence, no radioactivity is released and there is no associated dose increase due to this event.

Therefore, these stresses in a TC with a liquid neutron shield need not be evaluated for the accident condition.

As documented in Appendix U, Section U.3.3.7.4, the OS200 transfer cask has been evaluated for the NUHOMS®-32PTH1 DSC payload, which bounds that for the 32PT DSC.

For the OS200 transfer cask, the assessment of a complete loss of neutron shield, presented in Appendix U, Section U.11.2.5.2, is not changed.

M.11.2.5.3 Accident Dose Calculations for Loss of Neutron Shield

The postulated accident condition for the OS197/OS197H TC assumes that after a drop event, the water in the neutron shield is lost. The loss of neutron shield is modeled using the normal operation models described in Section M.5.4 by replacing the neutron shield with air. As discussed in Appendix M.5, the evaluation with the OS197/OS197H TC is bounding for the OS200 TC.

The accident condition dose rates for Configuration 2, from Chapter M.2, are summarized in Table M.11-2 and Figure M.11-1 for the bounding 32PT-L100 DSC loaded with design basis fuel plus BPRAs.

A comparison of the results in Table M.11-2 and Table M.5-3, demonstrates a maximum cask surface contact dose rate increase from *1178 mrem/hr (950 mrem/hr*1.24 scaling factor) to 5243 mrem/hr (4640 mrem/hr*1.13 scaling factor)*. These dose rates are *2.5* times those reported in Chapter 8, Section 8.2.5.3.2. Therefore, one would expect that the additional dose rate to an average on-site worker at an average distance of fifteen feet would also increase from 310 mrem/hr to *775* mrem/hr. Similarly the exposure to off-site individuals at a distance of 2000 feet would also be expected to increase from 0.04 mrem for an assumed eight hour exposure to *0.10* mrem. This exposure is still well within the limits of 10CFR72 for an accident condition. This corresponds to the exposure to an individual at a distance of 100 meters of approximately *48* mrem for the assumed eight hour duration*, which is* well within the limits of 10CFR72 for an accident condition.

M.11.2.5.4 Corrective Action

No change to Chapter 8, Section 8.2.5.4.

M.11.2.6 Lightning

No change. The evaluation presented in Chapter 8, Section 8.2.6 is not affected by the addition of the NUHOMS®-32PT DSC to the NUHOMS® System.

M.11.2.7 Blockage of Air Inlet and Outlet Openings

This accident conservatively postulates the complete blockage of the HSM ventilation air inlet and outlet openings on the HSM side walls.

Table M.11-1
Comparison of Total Dose Rates for HSM with and without Adjacent HSM Shielding Effects

Distance from Nearest HSM Wall, 2x10 Array (meters)	Normal Case Dose Rate[(1) (2)] (mrem/hr)	Accident Case Dose Rate[(1) (2)] (mrem/hr)
10	27	54
100	0.4	0.80
500	1.1×10^{-3}	4.4×10^{-3}
600	4.2×10^{-4}	1.7×10^{-3}

(1) Air scattered plus direct radiation

(2) *The dose rate results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section M.5.4.16. Loading 0.380 MTU fuel results in an increase in dose rates of 18%.*

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N.2.1 Spent Fuel to be Stored

There are two design configurations for the NUHOMS®-24PHB DSC: the 24PHBS and 24PHBL, which are nearly identical to the standard and long cavity 24P DSCs, respectively. Each of the DSC configurations is designed to store 24 intact PWR fuel assemblies, including reconstituted assemblies or up to 4 damaged and balance intact PWR fuel assemblies (including reconstituted) with characteristics described in Table N.2-1. The 24PHB DSC is designed to store intact or damaged B&W 15x15, intact WE 17x17, intact WE 15x15, intact CE 14x14, and intact WE 14x14 Class PWR fuel assemblies as specified in Table N.2.1. Control Components (CCs) and damaged fuel assemblies are allowed only in the B&W 15x15 class fuel assembly. Replacement assemblies by other manufacturers are also allowed provided they meet limiting features listed in Table N.2-1.

Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means and retrievability is ensured following normal and off-normal conditions. The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is ensured following normal and off-normal conditions. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to ensure retrievability.

The NUHOMS®-24PHB DSC may store PWR fuel assemblies arranged in one of two alternate Heat Load Zoning Configurations with a maximum decay heat of 1.3 kW per assembly and a maximum heat load of 24 kW per DSC. The Heat Load Zoning Configurations are shown in Figure N.2-1 and Figure N.2-2. The NUHOMS®-24PHB DSC is vacuum dried and backfilled with helium at the time of loading. The maximum (bounding) fuel assembly weight of 1682 lbs with a CC is identical to the NUHOMS®-24P DSC design.

The maximum fuel cladding temperature limit of 400°C (752°F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the 24PHB DSC per the guidance provided in NUREG-1536 [2.1]. In addition, NUREG-1536 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65°C (117°F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570°C (1058°F) is applicable to accidents or off-normal thermal transients [2.1].

The information provided in Table N.2-1 is based on the design basis B&W 15x15 fuel which is the bounding fuel assembly. The types of spent fuel considered in Appendix N include the following:

- B&W 15x15 Mark B2, B3, B4, B4Z, BZ, B5, B5Z, B6, B7, B8, B9, B10, B10D, B10E, B10F, B10G, B10L, B11, and B11A fuel assemblies, with or without CCs.
- B&W 15x15 reconstituted fuel assemblies with a maximum of 10 stainless steel rods per assembly or unlimited number of lower enrichment UO_2 rods instead of zirconium-alloy clad enriched UO_2 rods. The stainless steel rods are assumed to have two thirds the irradiation

time as the zirconium-alloy rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four.

- Control Components (CCs) with thermal and radiological characteristics as listed in Table N.2-2a are authorized for storage in the 24PHBL DSC. The CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources. Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs
- Intact WE 17x17, WE 15x15, CE 14x14 and WE 14x14 fuel assemblies, all without CCs.

The NUHOMS®-24PHB DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiation and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding and criticality evaluations for these rods are bounded by those of the regular rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Calculations are performed to determine the fuel assembly type which is most limiting for each of the analyses including shielding, criticality, heat load and confinement. Analyses performed demonstrated that limiting features associated with the design basis B&W 15x15 fuel assembly are bounding for assembly types listed in Table N.2-1.

N.2.1.1 General Operating Functions

No change to Chapter 3, Section 3.1.2.

Table N.2-1

The detailed information associated with this table can be found in CoC 1004 Amendment 15 Technical Specifications Table 1-1i.

N.3.6.2.3 Damaged Fuel Cladding Structural Evaluation for Normal and Off-Normal Loads

The damaged fuel structural evaluations as presented in Appendix Z, Section Z.3.6.3 are applicable for the fuel types allowed in the 24PHB DSC. *The structural analysis documented in this section conservatively evaluates a limiting configuration with a single rod and the spacer grids in designated locations without any support from the fuel compartment to provide assurance of limiting additional cladding damage. The changes to the fuel assembly configuration do not have impact on retrievability due to damage to spacer grids as long as the assembly is able to be handled by normal means and the retrievability is ensured following the normal and off-normal conditions. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to ensure retrievability. The criticality analysis documented in Section N.6.4 also considers the impact of damage to the fuel assembly that includes missing and damaged grid spacers, which results in limiting the enrichment of these fuel assemblies. Therefore, additional configurations for damaged grid spacers are not evaluated herein. Licensees can perform specific evaluations to demonstrate retrievability using actual configurations.* These evaluations demonstrate that large margins of safety are available in the cladding material for normal and off-normal loads, and, therefore, retrievability of the damaged fuel assemblies is assured.

check this identification number against the DSC loading plan which indicates which fuel assemblies (and CCs, if applicable) are acceptable for dry storage.

7. Position the fuel assembly for insertion into the selected DSC storage cell and load the fuel assembly. Repeat step 6 for each SFA loaded into the DSC. If loading damaged fuel assemblies, place top end caps over each damaged fuel assembly placed in the basket. A maximum of 4 damaged fuel assemblies may be loaded into the basket per Technical Specification 2.1. After the DSC has been fully loaded, check and record the *identity* and location of each fuel assembly and CCs, if applicable, in the DSC.

8. After all the SFAs (and CCs, if applicable) have been placed into the DSC and their identities verified, position the lifting yoke and the top shield plug and lower the shield plug onto the DSC.

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HSM-H/HSM-HS in either the OS197/OS197H/OS200 or OS197FC/OS200FC TC depending upon the heat load.

The 24PTH-S-LC DSC is a modified version of 24PTH-S DSC, provided with thinner top and bottom lead shield plugs instead of steel, resulting in a longer cavity length. This DSC type is designed for a maximum heat load of 24 kW per DSC and may be stored in either the currently licensed Standardized HSM Model 102, or in the new HSM-H, while the currently licensed Standardized TC (with a solid neutron shield) *or OS197/OS197H TC (with a liquid neutron shield) are* used for onsite transfer.

Fuel assemblies with CCs are to be stored in 24PTH-S, 24PTH-L and 24PTH-S-LC DSC Types.

These three alternate NUHOMS®-24PTH System configurations are summarized below:

System Configuration	24PTH DSC Type(1)	Basket Type	Max. Heat Load (kW) per DSC	Transfer Cask	Storage Module
1	24PTH-S or 24PTH-L	1A, 1B, or 1C(2)	40.8	OS197FC or OS200FC	HSM-H or HSM-HS
			31.2	OS197/OS197H or OS200	HSM-H or HSM-HS
2	24PTH-S or 24PTH-L	2A, 2B, or 2C(3)	31.2	OS197FC/*OS200FC*	HSM-H or HSM-HS
3	24PTH-S-LC	2A, 2B, or 2C(3)	24.0	Standardized TC (solid neutron shield)/*OS197/OS197H*	HSM (Model 102 or 202) or HSM-H or HSM-HS

(1) Allows storage of Control Components
(2) With heat conductive aluminum inserts in the R45 basket transition rail
(3) With no heat conductive aluminum inserts in the R45 basket transition rail

The NUHOMS®-24PTH system provides structural integrity, confinement, shielding, criticality control and passive heat removal independent of any other facility structures or components.

The format of this Appendix follows the guidance provided in NRC Regulatory Guide 3.61 [1.1]. The analysis presented in this Appendix shows that the NUHOMS®-24PTH system meets all the requirements of 10CFR72 [1.2]. A separate analysis will be submitted to address the safety related aspects of transporting spent fuel in the NUHOMS®-24PTH DSC in accordance with 10CFR71 [1.3].

Several sections of this Appendix have been identified as "No change". For these sections, the description or analysis presented in the corresponding sections of the UFSAR for the Standardized NUHOMS® system is also applicable to the 24PTH system. In addition, Tables and Figures presented in the UFSAR which remain unchanged due to the addition of the 24PTH system to the Standardized NUHOMS® system are not repeated in this Appendix.

Note: References to sections or chapters within this Appendix are identified with a prefix P (e.g., Section P.2.3 or Appendix P.2 or Chapter P.2). References to sections or chapters of the UFSAR outside of this Appendix (main body of the UFSAR) are identified with the applicable UFSAR section or chapter number (e.g., Section 2.3 or Chapter 2). The references used in this appendix are identified as [X.X] (e.g., [1.1] is reference 1.1 at the end of Section P.1).

P.1.1 Introduction

The NUHOMS®-24PTH system is designed to store up to 24 intact (including reconstituted) B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14, and WE 14x14 class PWR fuel assemblies. WE 15x15 Partial Length Shield Assemblies (PLSAs) are also authorized to be stored in the 24PTH system. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. %, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of *2.0* years. The 24PTH-S, 24PTH-L and 24PTH-S-LC DSC types are also designed to store up to 24 Control Components (CCs) which include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources. Furthermore, materials that are positioned or operated within the envelope of the fuel assembly during reactor operation are also considered as CCs. The design characteristics, including physical and radiological parameters of the payload, are described in Appendix P.2.

Reconstituted assemblies containing up to 10 replacement stainless steel rods per assembly or unlimited number of lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods are acceptable for storage in 24PTH DSC as intact fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four.

Provisions have been made for storage of up to 12 damaged fuel assemblies in lieu of an equal number of intact assemblies in cells located at the outer edge of the 24PTH basket. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater *than* hairline cracks or pinhole leaks. *The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means and the retrievability is ensured following normal and off-normal conditions.* The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

Provisions have also been made for storage of up to 8 failed fuel assemblies in lieu of an equal number of intact and/or damaged assemblies in cells located at the outer edge of the 24PTH basket as described in Appendix P.2.

The NUHOMS®-24PTH system consists of the following new or modified components:

- A 24PTH DSC, with three alternate configurations, described in detail in Section P.1.2, provides confinement, an inert environment, structural support, and criticality control for the 24 PWR fuel assemblies,
- An HSM-H module, described in Section P.1.2, or an HSM-HS module (described in Appendix U.1, Section U.1.2) is provided for environmental protection, shielding and heat rejection during storage, and
- OS197-FC or OS200FC transfer cask for onsite transfer of the 24PTH-S and 24PTH-L DSCs. The NUHOMS®-24PTH-S and 24PTH-L DSCs with Types 1A, 1B, 1C baskets can also be transferred in the OS197 or OS197H or OS200 TCs if the total heat load is 31.2 kW or less.

In addition to these new or modified components listed above, the 24PTH-S-LC DSC requires the use of the existing Standardized HSM Model 102 or the new HSM-H for storage and the Standardized Transfer Cask *or OS197/OS197H TC* for transfer.

The NUHOMS®-24PTH system requires the use of non-safety related auxiliary transfer equipment described in Chapter 1, Section 1.3.2.2 of the UFSAR. There is no change to any of these items except for the cask support skid. The cask support skid is modified by adding two industrial grade motor driven redundant blowers with associated ductwork for connecting to the TC ram cover plate opening. This modification provides a reliable source of external air circulation for the OS197FC TC.

Approval of the NUHOMS®-24PTH system components described in Section P.1.2 is sought under the provisions of 10CFR 72, Subpart L for use under the general license provisions of 10CFR 72, Subpart K. The 24PTH system components are intended for storage on a reinforced concrete pad.

P.2.1 Spent Fuel To Be Stored

As described in Appendix P.1, there are three design configurations for the NUHOMS®-24PTH DSC; S, L and S-LC. Each of the DSC configurations is designed to store intact (including reconstituted) and/or damaged and/or failed PWR fuel assemblies as specified in Table P.2-1 and Table P.2-3. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235. The maximum allowable assembly average burnup is limited to 62 GWd/MTU. The minimum required cool time for fuel to be stored with *380, 475, and 492* kgU/FA is explicitly specified as a function of burnup and enrichment in *Technical Specifications Table 1-3a through 1-3p*. For fuel with a kgU/FA loading between these values, the minimum required cool time for fuel to be stored as a function of burnup and enrichment is determined by using the interpolation methodology specified in the notes and examples following *Technical Specifications Table 1-3p.*

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The 24PTH-S, 24PTH-L and 24PTH-S-LC DSCs are also designed to store control components (CCs) with thermal and radiological characteristics as listed in Table P.2-2. The CCs include burnable poison rod assemblies (BPRAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs) , neutron source assemblies (NSAs), and neutron sources. Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as guide tube or instrument tube tie rods or anchors, guide tube inserts, BPRA spacer plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.

Partial length shield assemblies (PLSAs) for the Westinghouse 15x15 class, where part of the active fuel is replaced with steel are also included as authorized.

The NUHOMS®-24PTH DSC is also authorized to store fuel assemblies containing blended low enriched uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four.

The NUHOMS®-24PTH DSCs can also accommodate up to a maximum of 12 damaged fuel assemblies placed in cells located at the outer edge of the DSC as shown in Figure P.2-6.

Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater *than* hairline cracks, or pinhole leaks. *The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means.* The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is assured following normal and off-normal conditions. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

The NUHOMS®-24PTH*F* DSC, an alternative version of the NUHOMS®-24PTH DSC, is designed to accommodate up to a maximum of 8 failed fuel assemblies encapsulated in individual failed fuel cans and placed in cells located at the outer edge of the DSC as shown in Figure P.2-6. Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Failed fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means.

Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a rod storage basket are also considered as failed fuel. Loose fuel debris, not contained in a rod storage basket may also be placed in a failed fuel can for storage, provided the size of the debris is larger than the failed fuel can screen mesh opening and it is located at least 10″ above the top of the bottom shield plug of the DSC.

Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its contents shall be less than 1682 lbs.

A 24PTH DSC containing less than 24 fuel assemblies may contain either empty slots or dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly.

The NUHOMS®-24PTH-S and 24PTH-L DSCs may store up to 24 PWR fuel assemblies arranged in any of the four alternate heat load zoning configurations shown in Figure P.2-1 through Figure P.2-4 with a maximum decay heat of 2.0 kW per assembly and a maximum heat load of 40.8 kW per canister.

The 24PTH-S-LC may store up to 24 B&W 15x15 fuel assemblies arranged in accordance with heat load zoning configuration No. 5 with a maximum decay heat of 1.5 kW per assembly and a maximum heat load of 24.0 kW per DSC, as shown in Figure P.2-5.

The 24PTH DSC basket is designed with 2 alternate options: Type 1 basket, which includes aluminum inserts in the R45 transition rails, and Type 2 basket which does not include any aluminum inserts. Type 1 basket is the preferred option for canisters with high decay heat loads, since the aluminum inserts allow a more direct heat conduction path from the basket edge to the DSC shell. Type 2 basket offers the advantage of an adequate thermal performance but with a lower lifting weight requirement.

The NUHOMS®-24PTH DSC basket is designed with three alternate poison materials: Borated Aluminum alloy, Boron Carbide/Aluminum Metal Matrix Composite (MMC) and Boral®. For criticality analysis, 90% of B-10 content present in the borated aluminum and MMC poison plates is credited, while only 75% is credited for Boral®.

For each poison material, the NUHOMS®-24PTH DSC basket is analyzed for six alternate basket configurations, depending on the boron loadings analyzed (designated as "A" basket for low B-10 loading, "B" basket for moderate B-10 loading, and "C" basket for high B-10 loading) and Basket-Type (Type 1 or Type 2).

A summary of the alternate poison loadings considered and the corresponding credit taken in the criticality analysis for each poison material as a function of basket types is presented below:

Poison Type	24PTH Basket Type(1)	Poison Loading (B-10 mg/cm^2)	% Credit Used in Criticality Analysis
Borated Aluminum Alloy/MMC	1A or 2A	7	90
	1B or 2B	15	
	1C or 2C	32	
Boral®	1A or 2A	9	75
	1B or 2B	19	
	1C or 2C	40	

(1) Type 1A = Basket Type 1 with aluminum inserts in the R45 transition rails and Type A poison plate configuration; Type 2A = Basket Type 2 without aluminum inserts in the R45 transition rails and Type A poison plate configuration;

Table P.2-4 summarizes the maximum assembly average initial enrichment as a function of soluble boron concentration and basket neutron poison requirements for intact fuel assemblies. Table P.2-5 summarizes the maximum assembly average initial enrichment as a function of soluble boron concentration and basket neutron poison requirements for up to a maximum of 12 damaged fuel assemblies. Table P.2-5a summarizes the maximum assembly average initial enrichment as a function of soluble boron concentration and basket neutron poison requirements for up to a maximum of 8 damaged and/or failed fuel assemblies.

The method for determining the minimum required cooling times for the fuel assemblies with heavy metal loads between *380* and *492* kgU/FA is provided in Chapter 7, Section 7.2.3.2.

The NUHOMS®-24PTH DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight with a CC is 1682 lbs.

The maximum fuel cladding temperature limit of 400 °C (752 °F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-24PTH DSC per NUREG-1536 [2.1]. In addition, NUREG-1536 [2.1] does not permit thermal cycling of the fuel cladding with temperature differences greater than 65 °C (117 °F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570 °C (1058 °F) is applicable to accidents or off-normal thermal transients [2.1].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, thermal and confinement. These evaluations are performed in Chapter P.5, P.6, P.4 and P.7, respectively. The fuel assembly classes considered are listed in Table P.2-3. It was determined that the B&W 15x15 *may be used as a representative fuel assembly for dose rate calculations.* For criticality safety, the B&W 15x15 assembly is the most reactive assembly type for a given enrichment. This assembly is used to determine the most reactive configuration in the DSC. Using this most reactive configuration, criticality analysis for all other fuel assembly classes is performed to determine the maximum enrichment allowed as a function of the soluble boron concentration and fixed poison plate loading. For thermal analysis, the WE 14x14 fuel assembly is limiting for the 24PTH-S and -L DSCs, and B&W 15x15 fuel assembly for the 24PTH-S-LC DSC since they result in the lowest fuel conductivity. The confinement analysis is based on B&W 15x15 fuel assembly, since it results in a smaller free volume inside the DSC cavity as compared to a 14x14 fuel assembly.

For calculating the maximum internal pressure in the NUHOMS®-24PTH DSC, it is assumed that 1% of the fuel rods are damaged for normal conditions, up to 10% of the fuel rods are damaged for off normal conditions, and 100% of the fuel rods will be damaged following a design basis accident event. A minimum of 100% of the fill gas and 30% of the fission gases within the ruptured fuel rods are assumed to be available for release into the DSC cavity, consistent with NUREG-1536 [2.1].

P.2.5 Summary of NUHOMS®-24PTH DSC and HSM-H Design Criteria

P.2.5.1 24PTH DSC Design Criteria

The principal design criteria for the NUHOMS®-24PTH DSC are presented in Table P.2-18. The NUHOMS®-24PTH DSC is designed to store intact and/or damaged and/or failed PWR fuel assemblies with or without CCs with assembly average burnup, initial enrichment and cooling time as described in Table P.2-1 and Table P.2-3. The maximum total heat generation rate of the stored fuel is limited to *2.5* kW per fuel assembly (1.5 kW for 24PTH-S-LC DSC) and 40.8 kW per canister (24.0 kW for 24PTH-S-LC DSC) in order to keep the maximum fuel cladding temperature below the limit [2.5] necessary to ensure cladding integrity. The fuel cladding integrity is assured by the NUHOMS®-24PTH DSC and basket design which limits fuel cladding temperature and maintains a nonoxidizing environment in the DSC cavity as described in Appendix P.4.

The NUHOMS®-24PTH DSC (shell and closure) is designed and fabricated as a Class 1 component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [2.2], and the alternative provisions to the ASME Code as described in Table P.3.1-1.

The NUHOMS®-24PTH DSC is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. A combination of fixed neutron absorbers, soluble boron in the pool and favorable geometry are employed to maintain the upper subcritical limit of 0.9411. The fixed neutron absorbers are in the form of borated aluminum metallic plates or Boral®. The basket is designed and fabricated in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG-3200 [2.2] and the alternative provisions to the ASME Code as described in Table P.3.1-2.

The NUHOMS®-24PTH DSC design, fabrication and testing are covered by AREVA TN America's Quality Assurance Program, which conforms to the criteria in Subpart G of 10 CFR Part 72.

The NUHOMS®-24PTH DSC is designed to withstand the effects of severe environmental conditions and natural phenomena such as earthquakes, tornadoes, lightning and floods. Appendix P.11 describes the NUHOMS®-24PTH DSC behavior under these accident conditions.

P.2.5.2 HSM-H Design Criteria

The principal design criteria for the NUHOMS® HSM-H module and steel support structure are presented in Table P.2-18. The load combination and design criteria for concrete and support structure components are the same as those described in Chapter 3, Section 3.2.5.1. These criteria, provided in Chapter 3, Tables 3.2-4, 3.2-5, 3.2-8 and 3.2-10 are also applicable to the HSM-H design.

Table P.2-1

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1l.

Table P.2-3

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1m.

Table P.2-4

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1p

Table P.2-5

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1q.

Table P.2-5a

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1q1.

Table P.2-6
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Table P.2-7
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Table P.2-8
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Table P.2-10
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Table P.2-11
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Table P.2-12

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Table P.2-13
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Notes: Tables P.2-6 through P.2-13
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Table P.2-19
Design Pressures for Tornado Wind Loading

HSM-H Wall Orientation[(1)]	Velocity Pressure (psf)	Pressure Coefficient[(2)]	Max/Min Design Pressure (psf)
Front	344	+0.68	234
Left	344	-0.60	-207
Rear	344	-0.43	-148
Right	344	-0.60	-207
Roof	344	-0.60	-207

Notes:
(1) Wind direction assumed to be from front. Wind loads from other directions may be found by rotating table values to desired wind direction.
(2) Pressure coefficient (used) = Gust factor (0.85)* Max/Min pressure coefficient.

Table P.2-20
B10 Specification for the NUHOMS®-24PTH Poison Plates

NUHOMS®-24PTH DSC Basket Type[(1)]	Minimum B10 Areal Density, (grams/cm^2)	
	Borated Aluminum or MMC	Boral®
1A or 2A	0.007	0.009
1B or 2B	0.015	0.019
1C or 2C	0.032	0.040

(1) Basket Type 1 contains aluminum inserts in the R45 transition rails of the basket, Type 2 does not contain aluminum inserts.

Table P.2-21
Maximum Allowable Heat Load for the NUHOMS®-24PTH DSC

24PTH DSC Type[(1)]	Basket Type [(2)(3)]	Transfer Cask	Max. Heat Load (kW) per DSC
24PTH-S or 24PTH-L	1A, 1B, or 1C	OS197FC or OS200FC	40.8
		OS197 or OS197H or OS200FC	31.2
24PTH-S or 24PTH-L	2A, 2B, or 2C	OS197FC or OS200FC	31.2
24PTH-S-LC	2A, 2B, or 2C	Standardized TC (solid neutron shield)/*OS197/OS197H*	24.0

Notes:
(1) Allows storage of control components.
(2) Basket Type 1 (1A, 1B, 1C) has heat conductive aluminum inserts in the R45 basket transition rails.
(3) Basket Type 2 (2A, 2B, 2C) does not have heat conductive aluminum inserts in the R45 basket transition rails.

Table P.2-22
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Table P.2-23
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Table P.2-24
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The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-15a.

Figure P.2-9

Table P.3.1-1

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, Alternatives to the ASME Code for the NUHOMS-24PTH DSC Confinement Boundary.

Table P.3.1-2

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, Alternatives to the ASME Code for the NUHOMS-24PTH DSC Basket Assembly.

As discussed in Section P.3.6.1.5 the evaluations for non-thermal loads are not affected, with exception of the cask lid stresses, which are increased by 2.5%. This increase in primary stresses has been considered in the summary results included in Table P.3.6-13.

P.3.6.3 Damaged Fuel Integrity Assessment for Normal and Off-Normal Loads

Per the definition in Table P.2-1, damaged PWR fuel assemblies are fuel assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks, *or assemblies with damage to spacer grids, as long as the assembly is able to be handled by normal means*. The extent of cladding damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is assured following Normal/Off-Normal conditions.

This section summarizes the evaluations performed to demonstrate structural integrity of the damaged fuel under normal and off-normal operations loads. The evaluations consider the effects of cladding defect size, cladding rupture geometry, and reduced cladding thickness due to oxidation effects.

The structural analysis documented in this section conservatively evaluates configurations with a single rod and the spacer grids in designated locations without any support from fuel compartment to provide assurance of limiting additional cladding damage. The changes to the fuel assembly configuration do not have impact on retrievability due to damage to the spacer grids, as long as the assembly is able to be handled by normal means and the retrievability is ensured following the normal and off-normal conditions. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to ensure retrievability. The criticality analysis documented in Section P.6.4 also considers the impact of damage to the fuel assembly that includes missing and damaged grid spacers, which results in limiting the enrichment of these fuel assemblies. Therefore, additional configurations are not evaluated herein. Licensees can perform specific evaluations to demonstrate retrievability using actual configurations.

Normal operation loads for storage conditions include stresses due to dead weight, thermal, and handling loads resulting from DSC fuel loading/unloading, fuel transfer to the ISFSI, and DSC insertion to and retrieval from the HSM. These handling/transfer operations are performed slowly by trained operations personnel and follow detailed procedures. The applicable off-normal load for storage conditions is the off-normal handling load (i.e. jammed canister condition). Because the 24PTH DSC is a dual-purpose canister, it has been evaluated for loads that bound the Part 72 normal and off-normal storage conditions such as those defined in the NUHOMS® MP197 Transport SAR [3.33].

Both linear-elastic stress analysis and linear elastic fracture mechanics methods are employed to evaluate the integrity of the fuel cladding. Table P.3.6-15 shows a summary of the PWR fuel assemblies design parameters used in these evaluations. A cladding thickness reduction of 120 μm has been assumed in the structural integrity evaluations to account for waterside and inner surface oxidation.

The linear elastic stress analyses use basic stress equations, conservation of energy principles, and fundamental kinematic relationships to calculate cladding stresses due to normal and off-normal loads. The stress evaluations conservatively consider the full weight of the fuel pellets in the determination of cladding stresses. The handling/transfer loads produce the controlling stresses from normal and off-normal operation loads. The computed maximum stresses for the controlling loads are summarized in Table P.3.6-16. The computed maximum stresses are compared to the irradiated cladding yield stress, and a stress ratio is calculated. As shown in the table, the maximum stress ratios correspond to the hypothetical one-foot end and side drops. Substantial margins exist for all loads considered. All the stresses summarized in Table P.3.6-16 are compressive stresses with the exception of the one-foot side drop case, which produces tension stresses due to bending. Tension stresses are evaluated using fracture mechanics principles, as described below. The maximum compressive load obtained from all analyzed load cases is significantly lower than the calculated buckling capacity for the bounding fuel. Thus, stability of the fuel tube cladding is maintained during normal and off-normal loads.

P.4 Thermal Evaluation

P.4.1 Discussion

This chapter presents the thermal evaluations which demonstrate that the NUHOMS®-24PTH System meets the thermal requirements of 10CFR72 for the dry storage of spent fuel. The NUHOMS®-24PTH System is designed to passively reject decay heat during storage and transfer for normal, off-normal and accident conditions while maintaining temperatures and pressures within specified regulatory limits.

Several thermal design criteria are established for the thermal analysis of the 24PTH DSC basket as discussed below.

Maximum temperatures of the confinement structural components must not adversely affect the confinement function,

Maximum fuel cladding temperature limit of 400°C (752°F) is applicable to normal conditions of storage and all short term operations including vacuum drying and helium backfilling of the 24PTH DSC per Interim Staff Guidance (ISG) No. 11, Revision 3 [4.19]. In addition, ISG-11 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65°C (117°F) during drying and backfilling operations,

Maximum fuel cladding temperature limit of 570°C (1058°F) is applicable to accidents or off-normal thermal transients [4.19],

The maximum DSC cavity internal pressures during normal, off-normal and accident conditions must be below the design pressures of 15 psig, 20 psig and 120 psig, respectively, and

A total of *six* (*6*) Heat Load Zoning Configurations (HLZCs) are allowed for the 24PTH DSCs as shown in Figures P.2-1 through P.2-5*, and Figure P.2-9*. The maximum total heat load per DSC is 40.8 kW, 31.2 kW or 24 kW depending upon the specific DSC types and HLZCs. The thermal analysis is carried out for HLZC 1 shown in Figure P.4-36 with 40.8 kW heat load because it bounds the HLZCs 2 and 3. In HLZC 1, the fuel assemblies at the center 4 locations in the DSC basket have a maximum heat load of 1.7 kW per assembly as compared to zero (empty locations) for HLZC 2 and 1.5 kW for HLZC 3. Therefore, HLZC 1 results in higher fuel cladding and basket temperatures compared to HLZCs 2 or 3. To bound all possible combinations of heat loads from fuel assemblies allowed in HLZC 5 of Figure P.2-5, the thermal analysis is carried out for the HLZC shown in Figure P.4-38. In this thermal analysis configuration, the highest possible allowed decay heat assemblies are assumed to be at the center and upper half of the basket locations resulting in bounding fuel cladding and basket temperatures.

Additional HLZC restrictions are considered for storage of the 24PTH DSC with damaged/failed fuel assemblies in Section P.4.6.9.

Section P.4.10 discusses the thermal analysis of the 24PTH Type 1 DSC with HLZC #6 as shown in Appendix P.2, Figure P.2-9. As shown in Section P.4.10, the thermal analysis of HLZC #6 is bounded by the thermal analysis of HLZC #1 presented in Section P.4.6.

The thermal analysis is carried out for the three NUHOMS®-24PTH DSC configurations (24PTH-S, 24PTH-L, and 24PTH-S-LC DSC types in combination with six basket types of the NUHOMS®-24PTH system described in Section P.2.1). A summary of the three system configurations analyzed in this chapter are summarized below:

System Configuration	DSC Type	Aluminum Inserts in Basket	Fuel Type	Total Heat Load per DSC, kW	Transfer Cask	Storage Module
1	24PTH-S or 24PTH-L	With inserts	All Fuels	40.8	OS197FC/ OS200FC	HSM-H/ HSM-HS
				31.2	OS197/ OS197H/OS200	HSM-H/ HSM-HS
2	24PTH-S or 24PTH-L	No inserts	All Fuels	31.2	OS197FC/ OS200FC	HSM-H/ HSM-HS
3	24PTH-S-LC(1)	No inserts	B&W 15x15	24	Standardized TC/ *OS197/OS197H*	HSM-H/HSM-HS or HSM Model 102 or 202

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(1) The maximum heat load allowed in the 24PTH-S-LC DSC is 24 kW. The HSM Model 102 is designed for a maximum heat load of 24 kW from a NUHOMS® 24P DSC as described in Section 8.1.3. Therefore no additional analysis of HSM Model 102 is required with 24PTH-S-LC DSC. *Thermal evaluation of 24PTH-S-LC DSC in OS197 TC is presented in Section P.4.11.* AMD 15

The thermal evaluations presented herein include steady state and transient analyses of the thermal response of the NUHOMS®-24PTH system components to a defined set of thermal loading conditions. These loading conditions envelop the thermal conditions expected during all normal, off-normal, and postulated accident loading, transfer and dry storage operations for the design basis thermal conditions as defined in Section P.2. The applicable allowable temperatures are presented and comparisons are made with calculated temperatures as the basis for acceptance. 72.48

The analyses conservatively apply a uniform maximum peaking factor of 1.11 [4.1] along the active fuel length to bound the effect of the decay heat flux varying axially along the active fuel length.

A description of the detailed analyses performed for the storage of NUHOMS®-24PTH DSC under normal, off-normal, and accident conditions is provided in Sections P.4.1 and for transfer is provided in Section P.4.5. Section P.4.6 describes the 24PTH DSC basket and fuel cladding analysis for storage and transfer conditions. Section P.4.6.8 describes thermal analysis of the OS200 TC with the 24PTH DSC and Section P.4.6.9 describes evaluation of the 24PTH DSC with damaged/failed fuel assemblies (FAs). The DSC cavity internal pressures are also calculated in Section P.4.6 for all conditions of storage and transfer. Section P.4.7 describes the evaluation performed for loading/unloading conditions. The thermal evaluation concludes that each of the three NUHOMS®-24PTH systems configurations listed above meets all the design criteria.

The effective thermal conductivity of the fuel assemblies used in the 24PTH DSC thermal analysis is based on the conservative assumption of radiation and conduction heat transfer only, where any convection heat transfer is neglected. In addition, the lowest effective thermal conductivity among the fuel assemblies to be stored using 24PTH-S DSC, -L DSC, and -S-LC DSC is selected as the basis for the thermal analysis. Section P.4.8 presents the calculations that determined the fuel assembly effective thermal conductivity in a helium or vacuum environment. The thermal analysis model conservatively neglects convection heat transfer in the basket regions.

The DSC basket and fuel cladding temperature calculation methodology has been benchmarked [4.20] against experimental data [4.21] obtained for the TN-24 cask.

P.4.6 NUHOMS®-24PTH DSC Basket Thermal Analysis

The thermal analysis of the NUHOMS® 24PTH DSC is based on finite element models developed using the ANSYS computer code [4.6]. ANSYS is a comprehensive thermal, structural and fluid flow analysis package. It is a finite element analysis code capable of solving steady state and transient thermal analysis problems in one, two or three dimensions. Heat transfer via a combination of conduction, radiation and convection can be modeled by ANSYS. Solid entities are modeled by SOLID70 elements for 3-D.

A total of *six* (*6*) Heat Load Zoning Configurations (HLZCs) are allowed for the 24PTH DSCs as shown in Figures P.2-1 through P.2-5*, and Figure P.2-9*. The maximum total heat load per DSC is 40.8 kW, 31.2 kW or 24 kW depending upon the specific DSC types and HLZCs. The thermal analysis is carried out for HLZCs shown in Figure P.4-36 for the 40.8 kW heat load because it bounds the HLZCs 2 and 3. In HLZC 1, the fuel assemblies at the center 4 locations in the DSC basket have a maximum heat load of 1.7 kW per assembly as compared to zero (empty locations) for HLZC 2 and 1.5 kW for HLZC 3. Therefore, HLZC 1 results in higher fuel cladding and basket temperatures compared to HLZCs 2 or 3. Thermal analysis is carried out for HLZC 4 shown in Figure P.4-37 for 31.2 kW total DSC heat load. To bound all possible combinations of heat loads from fuel assemblies allowed in HLZC 5 of Figure P.2-5, the thermal analysis is carried out for the HLZC shown in Figure P.4-38. In this thermal analysis configuration, the highest possible allowed decay heat assemblies are assumed to be at the center and upper half of the basket locations resulting in bounding fuel cladding and basket temperatures.

This section discusses the thermal analysis of the 24PTH DSC with HLZCs #1 through #5. Section P.4.10 discusses the thermal analysis of the 24PTH Type 1 DSC with HLZC #6 as shown in Appendix P.2, Figure P.2-9. As shown in Section P.4.10, the thermal analysis of HLZC #6 is also bounded by the thermal analysis of HLZC #1 presented in this section.

P.4.6.1 NUHOMS® 24PTH DSC Basket and Payload Model

The three-dimensional model (Figure P.4-39) represents the NUHOMS®-24PTH DSC and includes the geometry and material properties of the basket components, the basket rails, and DSC shell. The 24PTH DSC basket components are shown in Figure P.4-40. Each component of the basket (fuel compartment tubes, poison plates, aluminum plates, R90 rails, R45 rail, and inserts) is modeled individually with SOLID70 elements. The gaps between adjacent basket components are also represented with SOLID70 elements with helium or air conductivity as appropriate. The material properties from Section P.4.1 are used for the fuel region. Within the model, heat is transferred via conduction through fuel regions, fuel compartment tubes, aluminum and poison plates, and the gas gaps between all aluminum, poison and steel members. Generally, good surface contact is expected between adjacent components within the basket structure. However, to bound the heat conductance uncertainty between adjacent components owing to imperfect contact between the neutron poison material, aluminum and steel basket components, uniform gaps along the entire surfaces are assumed. This is a conservative assumption, because although there will be imperfect contact between the adjacent plates, they will be in contact with each other at most of the locations. Therefore, thermal resistance to heat flow from the fuel assembly out to the DSC surface is lower with the actual imperfect contact as compared with the modeled uniform gaps along the entire surfaces. The gaps used in the thermal analysis of the 24PTH DSC are shown in the Figure P.4-50.

The 3D models are longitudinally full-length, one-half (180°) cross section (right half) models of the 24PTH DSC. The ANSYS models comprise the DSC shell assembly (including the shell, and top and bottom end components) and the basket assembly (including basket fuel compartment tubes, aluminum and poison plates, and the aluminum R90 and steel R45 transition rails).

A summary of the maximum accident operating pressures for the various 24PTH DSC configurations are presented in Table P.4-29.

P.4.6.7.6 Evaluation of the 24PTH DSC Performance During Accident Conditions

The NUHOMS®-24PTH DSC shell and basket are evaluated for the accident conditions pressures and temperatures as described in Section P.3.

The maximum fuel cladding temperature of 914°F is below the short-term limit of 1058°F (570°C). The accident pressure in the NUHOMS®-24PTH-S or -L DSC of 97.2 psig remains below the accident design pressure of 120 psig. The accident pressure in the 24PTH-S-LC DSC of *85.8* psig is also below the accident design pressure of 90 psig. It is concluded that the NUHOMS®-24PTH system maintains confinement during the postulated accident condition. AMD 15

P.4.6.8 Thermal Analysis of OS200 TC with 24PTH DSC

The OS200 TC model with the 24PTH DSC is identical to the model described in Appendix T, Section T.4.5.5.1 with the following exceptions.

The dimensions of the DSC are modified to reflect the dimensions of the 24PTH DSC. The shortest cavity length based on the 24PTH-S DSC is selected to envelop the maximum heat generation. The dimensions of the 24PTH-S DSC used in the model are listed in the following table. 72.48

Nominal 24PTH DSC Dimensions in the OS200 TC

Dimension	(in.)
Outer top cover thickness	1.50
Inner top cover thickness	1.25
Top shield plug thickness	6.25
Total top end	9.00
Inner bottom cover thickness	1.75
Bottom shield plug thickness	4.00
Outer bottom cover thickness	1.75
Total bottom end	7.50
DSC shell thickness	0.50
Cavity length	169.60
Canister length	186.10
Basket height	168.60
DSC shell outer diameter	67.19
Sleeve inner diameter	68.0

Heat load is simulated by heat generation distributed uniformly over the basket length on the homogenized region. The fuel basket is assumed to be centered within the DSC cavity. As such, a 0.5-*inch* long helium filled gap is assumed between the top and bottom of the fuel basket region and the DSC's top and bottom end plates. The volumetric heat generation rate is calculated as:

$$q''' = \frac{Q}{\pi\,(D_i/2)^2\ L_b}$$

P.4.7 Thermal Evaluation for Loading/Unloading Conditions

All fuel transfer operations occur when the NUHOMS®-24PTH DSC and TC are in the spent fuel pool. The fuel is always submerged in free-flowing pool water permitting heat dissipation. After fuel loading is complete, the TC cask and DSC are removed from the pool and the DSC is drained, dried, sealed and backfilled with helium.

The unloading operation considered is the reflood of the 24PTH DSC with water.

This section discusses the thermal evaluation of the loading/unloading conditions of the 24PTH DSC with HLZCs #1 through #5. Section P.4.10 discusses the thermal analysis of the 24PTH Type 1 DSC with HLZC #6 as shown in Appendix P.2, Figure P.2-9. As shown in Section P.4.10, the thermal analysis of HLZC #6 is also bounded by the thermal analysis of HLZC #1 presented in this section.

P.4.7.1 Maximum Fuel Cladding Temperatures During Vacuum Drying

The loading condition evaluated for the NUHOMS®-24PTH DSC is the heatup of the DSC before its cavity is backfilled with helium. This typically occurs during the performance of the vacuum drying operation of the DSC cavity with the TC in the vertical configuration inside the fuel handling building, and the annulus between the TC and the DSC is full of water.

Analyses were performed for the vacuum drying condition in order to ensure that the fuel cladding and 24PTH DSC structural component temperatures remain below the maximum allowable material limits.

During vacuum drying operation, water in the DSC cavity is forced out of the cavity (blowdown operation) using helium before the start of vacuum drying.

The vacuum drying of the DSC is assumed not to reduce the pressure sufficiently to reduce the thermal conductivity of the water vapor or helium in the DSC cavity [4.17] and [4.35]. Radiation in the gaps within the basket and rail components is conservatively neglected.

Thermal analysis is performed using the three-dimensional model developed in Section P.4.6, with decay heat loads for HLZCs 1, 4, and 5 and an initial DSC shell surface temperature of 215°F. The initial temperature of the DSC, basket and fuel is assumed to be 215°F, based on the saturation boiling temperature of the fill water. Table P.4-30 provides the maximum calculated temperatures for the fuel cladding and Table P.4-31 through Table P.4-33 provide the maximum calculated basket component temperatures for all three configurations. The maximum cladding temperature of 578°F for HLZC 4 during vacuum drying is well below the limit of 752°F [4.19].

P.4.7.2 Evaluation of Thermal Cycling of Fuel Cladding During Vacuum Drying, Helium Backfilling and Transfer Operations

ISG-11 [4.19] also states that thermal cycling is to be minimized and imposes a limit of 65°C (117°F) on thermal cycling (reduction in fuel clad temperature from previous peak temperature). The basis for the limit is that as the cladding temperature is reduced more than 65°C the concentration of hydrogen available for hydride reorientation becomes significant.

The thermal analysis of the 24PTH DSC during blowdown operation assumes helium is used to drain the water from the 24PTH DSC cavity and subsequent vacuum drying occurs with a helium environment. This option eliminates a fuel cladding temperature drop that would take place during

P.4.10 *Thermal Evaluations of 24PTH Type 1 DSC Loaded with HLZC #6*

This section evaluates the thermal performance of the 24PTH Type 1 DSC based on the HLZC #6 during storage and transfer conditions with intact FAs. HLZC #6 has a maximum heat load of 35.2 kW, as shown in Appendix P.2, Figure P.2-9.

A review of Table P.4-14 through Table P.4-17, Table P.4-20 through Table P.4-23, and Table P.4-25 through Table P.4-28 shows that the 24PTH DSC with HLZC #1 stored in HSM-H with flat stainless steel heat shields and 100 °F ambient is the bounding case, since it has the highest maximum fuel cladding temperature among all normal and off-normal conditions, and the least margin to the temperature limits. Since no other changes are considered to the 24PTH Type 1 DSC except for the HLZC, the thermal evaluation of 24PTH Type 1 DSC with HLZC #6 is based on a sensitivity study of this bounding case. Therefore, a sensitivity study based on the normal storage (100 °F ambient) in HSM-H with flat stainless steel heat shields is selected to re-evaluate the thermal performance of the 24PTH DSC with HLZC #6, following the same methodology as that discussed in Section P.4.4.4 and Section P.4.6.5.

The following table compares the maximum fuel cladding and DSC component temperatures for the 24PTH DSC with HLZC #6 for normal storage (100 °F ambient) from the sensitivity evaluation to the design basis values presented in Table P.4-14 and Table P.4-15. The design basis values for the 24PTH DSC are based on the normal storage (100 °F ambient) in HSM-H with HLZC #1 with flat stainless steel heat shields as discussed in Section P.4.6.5.

Maximum Component Temperatures for 24PTH DSC for Normal Storage (100 °F Ambient)

HLZC	T_{fuel} (°F)	T_{alum} (°F)	T_{poison} (°F)	T_{tube} (°F)	$T_{DSC\ shell}$ (°F)
Design Basis (HLZC #1, 40.8 kW, Tables P.4-14 and P.4.15)	734	665	666	668	461
HLZC #6, 35.2 kW	697	589	590	591	421
ΔT ($T_{HLZC\ \#6} - T_{Design\ Basis}$)	-37	-76	-76	-77	-40

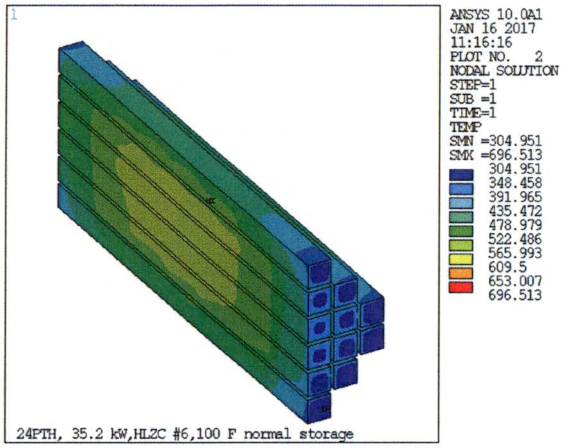
As shown in the above table, the maximum fuel cladding temperature for the 24PTH DSC with HLZC #6 is 697 °F with a significant margin to the temperature limit of 752 °F for normal storage condition. Furthermore, the maximum fuel cladding temperature is 37 °F lower than that determined for the design basis evaluation with HLZC #1. In addition, the maximum temperatures for the basket components determined for 24PTH DSC with HLZC #6 also remain bounded by the maximum temperatures determined for HLZC #1 in Section P.4.6.

Similar to the sensitivity run, temperatures determined for HLZC #1 (40.8 kW) in Section P.4.6 will bound those of HLZC #6 (35.2 kW) for all normal, off-normal and accident conditions, since the only change is the reduction in the heat load. Therefore, the time limits specified for the transfer operation in Section P.4.5.5 for the 24PTH DSC with HLZC #1 at 40.8 kW are also applicable for the 24PTH DSC with HLZC #6 at 35.2 kW.

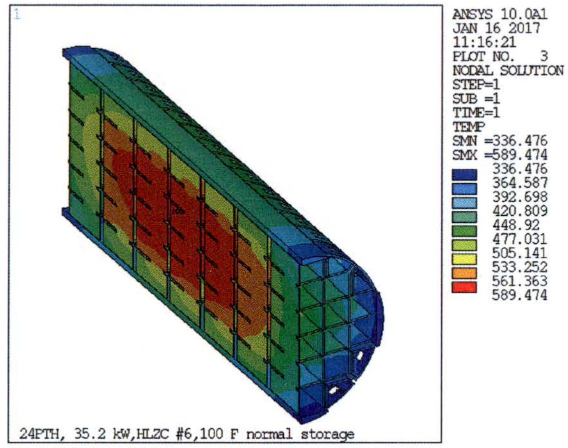
In addition, the average helium temperature determined for HLZC #6 is bounded by that of HLZC #1. Therefore, the maximum internal pressures in Table P.4-19, Table P.4-24, and Table P.4-29 remain bounding for HLZC #6 under normal, off-normal, and accident transfer conditions, respectively.

Based on this discussion, no further evaluations are required for the 24PTH Type 1 DSC with HLZC #6, and all design criteria described in Section P.4.1 are satisfied.

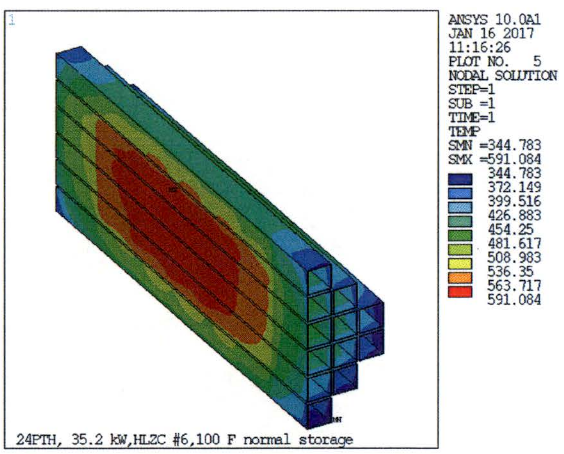
The following figures show typical temperature plots for the 24PTH Type 1 DSC with HLZC #6 for the normal storage condition (100 °F ambient) in HSM-H with flat stainless steel heat shields.



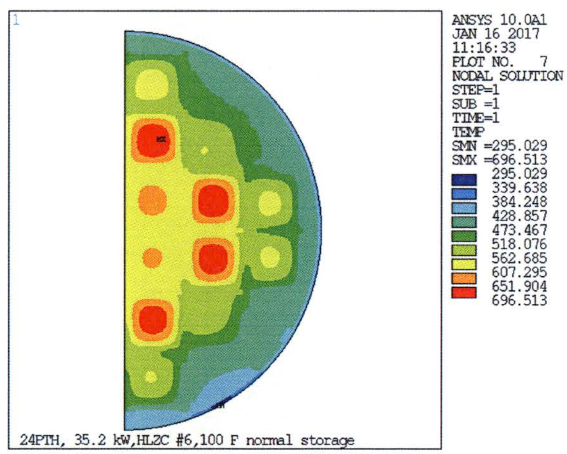
(a) Fuel



(b) Aluminum plates



(c) Tubes



(d) Hottest Cross Section

Temperature Distributions for 24PTH DSC with HLZC #6

Thermal Evaluation of 24PTH-S-LC DSC in OS197 TC

Thermal performance of the 24PTH-S-LC DSC during transfer operations in Standardized TC is based on a two-step approach. In Step 1, the DSC shell temperatures are evaluated as noted in Section P.4.5.1. In Step 2, the DSC temperatures evaluated in Step 1 are utilized as boundary conditions to determine the maximum fuel cladding and basket component temperatures as noted in Section P.4.6.5.2. The temperatures resulting from Step 1 are listed in Table P.4-39.

A similar evaluation to that described in Step 1 was performed to evaluate the thermal performance of a 32PT DSC during transfer operation in the OS197 TC as noted in Appendix M, Section M.4.4.1.6.1. This evaluation considers a two-dimensional (2D) cross section of the 32PT DSC in OS197 TC. Since a 2D cross-section model is employed the results of this evaluation are applicable to any configuration wherein the diameter of the DSC shell, the material of the shell and the heat load are the same. Since the outer diameter, material, and the maximum heat load (i.e. 24 kW) of the 24PTH-S-LC DSC and the 32PT DSC are identical, the DSC shell temperatures presented in Section M.4.4.1.6.1 can be applied to the 24PTH-S-LC DSC. The following table presents a comparison of the DSC shell temperatures determined for the 24PTH-S-LC DSC in the Standardized TC to the temperatures determined for 32PT DSC in the OS197 TC:

Comparison of DSC Shell Maximum Temperatures

Operating Condition	***Standardized TC @ 24 kW***	***OS197 @ 24 kW***
Normal, 100 °F Ambient	*448 [Table P.4-39]*	*445 [Table M.4-3]*
Off-Normal, 117 °F Ambient	*470 [Table P.4-39]*	*433 [Table M.4-9]*
Accident, 117 °F Ambient	*487 [Table P.4-39]*	*600 [Table M.4-14]*

A comparison of the DSC shell maximum temperatures shows that for normal and off-normal conditions, the maximum temperatures determined in the Standardized TC bound that of the OS197 TC. Therefore, no further evaluation is required for normal and off-normal conditions.

For accident conditions, the DSC shell maximum temperature of the OS197 TC is 600 °F and is significantly higher than 487 °F determined in the Standardized TC. This is because, the liquid neutron shield, which improves the thermal performance of the OS197 TC compared to Standardized TC during normal and off-normal conditions, is considered lost during the accident evaluation.

However, this temperature of 600 °F is bounded by the blocked vent accident condition of the 24PTH-S-LC DSC in HSM Model 102, which was analyzed based on a shell temperature of 613 °F as shown in Table P.4-28. As shown in Table P.4-25 and P.4-28 for HLZC # 5, the maximum fuel cladding temperature for blocked vent accident conditions when analyzed based on a bounding 613 °F shell temperature is 821 °F with significant margin to the temperature limit of 1058 °F. Therefore, even under accident conditions in the OS197 TC, the 24PTH-S-LC DSC will maintain the fuel cladding temperature significantly below the temperature limit of 1058 °F.

To estimate the impact on the internal pressure of 24PTH-S-LC DSC during accident conditions due to this temperature increase, the average helium temperature determined for blocked vent accident condition, i.e., 618 °F (See Section P.4.6.7.5) is also assumed for the transfer accident case. The maximum internal pressure for 24PTH-S-LC during a postulated transfer accident is then calculated as:

$$P_{24PTH\text{-}S\text{-}LC,OS197,Acc} = P_{24PTH\text{-}S\text{-}LC,Std,Acc} \times T_{24PTH\text{-}S\text{-}LC,OS197,Acc} / T_{24PTH\text{-}S\text{-}LC,Std,Acc}$$
$$= 95.1 \text{ psia} \times 1078\ °R / 1020\ °R$$
$$= 100.51 \text{ psia}$$
$$= 85.81 \text{ psig.}$$

Where,

$P_{24PTH\text{-}S\text{-}LC,Std,Acc}$ = *Maximum internal pressure in a 24PTH-S-LC DSC during accident conditions in Standardized TC = 80.4 psig = 95.1 psia (See Table P.4-29)*

$T_{24PTH\text{-}S\text{-}LC,Std,Acc}$ = *Average helium temperature in a 24PTH-S-LC DSC during accident conditions in Standardized TC = 560 °F = 1020 °R*

$T_{24PTH\text{-}S\text{-}LC,OS197,Acc}$ = *Average helium temperature in a 24PTH-S-LC DSC during accident conditions in OS197 TC = 618 °F = 1078 °R*

The maximum internal pressure of a 24PTH-S-LC DSC during accident transfer conditions in an OS197 TC is 85.8 psig and remains below the allowable limit of 90 psig listed in Table P.4-29. Therefore, there is no impact on the internal pressure limits during the postulated accident condition.

Table P.4-29
24PTH DSC Maximum Accident Condition Pressures

DSC Type / HLZC #	DSC Cavity Volume (in^3)	DSC Cavity Helium Fill Gas (g-moles)	Fuel Rod Helium-Fill Gas (g-moles)	CC Gas (g-moles)	Fission Products Gases (g-moles)	Total Gas (g-moles)	DSC Cavity Pressure (psig)	DSC Design Pressure (psig)	Bounding Case
24PTH-S DSC / HLZC 1	249,000	128.2	131.4	53.8	342.8	656.2	102.1	120	40.8 kW transfer accident in OS197FC, 117°F amb
24PTH-L DSC / HLZC 1	263,000	135.4	131.4	53.8	342.8	663.3	97.2	120	40.8 kW transfer accident in OS197FC, 117°F amb
24PTH-S-LC DSC / HLZC 5	292,000	153.5	131.4	53.8	342.8	681.4	80.4[(1)]	90	24 kW transfer accident in Standard cask, 117°F amb

Note:

1) *DSC cavity pressure for 24PTH-S-LC during accident transfer conditions in OS197 TC is 85.8 psig as discussed in Section P.4.11.*

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These design features results in the occupational and site dose rates ALARA.

The NUHOMS® 24PTH DSC can also be stored within an upgraded HSM model, designated as HSM-HS as described in Appendix U. From a shielding standpoint, the HSM-HS module is identical to the HSM-H module. Therefore, all calculations performed with the HSM-H are applicable to the HSM-HS.

The NUHOMS® 24PTH DSC is also transferred in a modified version of the OS200 TC as described in Appendix U. The OS200 TC is fitted with an aluminum sleeve to accommodate the smaller diameter 24PTH DSC.

The basket layout for the three DSC configurations is identical except for the length of the DSC components and the shield plug design. The 24PTH-S DSC and 24PTH-L DSC differ in DSC and cavity length, while the 24PTH-S-LC DSC and 24PTH-S DSC differ in cavity length due to a different shield plug design. The 24PTH-L/S has carbon steel shield plugs, while the 24PTH-S-LC has thinner lead shield plugs to increase cavity length to allow for greater fuel lengths in a shorter canister.

Each DSC configuration is designed to store up to 24 intact (and up to 12 damaged, with remaining intact) PWR fuel assemblies. The 24PTH-L and 24PTH-S-LC DSCs are also designed to store up to 24 intact standard PWR fuel assemblies with or without CC; such as burnable poison rod assemblies (BPRAs), Control Rod Assemblies (CRAs), Thimble Plug Assemblies (TPAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and neutron sources. For shielding purposes, the 24PTH-L and the 24PTH-S DSC are identical. Therefore, the shielding evaluation presented herein is not performed for the 24PTH-S DSC.

The 24PTH DSCs are also authorized to store Westinghouse 15x15 class Partial Length Shield Assemblies (PLSAs). The PLSAs are similar to regular fuel assemblies except that a portion (axial section) of the active fuel is replaced by stainless steel rods. In essence, a PLSA rod would therefore consist of a fuel section and a steel section. Fuel qualification of these PLSAs, therefore, requires that the combined source term from the irradiated active fuel and steel regions be bounded by the design basis source terms.

The NUHOMS® 24PTH DSC is also designed to store up to 8 failed fuel assemblies in the peripheral locations of the basket. Each failed fuel assembly is housed inside a failed fuel canister prior to loading in these designated positions within the basket.

Dose rates are calculated for the 24PTH-L DSC within HSM-H. Dose rates are also estimated for the 24PTH-S-LC within a HSM-Model 102. As the HSM-Model 102 provides less shielding than the HSM-H, shielding estimates are not made for the 24PTH-S-LC within HSM-H as the dose rates provided bound this scenario.

The design of the OS197FC TC is identical to the design of OS197/OS197H TC except that the OS197FC TC has a modified top lid. For shielding analysis of 24PTH-S and –L DSCs, OS197FC TC is used to bound the OS197/OS197FC TC also because the design features in the TC radial direction are identical for all three TCs; and OS197FC top axial geometry bounds other TCs.

The design-basis PWR fuel source terms are derived from the bounding fuel, B&W 15x15 Mark B assembly design as described in Section P.5.2.

The NUHOMS®-24PTH DSCs is designed to store PWR fuel assemblies and CC with the characteristics described in Table P.2-1. The 24PTH-S/L DSCs have a maximum decay heat of 2.0 kW per assembly and a maximum heat load of 40.8 kW per canister. Fuel in the 24PTH-S/L DSCs may be stored in *five* alternate heat zoning configurations as shown in Figure P.2-1 through Figure P.2-4 *and Figure P.2-9*. The 24PTH-S-LC DSC has a maximum decay heat of 1.5 kW per assembly and a maximum heat load of 24 kW per canister. The heat zoning configuration to be used for the 24PTH-S-LC DSC is shown in Figure P.2-5. Note that while the B&W, CE, and Westinghouse fuel designs are specifically listed, storing reload fuel designed by other manufacturers is also allowed provided an analysis is performed to demonstrate that the limiting features listed in Table P.2-1 and Table P.2-3 bound the specific manufacturer's replacement fuel. The limiting features are burnup, initial enrichment, cooling time, number of fuel rods, cobalt impurities in the hardware and initial heavy metal weight.

The original 490 kgU fuel qualification tables (FQTs) developed for the 24PTH DSC using SCALE4.4/SAS2H and the 400 kgU FQTs developed using SCALE5/SAS2H have been replaced with FQTs for 380 kgU, 475 kgU, and 492 kgU developed by SCALE6.0/ORIGEN-ARP. These FQTs are documented in Section M.5.2.6.

In this chapter, the original design basis source terms developed by SCALE4.4/SAS2H, and the associated dose rate analysis, are retained as the analysis of record. However, SCALE6.0/ORIGEN-ARP design basis source terms are also developed consistent with the unified FQTs (Technical Specifications Tables 1-3a through 1-3p). These source terms are documented in Section P.5.2.6. The CC source terms from Table P.5-12 are added to the fuel source terms. The MCNP4C2 24PTH-L DSC/HSM-H and OS197FC input files are rerun in MCNP5 using the SCALE6.0/ORIGEN-ARP design basis source terms to determine the impact on the dose rates. To minimize rework, the original analysis is maintained as the analysis of record; however, scaling factors are developed that are used to scale up the dose rates of the analysis of record. These scaling factors are also applied to the 24PTH-S-LC DSC/HSM Model 102 dose rates and the 24PTH-S-LC DSC/Standardized TC dose rates. This updated shielding analysis that determines the effect of uranium loading (380 kgU per assembly) on the dose rates is summarized in Section P.5.4.11.

For the 24PTH-L DSC, Heat Load Zoning Configuration 2 (Figure P.2-2) is the configuration that produces the highest dose rates on the surfaces of the HSM-H and OS197FC TC as compared to configurations 1, 3, *4, and 6* because the highest source fuel assemblies are on the outer periphery of the basket region where self-shielding due to adjacent assemblies is limited. This configuration 2 consists of 20 2.0 kW fuel assemblies located in the outer regions of the DSC. For the 24PTH-S-LC, which has only one heat load zoning configuration (Configuration 5, Figure P.2-5). To bound the shielding analysis for heat load zoning configuration 5, fuel assemblies with a decay heat of 1.5 kW at all 24 location is used. This results in a shielding analysis corresponding to a total of 36 kW decay heat per DSC which is very conservative because the total decay heat in 24PTH-S-LC DSC is limited to 24kW. These bounding gamma and neutron source terms are then used in the radiation shielding models to conservatively calculate dose rates on and around the NUHOMS®-24PTH system. AMD 15 AMD 15

The bounding burnup, minimum initial enrichment and cooling time combinations for the fuel assemblies used in the shielding analyses of the 24PTH-L DSC in the HSM-H and the OS197FC TC are as follows:

- Dose rates with 24PTH-L DSC in HSM-H: 41 GWd/MTU, 3.3 wt. % U-235, 3.0-year cooled fuel

- Dose rates with 24PTH-L DSC in OS197FC TC: 62 GWd/MTU, 3.4 wt. % U-235, 5.6-year cooled fuel

The bounding burnup, minimum initial enrichment and cooling time combinations for the fuel assemblies used in the shielding analysis of the 24PTH-S-LC DSC are as follows:

- Dose rates with 24PTH-S-LC DSC in Standardized TC: 32 GWd/MTU, 2.6 wt. % U-235, 3.0-year cooled fuel

- Dose rates with 24PTH-S-LC DSC in HSM-Model 102: 32 GWd/MTU, 2.6 wt. % U-235, 3.0-year cooled fuel (same as for Standardized TC)

Note that for the 24PTH-L DSC, the source terms are different for calculating dose rate when in HSM-H and OS197FC TC. However, for the 24PTH-S-LC DSC, the source terms are the same for calculating the dose rates when in HSM-Model 102 and Standardized TC. The method of selecting the bounding source terms is explained in detail in Section P.5.2.

The design basis CC source term that envelops all CCs allowed in the 24PTH DSCs is taken from Appendix J for BPRAs with burnups up to 36 GWd/MTU. While Appendix J was developed to specifically address the additional source from a BPRA, this source term is selected as the bounding source term for all CCs. The TPAs and ORAs do not extend into the active fuel region of a fuel assembly. Therefore, they are limited to the source term equivalent to the top AMD 15 & 72.48

plus plenum region source term of a BPRA. However, to be conservative, the full total source term of BPRA is used in the shielding analysis to bound all CCs. The total per canister source term allowed for these CCs is shown in Table P.2-2. The source term energy distribution is shown in Table P.5-12. Any CC to be stored in a 24PTH DSC must be bounded by this source term.

Reconstituted and/or damaged fuel is also acceptable for the DSC payload. Reconstituted fuel may contain up to 10 *irradiated* solid stainless steel rods *per fuel assembly* or unlimited number of lower enriched UO2 rods that replace damaged fuel rods. Note that lower enriched UO2 rods are of similar design and behavior as the standard fuel rods aside from the uranium enrichment. The reconstituted rods can be at any location in the fuel assemblies and the reconstituted assemblies can be placed anywhere in the basket. Reconstituted fuel has a rather small effect on the dose rate such that for cooling times less than 10 years, 1 year of cooling time is added if *10 irradiated stainless* reconstituted rods are present. *Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.* Damaged fuel has essentially no impact on the dose rate as the source term would not be impacted and gross axial source redistribution is not likely. Therefore, shielding analysis results with intact fuel are also applicable to the damaged fuel.

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The fuel qualification for the PLSAs is performed such that the resulting source terms are bounded by those for the design basis B&W 15x15 fuel assemblies. The bounding burnup, enrichment and cooling time combination for the PLSA used in the source term evaluation are as follows:

- 40 GWD/MTU, 1.2 wt. % U-235, 6.5 year cooled fuel

The shielding evaluations for loading and transfer configurations documented herein are based on the OS197 FC TC and are bounding for the OS200 TC. This is due to the fact that the neutron and gamma shielding material thicknesses are slightly higher for the OS200 TC. Further, the aluminum sleeve employed to accommodate the smaller diameter 24PTH DSC within the OS200 TC also provides for slightly enhanced gamma shielding. Therefore, no additional shielding calculations are necessary for the OS200 TC.

The fuel qualification requirements for failed fuel assemblies limits the maximum heat load of failed fuel assemblies to 1.0 kW per assembly. Therefore, the shielding evaluation for the basket containing failed fuel assemblies is bounded by that of the intact fuel assemblies which assume a maximum decay heat of 2.0 kW per assembly. Further, the presence of the failed fuel canister also results in increased gamma shielding within the basket. Therefore, no additional shielding calculations are necessary for the failed fuel assemblies.

The methodology, assumptions, and criteria used in this evaluation are summarized in the following subsections.

P.5.1 Discussion and Results

All 24PTH-L DSC MCNP calculations are performed for heat-load zoning configuration 2 which includes 20 design-basis PWR fuel assemblies (with CC) using 2.0 kW fuel. All 24PTH-S-LC DSC MCNP calculations are performed for 24 design-basis PWR fuel assemblies (with CC) using 1.5 kW fuel.

Table P.5-1 summarizes the maximum and average dose rates for the NUHOMS®-24PTH-L DSC loaded into the NUHOMS® HSM-H.

Table P.5-2 summarizes the maximum and average dose rates for the NUHOMS®-24PTH-S-LC DSC loaded into the NUHOMS® HSM-Model 102. Note that the HSM-H is more heavily shielded than the HSM-Model 102 (thicker roof, shield walls, front and back wall including HSM door); therefore, HSM-Model 102 is conservatively modeled to bound HSM-H.

Table P.5-3 provides a summary of the dose rates on and around the OS197FC TC for transfer of the 24PTH-L DSC under normal, off-normal and accident conditions.

Table P.5-4 provides a summary of the dose rates on and around the OS197FC TC for decontamination and welding operations for the 24PTH-L DSC.

Table P.5-5 provides a summary of the dose rates on and around the Standardized TC for transfer of the 24PTH-S-LC DSC under normal, off-normal and accident conditions.

The dose rates reported in Tables P.5-1 through P.5-5 are scaled by footnotes to account for dose rate increases due to the unified FQTs and corresponding source terms. The unified FQTs are documented in Section M.5.2.6, and the corresponding source terms are documented in Section P.5.2.6. The scaling factors are developed in Section P.5.4.11.

A discussion of the method used to determine the design-basis fuel source terms is included in Section P.5.2. The design basis CC source term which is from Appendix J is shown in Table P.5-12. The shielding material densities are given in Section P.5.3. The method used to determine the dose rates due to design-basis fuel assemblies with CC in the various NUHOMS® 24PT DSC design configurations is provided in Section P.5.4. The shielding evaluation is performed with the MCNP4C2 [5.2] *or MCNP5 [5.19]* code with the ENDF/B-VI cross section library. Sample input files used for calculating neutron and gamma source terms and dose rates are included in Section P.5.5.

The NUHOMS®-24PTH DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. [

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P.5.2 Source Specification

The design basis source terms were originally generated using the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1]. The development of these source terms are provided in this section. The original SAS2H FQTs have been replaced with unified FQTs developed using SCALE6.0/ORIGEN-ARP [5.20]. The development of these FQTs is documented in Section M.5.2.6. Because the FQTs have changed, SCALE6.0/ORIGEN-ARP design basis source terms are developed consistent with the unified FQTs (Technical Specifications Tables 1-3a through 1-3p) to determine the impact on the dose rates and are provided in Section P.5.2.6. AMD 15

The B&W 15x15 assembly is *used as the* fuel assembly design for shielding purposes because it has the highest CO-59 content of the hardware regions as compared to the 14x14, other 15x15, and 17x17 fuel assemblies which are also authorized contents of the NUHOMS®-24PTH DSC. The neutron flux during reactor operation is peaked in the active fuel or in-core region of the fuel assembly and drops off rapidly outside the active fuel region. Much of the fuel assembly hardware is outside of the active fuel region of the fuel assembly. To account for this reduction in neutron flux, the fuel assembly is divided into four exposure "regions." The four axial regions used in the source term calculation are: the bottom (nozzle) region, the active fuel region, the (gas) plenum region, and the top (nozzle) region. The B&W 15x15 fuel assembly masses for each irradiation region are listed in Table P.5-6. The light elements that make up the various materials for the various fuel assembly materials are taken from reference [5.4] and are listed in Table P.5-7. The design basis radiological sources are determined with SAS2H\ORIGEN-S depletion models corresponding to 0.490 MTU/FA heavy metal weight. These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. To account for the reduction in neutron flux outside the active fuel regions neutron flux (fluence) correction factors are applied to light element composition for each region. The neutron flux correction factors which are from Reference [5.15] are given in Table P.5-8.

The relevant design characteristics of the PLSAs important for source term evaluation are shown in Table P.5-6. The calculation of the source terms for the active region portion of the PLSA is identical to that of the design basis fuel assembly outlined above. A neutron flux correction factor is applied to the stainless steel rod section of the PLSA to account for the reduction in the neutron flux at these locations. A flux correction factor is required because these fuel assemblies are irradiated in the peripheral locations of the reactor core and the neutron flux around the steel rods is a few orders of magnitude lower than that around the fuel rods due to absence of any fission source. 72.48 and AMD 15

A flux correction factor of 0.3 (shown in Table P.5-8) which is 1.5 times that for the plenum region is utilized to determine the source terms from the steel rods of the PLSA. The plenum region flux correction factor is chosen due to the similarities in the proximity to the active fuel region. Further, to account for the small variations in the radial flux distribution, an additional factor of 1.5 is utilized so that the resulting correction factor is conservative. This correction factor, therefore, is applicable to the outer row of the PLSAs that "see" the neutron flux from the adjacent, regular, fuel assemblies in the reactor core. For the other rows of steel rods, where the neutron flux practically drops to zero, this factor is conservative.

Evaluations of the existing data with SAS2H and the 44-group ENDF/B-V library used in the analysis are documented in References [5.11] and [5.12]. These comparisons all show generally good agreement between the calculations and measurements, and show no trend as a function of burnup in the data that would suggest that the isotopic predictions, and therefore, neutron and gamma source terms, would not be in good agreement. A similar conclusion is also reached by the results documented in JAERI report [5.13]. In fact, for the case with 46,460 MWd/MTU burnup, the isotopic predictions are all within 2% of those measured. Therefore, the uncertainty in the gamma source term, and associated dose rates, is estimated to be within ±5 %.

The above discussion does not include high-burnup data up to 62 GWd/MTU. However, as documented in Reference [5.14] and confirmed in the SAS2H analysis, the total neutron source with increasing burnup is more and more dominated by spontaneous fission neutrons. Reviewing the output from the SAS2H runs, the neutron source term is due almost entirely to the spontaneous fission of Cm-244 (~94% of all neutrons both spontaneous fission and (α,n)). After reviewing the measured Cm-244 content compared to the Cm-244 content predicted by SAS2H and the 44-group ENDF/B-V library documented in References [5.11] and [5.12] for burnups up to 46,460 MWd/MTU, it is readily apparent that the calculated values are within ±11 % of the measured values, with most of the predicted values within ±5% of the measured. Finally, there is no observed trend as a function of burnup in the data that would indicate that the predicted Cm-244 content is significantly different at higher burnups. Therefore, as the CM-244 isotope accounts for more than 94% of the total neutron source term, the uncertainty in the neutron source and associated neutron dose rates is expected to be less than ±11 %.

As documented in Reference [5.14] and as observed in preparing the fuel qualification tables, the gamma dose rate increases nearly linearly with burnup relative to the direct gamma component and the neutron dose rate increases with burnup to the fourth power. Therefore, as burnups go beyond 45 GWd/MTU, the contribution from neutron (and associated n,γ) components to the total dose rates measured on the surfaces of the DSC, TC and HSM (HSM-H and HSM Model 102) increase in relative importance to that of the gamma component. However, this increase in the importance of the neutron source term has a relatively minor effect on the area dose rates on and around the HSM as these are dominated by the gamma component as shown in Table P.5-1 and Table P.5-2. The surface dose rates on the HSM are dominated by the gamma component because the HSM is constructed of thick reinforced concrete, which is an excellent neutron shield. Therefore, even a postulated substantial increase in the neutron source term would have a relatively minor effect on the site dose rate evaluation presented in Section P.10 of the amendment application.

The occupational exposure calculations demonstrate that most of the dose received by workers during cask loading and transfer operations is due to the gammas on and around the cask. The only surface of the TC that is dominated by neutrons is at the bottom of the cask. A small fraction of the total occupational exposure is due to the doses around the bottom of the cask because very little work is performed on or around the bottom of the cask with fuel in the TC.

As discussed above, any impact of uncertainties in source terms is expected to be negligible for the 24PTH system. Therefore, isotopic depletion calculations with SAS2H for fuel burned above 45 GWd/MTU are appropriate.

The above discussion on the applicability of SCALE 4.4/SAS2H to compute gamma and neutron high-burnup source terms is also largely applicable to SCALE 6.0/ORIGEN-ARP because both code systems utilize ORIGEN-S for the depletion calculation. For PWR fuel assemblies, SAS2H and ORIGEN-ARP generate comparable source terms for equivalent program inputs. The TRITON T-DEPL module of SCALE 6.0 is used to generate ORIGEN-ARP libraries applicable to B&W 15x15 fuel assemblies. These libraries are then used by ORIGEN-ARP to compute gamma and neutron source terms.

Oak Ridge National Laboratory has benchmarked TRITON based on measured data from six different PWRs. This benchmarking is documented in NUREG/CR-6968 [5.16], NUREG/CR-7012 [5.17], and NUREG/CR-7013 [5.18] and includes measurement samples up to a burnup of 78.3 GWd/MTU. A summary of experimental samples including the number of samples for each isotope utilized in the benchmark analysis is provided in Table P.5-27. The benchmark references show that TRITON computed results agree well with experiments, thus verifying the use of SCALE 6.0/ORIGEN-ARP to compute gamma and neutron source terms for high-burnup fuel (burnup ≤ 62 GWd/MTU).

Further, the significant documentation on SCALE TRITON/ORIGEN-ARP benchmark, for example NUREG/CR-6969, NUREG/CR-7162 and ORNL paper ("Analysis of isotopic assay data from the MALIBU program," ORNL paper for the International Conference on Reactor Physics, Switzerland, 2009) indicates that TRITON/ORIGEN-ARP and associated cross-section library are appropriate for decay heat and source term calculations.

Reconstituted and/or damaged fuel is also acceptable for the DSC payload. Reconstituted fuel may contain up to 10 solid stainless steel rods that replace fuel rods. Reconstituted fuel has a rather small effect on the dose rate such that for cooling times less than 10 years, 1 year of cooling time is added if reconstituted stainless steel rods are present. If the cooling time is greater than 10 years, no additional cooling time is needed. *Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.* Additional discussion on the method used to analyze reconstituted fuel is provided in Section P.5.2.5. Damaged fuel has essentially no impact on the dose rate as the source term would not be impacted and gross axial source redistribution is not likely. AMD 15 & 72.48

The design-basis source terms are defined as the burnup/initial enrichment/cooling time combination given in the fuel qualification tables that result in the maximum dose rate on the surface of the HSM (either type) or TC (all types). Note that for a given DSC design, the design basis HSM source will not necessarily be the same as the corresponding design basis TC source. The 1-D discrete ordinates code ANISN [5.5] and the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.3] is used to determine the HSM and TC dose rate for each entry in the fuel qualification tables and thereby determine the design basis source. As ANISN is a 1-D code, a single dose location must be selected for both the HSM and TC for analysis purposes. For the HSM, the roof is selected as the dose location, and for the TC the cask side is selected as the dose location. This approach, described in detail in Section P.5.2.4, is consistent with the method used to determine the fuel qualification tables for the Standardized NUHOMS® canister designs described in Section 7.2.3 and Appendix M.5. The radiological source terms generated in the SAS2H/ORIGEN-S runs are used in the ANISN evaluations to calculate the surface dose rates. The ANISN models are similar to the appropriate MCNP4C2 models for the locations of interest.

entry in the fuel qualification tables. For each qualification table, the burnup/enrichment/cooling time combination that results in the highest dose rate is selected as the design basis source.

The results of the ANISN response function evaluation are given in Table P.5-23 and Table P.5-24 for the 2.0 kW OS197FC TC and HSM-H cases, respectively. The results for the 1.5 kW OS197FC TC and HSM-H cases are given in Table P.5-25 and Table P.5-26, respectively. Note that the 1.5 kW results are assumed to be applicable to the Standardized TC and HSM-Model 102. The maximum dose rate for each table corresponds to the design basis source for that decay heat and shielding configuration. *The response function results in Table P.5-23 through P.5-26 are based on the original FQTs and are retained as an example of how to use the response functions to determine the bounding source terms.*

The results of the ANISN response function evaluation with the PLSAs indicate that they are bounded by the design basis fuel source terms. The dose rate for the DSC with PLSA in the OS197FC TC for is 876 mrem/hour which is below the *original* design basis value of 907 mrem/hour. The dose rate for the DSC with PLSA in the HSM-H is 3.7 mrem/hour which is below the *original* design basis value of 6.1 mrem/hour. Therefore, the source terms for the PLSA are bounded by the design basis source terms.

P.5.2.5 Reconstituted Fuel

As explained in Section P.5.2, reconstituted fuel assemblies may contain up to 10 stainless steel rods that replace damaged fuel rods. Because steel rods replace fuel rods, the decay heat of a reconstituted assembly is typically less than the decay heat of an equivalent standard assembly. Conversely, because steel contains Co-59 which activates to form Co-60, for low cooling times a reconstituted assembly typically generates higher dose rates than an equivalent standard assembly. As the half-life of Co-60 is 5.27 years, after 10 years the Co-60 activity has reduced by almost a factor of four and a reconstituted assembly no longer generates higher dose rates than an equivalent standard assembly. To bound this effect, the fuel qualification tables require that for reconstitute rods with cooling times less than 10 years, additional one year of cooling time is required. For cooling times of 10 years or greater, no additional cooling time is required to bound the reconstituted fuel with steel rods.

To quantify this statement, additional SAS2H runs are generated for reconstituted assemblies. For each burnup and enrichment corresponding to a transition point in a fuel qualification table (i.e., the point where the cooling time experiences a change of 0.5 years), reconstituted assembly SAS2H models are developed.

The SAS2H input files for a reconstituted assembly are very similar to the input files for a standard assembly except for the following changes: (1) The number of fuel rods is reduced from 208 to 198, (2) the POWER input variable is adjusted to maintain the correct burnup for the reduced fuel loading, and (3) the light elements change to reflect that 10 fuel rods have been replaced with steel rods. The constituent masses of the reconstituted fuel assembly required for the SAS2H input is provided in Table P.5-6.

Note that a reconstituted rod cannot be irradiated for more than two cycles because the first cycle will always contain fresh, undamaged fuel. To accurately model this behavior, two SAS2H models are generated for each transition point. The first SAS2H model is for only one cycle of irradiation of 10 reconstituted rods, while the second SAS2H model is for three cycles of irradiation of 10 reconstituted rods. By subtracting the single cycle source term of the reconstituted rods from the total source term (fuel and reconstituted rods) for three cycles, the source term for three cycle irradiation of fuel and two cycle irradiation of reconstituted rods is generated.

This source term is inserted into the HSM-H and OS197FC TC response functions to determine the dose rates for comparison to the design basis source dose rates. If the reconstituted fuel dose rate for either the HSM-H or OS197FC TC exceeded the dose rate with design basis fuel, an additional 0.5 year of cooling time is added to the reconstituted fuel source term. When the reconstituted fuel is examined in this fashion, no more than one additional year of cooling time is required for reconstituted fuel to be bounded by the design basis source if the decay time listed in the fuel qualification table is less than 10 years. After a cooling time greater than 10 years the effects of reconstituted fuel become insignificant. *Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.* AMD 15 & 72.48

A sensitivity study performed using source terms consistent with the unified FQTs (Technical Specifications Tables 1-3a through 1-3p) indicates that the additional cooling time requirements for reconstituted fuel remain valid for the unified FQTs.

P.5.2.6 *<u>SCALE6.0/ORIGEN-ARP Source Terms</u>*

Because the original FQTs have been replaced with unified FQTs (Technical Specifications Tables 1-3a through 1-3p), the design basis source terms developed in Section P.5.2 are obsolete because they are based upon burnup, enrichment, and cooling time combinations that are no longer applicable. Therefore, SCALE6.0/ORIGEN-ARP design basis source terms are developed based on the unified FQTs. The FQTs are documented in Section M.5.2.6. AMD 15

The methodology used to develop the SCALE6.0/ORIGEN-ARP design basis source terms for HLZC#2 is the same as described in Section P.5.2. The ANISN transfer cask and HSM response functions developed in Section P.5.2.4 are used to evaluate the source terms for each FQT burnup, enrichment, and cooling time (BECT) combination. The BECT combination that results in the maximum dose rate is selected as the design basis source. For HLZC#6, because the heat load zone configuration is not uniform, the three-zone 32PTH1 DSC response functions from Chapter U.5, Tables U.5-15 and U.5-16, are used. Because the response functions are only used to rank the BECT combinations, using the 32PTH1 DSC response functions for the 24PTH DSC HLZC#6 is acceptable.

It is demonstrated in Section P.5.2 that Heat Load Zone Configuration 2 (HLZC#2) bounds Configurations 1, 3, 4, and 5. However, the analysis in Section P.5.2 does not consider HLZC#6, which has been added at a later time. Comparing HLZC#2 and #6, it is observed that HLZC#2 has both a larger overall heat load and hotter fuel assemblies in the peripheral zone. However, HLZC#6 has the hottest overall fuel assembly (2.5 kW). Therefore, source terms are developed both for HLZC#2 and #6.

Based on the ANISN response function analysis, the SCALE6.0/ORIGEN-ARP design basis source terms for HLZC#2 and #6 are:

HLZC#2

HSM

- *Table P.5-28: 2.0 kW/FA, 29 GWd/MTU, 1.1 wt.% U-235, 2.0 years cooled*

Transfer cask

- *Table P.5-29: 2.0 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 2.9 years cooled*

Note that for HLZC#6, 0.6 kW/FA is a heat load limit for fuel locations in both the innermost and second rows. The innermost 0.6 kW fuel assembly is labeled "0.6 kW (inner)" and the 0.6 kW fuel assembly in the second row is labeled "0.6 kW (outer)." Note also that the HSM and TC source terms are the same for the 0.6 kW/FA (inner) and 2.5 kW/FA zones.

HLZC#6:

HSM

- *Table P.5-30: 0.6 kW/FA (inner), 45 GWd/MTU, 1.1 wt.% U-235, 18.2 years cooled*
- *Table P.5-31: 0.6 kW/FA (outer), 10 GWd/MTU, 0.7 wt.% U-235, 2.6 years cooled*
- *Table P.5-32: 1.3 kW/FA, 16 GWd/MTU, 0.8 wt.% U-235, 2.0 years cooled*
- *Table P.5-33: 2.5 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 2.3 years cooled*

Transfer cask

- *Table P.5-30: 0.6 kW/FA (inner), 45 GWd/MTU, 1.1 wt.% U-235, 18.2 years cooled*
- *Table P.5-34: 0.6 kW/FA (outer), 31 GWd/MTU, 1.1 wt.% U-235, 6.6 years cooled*
- *Table P.5-35: 1.3 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 4.6 years cooled*
- *Table P.5-33: 2.5 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 2.3 years cooled*

Because the FQTs in Section M.5.2.6 are developed for uranium loadings of 380 kgU, 475 kgU, and 492 kgU for fixed heat loads, it is observed that 380 kgU source terms bound 475 or 492 kgU source terms because self-shielding of the sources by the uranium in the fuel matrix is reduced. Therefore, SCALE 6.0/ORIGEN-ARP design basis source terms are developed only for 380 kgU.

The MCNP5 neutron models are run with the NONU card to suppress subcritical neutron multiplication. Subcritical neutron multiplication is addressed by multiplying the neutron source computed by ORIGEN-ARP by $1/(1 - k_{eff})$. Values of k_{eff} appropriate for the burnups of the sources are provided in the source term tables (Table P.5-28 through P.5-35).

The gamma source in the active fuel region is modeled with the axial burnup profile appropriate for the burnup of the source and is obtained from Table 20 of ORNL/TM-12973 [5.21]. These profiles are summarized in Appendix M.5, Table M.5-53. The neutron profile is derived as the 4^{th} power of the gamma profile and is also summarized in Table M.5-53. The burnup peaking factor accounts for the increase in the neutron source magnitude due to the axial burnup profile. The burnup peaking factors used in the neutron calculations are provided in the source term tables (Table P.5-28 through P.5-35).

The CC source terms provided in Table P.5-12 are applicable and may be added to the fuel-only source terms provided in Table P.5-28 through Table P.5-35.

P.5.4 Shielding Evaluation

Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from 20 or 24 *0.490 MTU* fuel assemblies with *control components (*CCs*)* are calculated with the MCNP4C2 Code [5.2] at various location*s* on and around the NUHOMS® -24PTH DSCs, HSM, and TC.

The following shielding evaluation discussion specifically addresses the NUHOMS®-24PTH-L in an HSM-H or OS197FC TC, and the 24PTH-S-LC DSC in an HSM-Model 102 or Standardized TC using the *0.490 MTU* design-basis source terms determined in Section P.5.2.

Dose rate contributions from the bottom, in-core, plenum and top regions, as appropriate, from 24 0.380 MTU fuel assemblies with CCs are also calculated with the MCNP5 Code [5.21] at various locations on and around the NUHOMS® 24PTH DSCs within the HSM and TC.

The shielding evaluation that determines the effect of loading 0.380 MTU per assembly on the dose rates is described in Section P.5.4.11.

P.5.4.1 Computer Program

MCNP4C2 [5.2] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Pointwise (continuous energy) cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation are accounted for in the cross section set. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. Important standard features that make MCNP4C2 very versatile and easy to use include a powerful general source; an extensive collection of cross-section data; and an extensive collection of variance reduction techniques that can be employed to track particles through very complex deep penetration problems.

An updated version of the MCNP code, MCNP5 [5.21] with the continuous energy ENDFB-VI cross section library is used to determine the dose rates for the shielding analysis described in Section P.5.4.11. MCNP5 has been used to perform the shielding analysis of the 24PTH System (Appendix P5), the 32PT System (Appendix M5), the 32PTH1 System (Appendix U.5), and the 37PTH System (Appendix Z.5).

P.5.4.2 Spatial Source Distribution

The source components are:

- The neutron sources due to the active fuel region,
- The gamma source due to the active fuel region,

- The gamma source due to the plenum,
- The gamma source due to the top region,
- The gamma source due to the bottom region,
- The gamma source due to the CC in the active fuel region,
- The gamma source due to the CC in the plenum region, and
- The gamma source due to the CC in the top region.

Axial peaking is accounted for in the active fuel region by inputting an axial shape, as discussed in Section P.5.2.3.

Cask Decontamination. The 24PTH-L DSC and the OS197FC TC are assumed to be completely filled with water, including the region between 24PTH-DSC and cask, which is referred to as the "cask/24PTH-DSC annulus." The 24PTH-DSC inner cover plate is assumed to be in place and the temporary shielding has not yet been installed. Results for this case are provided in Table P.5-4.

Welding and 24PTH-L DSC Draining. Before the start of welding operation, approximately 60% of the water in the DSC cavity is removed due to hydrogen generation. A dry DSC cavity is assumed in all welding models to be conservative. Temporary shielding consisting of three inches of NS3 and one inch of steel is assumed to cover the 24PTH-L DSC top shield plug. In addition, the DSC outer top cover plate is not present. The cask/24PTH-DSC annulus is assumed to remain completely filled with water. Results for this case are provided in Table P.5-4.

P.5.4.10 Impact on Dose Rates due to Reduced Density Concrete and Gaps between HSMs

A bounding analysis is performed by employing a minimum concrete density of 140 pounds per cubic foot (pcf) in the HSM-H MCNP model combined with a maximum gap of 1.5 inches between adjacent HSM-H modules and shield walls to determine the effect on maximum and average dose rates due to a fully loaded 32PTH1 DSC. These calculations are documented in Appendix U.5, Section U.5.4.10. The ratios shown in Appendix U.5, *Table U.5-18* and *Table U.5-19* can be used as scaling factors to increase the maximum and surface-average dose rates of the 24PTH in the HSM-H to account for low density concrete and 1.5-inch gaps during HSM fabrication and installation. Note that the HSM-H concrete contains high density rebar which is not credited in the MCNP models. Further, the modules are installed adjacent to each other such that there will not be a "uniform" gap of 1.5 inches. Ignoring the effect due to increased vent dose rates, the increase in the average dose rates caused by both the maximum postulated uniform gaps and the minimum postulated concrete density is expected to be less than 20% at the front and roof surfaces of the HSM-H module. Dose reduction hardware may be installed to further reduce these dose rates.

P.5.4.11 Shielding Analysis with a Loading of 0.380 MTU per Fuel Assembly

As discussed in Section P.5.4, additional shielding analysis is performed with a reduced Uranium loading of 0.380 MTU per fuel assembly. The objective of this analysis is to determine the impact that reduced uranium loading has on system dose rates. The results of this analysis are employed to scale the dose rate results for the 24PTH System (all DSCs). For this purpose, the MCNP4C2 models used for the 0.490 MTU analyses are rerun using MCNP5 with updated source terms as described in P.5.2.6, and with updated material specifications to reflect the reduction in MTU. MCNP5 calculations are performed for the 24PTH-L DSC inside the HSM-H in the normal storage configuration, and dose rate scaling factors are derived using the same methodology as that described in Appendix U, Section U.5.4.12. MCNP5 calculations are also performed for the 24PTH-L DSC inside the OS197 TC in the decontamination and welding configurations, and in the normal and accident transfer configurations, and dose rate and occupational exposure scaling factors are derived using the same methodology as that described in Appendix U, Section U.5.4.12. These results are also applicable to the 24PTH-S and 24PTH-S-LC DSCs. Based on the updated results, six scaling factors are determined and are summarized as follows:

- *The dose rates for the HSM-H front and roof are to be scaled by 1.13.*
- *The dose rates for the HSM-H side and rear are to be scaled by 1.30.*
- *The site dose for the HSM is to be scaled by 1.13.*
- *The dose rates for the TC for normal, welding and decontamination are to be scaled as follows:*
    - *No scaling is required for the side,*
    - *by 1.18 for the top,*
    - *by 1.19 for the bottom.*
- *The dose rates for the TC for accidents are to be scaled by 1.13.*
- *The occupational exposure for the TC loading and storage operations is to be scaled by 1.10.*

These scaling factors are included as footnotes in the dose rate results summarized in Table P.5-1 through Table P.5-5, Table P.5-21, Table P.5-22, Table P.5-24, and Table P.5-26.

These scaling factors are also used to scale the occupational exposure and generic site dose (2X10 back-to-back and front-to-front arrays) results calculated for the 24PTH System in Appendix P.10, and to scale the dose rate consequences of accidents for the 24PTH System in Appendix P.11.

P.5.6 References

5.1 Oak Ridge National Laboratory, RSIC Computer Code Collection, “SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluations for Workstations and Personal Computers,” NUREG/CR-0200, Revision 6, ORNL/NUREG/CSD-2/V2/R6.

5.2 “Monte Carlo N-Particle Transport Code System,” CCC-701, Oak Ridge National Laboratory, RSICC Computer Code Collection, June 2001.

5.3 CASK-81 - 22 Neutron, 18 Gamma-Ray Group, P3, Cross Sections for Shipping Cask Analysis,” DLC-23, Oak Ridge National Laboratory, RSIC Data Library Collection, June 1987.

5.4 Ludwig, S.B., and J.P. Renier, “Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code,” ORNL/TM-11018 Oak Ridge National Laboratory, December 1989.

5.5 “ANISN-ORNL - One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering”, CCC-254, Oak Ridge National Laboratory, RSIC Computer Code Collection, April 1991.

5.6 “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,” NUREG/CR-6801, Oak Ridge National Laboratory.

5.7 Jenal, J. P., P. J. Erickson, W. A. Rhoades, D. B. Simpson, and M. L. Williams, “The Generation of a Computer Library for Discrete Ordinates Quadrature Sets,” ORNL/TM-6023, Oak Ridge National Laboratory, October 1977.

5.8 “American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors,” ANSI/ANS-6.1.1-1977, American Nuclear Society, LaGrange Park, Illinois, March 1977.

5.9 K.Ueki, N. Nariyama, A. Ohashi, A. Yamaji. “Measurement of Dose-Equivalent Rates around a Cask and Monte Carlo Analysis with Actual Configuration of Fuel Basket”. Journal of Nuclear Science and Technology, (Supplement 1, p.324-328), Atomic Energy Society of Japan, March 2000, ISSN 0022-3131.

5.10 *Deleted.*

5.11 MD DeHart and OW Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, September 1996.

5.12 OW Hermann, SM Bowman, MC Brady, CV Parks, "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, March 1995.

5.13 Japan Atomic Energy Research Institute, "Technical Development on Burn-up Credit for Spent LWR Fuels," JAERI-Tech 2000-071, September 21, 2000.

5.14 U.S. Nuclear Regulatory Commission, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High Burnup LWR Fuel," NUREG/CR-6700, Published January 2001, ORNL/TM-2000/284.

5.15 "Characteristics of Potential Repository Waste," DOE/RW-0184-R21, Volume 1, Oak Ridge National Laboratory, Tennessee, July 1992.

5.16 U.S. Nuclear Regulatory Commission, "Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation-Calvert Cliffs, Takahama, and Three Mile Island Reactors," NUREG/CR-6968, Published February 2010, ORNL_TM-2008-71.

5.17 U.S. Nuclear Regulatory Commission, "Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel," NUREG/CR-7012, Published January 2011, ORNL-TM-2010-41.

5.18 U.S. Nuclear Regulatory Commission, "Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation -- Vandellós II Reactor," NUREG/CR-7013, Published January 2011, ORNL-TM-2009-321.

5.19 LA-UR-03-1987, MCNP - A General Monte Carlo N-Particle Transport Code, Version 5, Los Alamos National Laboratory, April 2003.

5.20 ORNL/TM-2005/39, Version 6, SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Oak Ridge National Laboratory, January 2009.

5.21 "Monte Carlo N-Particle Transport Code System," CCC-730, Oak Ridge National Laboratory, RSICC Computer Code Collection, August 2001.

Table P.5-1
Summary of NUHOMS®-24PTH-L DSC in HSM-H, Maximum and Average Dose Rates, Configuration 2 (2)

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1, 5) (mrem/hr)	Total MCNP 1σ Error
HSM Roof (centerline) *(3)*	20.1	0.038	0.5	0.018	20.6	0.037
HSM Roof Birdscreen*(3)*	205.8	0.019	4.1	0.012	209.9	0.018
HSM End (Side) Shield Wall Surface*(4)*	3.4	0.081	0.1	0.016	3.5	0.079
HSM Door Exterior Surface (centerline) *(3)*	1.3	0.143	0.1	0.524	1.3	0.139
HSM Front Birdscreen*(3)*	1232.0	0.068	5.5	0.076	1237.0	0.068

Dose Rate Location	Average (mrem/hr)	Gamma MCNP 1σ Error	Average Neutron (mrem/hr)	Neutron MCNP 1σ Error	Average Total (mrem/hr)*(5)*	Total MCNP 1σ Error
HSM Roof*(3)*	20.3	0.011	0.5	0.006	20.8	0.011
HSM End (Side) Shield Wall Surface*(4)*	1.0	0.016	0.1	0.033	1.1	0.015
HSM Front *(3)*	32.2	0.047	0.1	0.066	32.3	0.047
HSM Back Shield Wall*(4)*	0.6	0.074	0.1	0.025	0.6	0.074

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) Dose *rates* calculated using Configuration 2 in 24PTH-L DSC bounds configurations 1, 3, 4, *and 6*. Dose rates can be higher by 6% to account for the use of grout during HSM fabrication and installation.

(3) These dose rates increase by 13% when loading 0.380 MTU FAs.

(4) These dose rates increase by 30% when loading 0.380 MTU FAs.

(5) Use the ratios shown in Appendix U.5, *Table U.5-18 and Table U.5-19* to increase the maximum and surface-average dose rates, respectively to account for reduced density concrete and gaps of up to 1.5″ as described in Appendix U.5, Section U.5.4.10.

Table P.5-2
Summary of NUHOMS®-24PTH-S-LC DSC in HSM-Model 102, Maximum and Average Dose Rates, Configuration 5[(1)]

Dose Rate Location	Maximum Gamma (mrem/hr)	Maximum Neutron (mrem/hr)	Maximum Total (mrem/hr)
HSM Roof (centerline)[(2)]	59.3	0.2	59.5
HSM Roof Birdscreen[(2)]	976.5	3.2	979.7
HSM End (Side) Shield Wall Surface[(3)]	266.9	0.3	267.2
HSM Door Exterior Surface [(2)] (centerline)	60.6	1.6	62.2
HSM Front Birdscreen[(2)]	489.6	2.5	492.1
HSM Back Shield Wall[(3)]	2.5	0.02	2.5

Dose Rate Location	Average Gamma (mrem/hr)	Average Neutron (mrem/hr)	Average Total (mrem/hr)
HSM Roof[(2)]	47.3	0.2	47.5
HSM End (Side) Shield Wall Surface[(3)]	31.7	0.1	31.8
HSM Front[(2)]	45.6	0.9	46.5
HSM Back Shield Wall[(3)]	0.8	0.01	0.8

Notes:

(1) Dose rates can be higher by 6% to account for the use of grout during HSM fabrication and installation.

(2) These dose rates increase by 13% when loading 0.380 MTU FAs.

(3) These dose rates increase by 30% when loading 0.380 MTU FAs.

Table P.5-3
Summary of NUHOMS®-24PTH-L DSC, OS197FC TC Maximum Dose Rates During Transfer Operations, Configuration 2

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1) (mrem/hr)	Total MCNP 1σ Error
Cask Side Surface (Radial)(3)	7.45E+02	0.0180	7.56E+02	0.0120	1.50E+03	0.0108
Cask Top Axial Surface(4)	2.37E+02	0.0566	4.48E+01	0.0499	2.61E+02	0.0523
Cask Bottom Axial Surface(5)	1.66E+03(2)	0.0353	2.57E+03(2)	0.0246	4.23E+03(2)	0.0204
1 ft from Cask Side (Radial)(3)	4.76E+02	0.0179	4.82E+02	0.0107	9.58E+02	0.0104
1 ft from Cask Top Axial Surface(4)	7.86E+01	0.0741	3.11E+01	0.0455	9.58E+01	0.0636
1 ft from Cask Bottom Axial Surface(5)	9.80E+02	0.0355	9.44E+02	0.0340	1.92E+03	0.0246
3 ft from Cask Side (Radial)(3)	2.85E+02	0.0163	2.78E+02	0.0096	5.63E+02	0.0095
3 ft from Cask Top Axial Surface(4)	3.95E+01	0.1189	1.47E+01	0.0550	5.05E+01	0.0972
3 ft from Cask Bottom Axial Surface(5)	3.44E+02	0.0341	2.79E+02	0.0600	6.23E+02	0.0328
Cask 1 m (Radial) Accident Condition(6)	3.10E+02	0.0011	3.19E+03	0.0124	3.51E+03	0.0128
Cask 100 m (Radial) Accident Condition(6)	1.50E-01	0.0366	5.10E-01	0.0134	6.61E-01	0.0124
Cask 500 m (Radial) Accident Condition(6)	4.95E-04	0.8299	4.10E-04	0.0305	9.05E-04	0.0194

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 340 mrem/hr gamma, 419 mrem/hr neutron for a total average dose rate of 758 mrem/hr.

(3) The Side dose rates do not need to be scaled when loading 0.380 MTU FAs.

(4) The Top dose rates increase by 18% when loading 0.380 MTU FAs.

(5) The Bottom dose rates increase by 19% when loading 0.380 MTU FAs.

(6) The Accident dose rates increase by 13% when loading 0.380 MTU FAs.

Table P.5-4
Summary of NUHOMS®-24PTH-L DSC, OS197FC TC Maximum Dose Rates During Decontamination and Welding Operations, Configuration 2

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1) (mrem/hr)	Total MCNP 1σ Error
Decontamination						
Cask Side Surface (Radial)(4)	4.34E+02	0.0210	8.23E+02	0.0069	1.26E+03	0.0085
Top Axial Surface(5)	7.83E+02	0.0272	3.14E-01	0.2858	7.83E+02	0.0272
Cask Bottom Axial Surface(6)	1.15E+03(2)	0.0478	5.83E+01(2)	0.0126	1.21E+03(2)	0.0455
1 ft from Cask Side (Radial)(4)	2.82E+02	0.0206	5.32E+02	0.0060	8.14E+02	0.0082
1 ft from Top Axial Surface(5)	5.93E+02	0.0262	1.05E+01	0.0304	5.93E+02	0.0262
1 ft from Cask Bottom Axial Surface(6)	7.07E+02	0.0486	2.73E+01	0.0215	7.29E+02	0.0471
3 ft from Cask Side (Radial)(4)	1.68E+02	0.0187	3.14E+02	0.0054	4.83E+02	0.0074
3 ft from Top Axial Surface(5)	4.02E+02	0.0305	9.40E+00	0.0099	4.03E+02	0.0305
3 ft from Cask Bottom Axial Surface(6)	2.54E+02	0.0494	1.86E+01	0.0085	2.62E+02	0.0480
Welding						
Cask Side Surface (Radial)(4)	6.22E+02	0.0224	5.46E+02	0.0123	1.17E+03	0.0132
Top Axial Surface(5)	8.56E+02	0.0264	3.37E+01	0.0658	8.84E+02	0.0256
Cask Bottom Axial Surface(6)	1.64E+03(3)	0.0397	2.69E+03(3)	0.0297	4.34E+03(3)	0.0238
1 ft from Cask Side (Radial)(4)	4.06E+02	0.0217	3.51E+02	0.0108	7.58E+02	0.0127
1 ft from Top Axial Surface(5)	6.48E+02	0.0371	2.42E+01	0.0814	6.69E+02	0.0360
1 ft from Cask Bottom Axial Surface(6)	9.78E+02	0.0401	9.23E+02	0.0395	1.90E+03	0.0282
3 ft from Cask Side (Radial)(4)	2.47E+02	0.0191	2.05E+02	0.0097	4.52E+02	0.0113
3 ft from Top Axial Surface(5)	4.44E+02	0.3175	1.33E+01	0.0815	4.51E+02	0.3124
3 ft from Cask Bottom Axial Surface(6)	3.41E+02	0.0386	2.51E+02	0.0663	5.92E+02	0.0358

Notes:
(1) Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.
(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 238 mrem/hr gamma, 13 mrem/hr neutron for a total average dose rate of 251 mrem/hr.
(3) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 331 mrem/hr gamma, 417 mrem/hr neutron for a total average dose rate of 748 mrem/hr. Note that this bottom axial dose rate has no impact on the occupational exposure because no operations are performed near bottom axial location.
(4) The Side dose rates do not need to be scaled when loading 0.380 MTU FAs.
(5) The Top dose rates increase by 18% when loading 0.380 MTU FAs.
(6) The Bottom dose rates increase by 19% when loading 0.380 MTU FAs.

Table P.5-5
Summary of NUHOMS®-24PTH-S-LC DSC, Standardized TC Maximum Dose Rates During Transfer Operations, Configuration 5

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1) (mrem/hr)	Total MCNP 1σ Error
Cask Side Surface (Radial)(3)	4.19E+02	0.0600	1.81E+02	0.0178	5.77E+02	0.0273
Cask Top Axial Surface(4)	3.01E+01	0.0894	8.03E+00	0.0778	3.25E+01	0.0829
Cask Bottom Axial Surface(5)	4.42E+03(2)	0.1154	3.30E+02(2)	0.0276	4.75E+03(2)	0.1074
1 ft from Cask Side (Radial)(3)	2.95E+02	0.0536	1.14E+02	0.0176	3.78E+02	0.0242
1 ft from Cask Top Axial Surface(4)	2.06E+01	0.0251	5.44E+00	0.0576	2.43E+01	0.0224
1 ft from Cask Bottom Axial Surface(5)	2.31E+03	0.1261	1.17E+02	0.0308	2.43E+03	0.1200
3 ft from Cask Side (Radial)(3)	1.89E+02	0.0449	9.20E+01	0.0158	2.31E+02	0.0368
3 ft from Cask Top Axial Surface(4)	1.06E+01	0.0201	2.73E+00	0.0567	1.31E+01	0.0165
3 ft from Cask Bottom Axial Surface(5)	8.82E+02	0.1398	3.33E+01	0.0403	9.15E+02	0.1347
Cask 1 m (Radial) Accident Condition(6)	3.44E+02	0.0018	4.18E+02	0.0219	7.62E+02	0.0258
Cask 100 m (Radial) Accident Condition(6)	1.67E-01	0.0872	6.76E-02	0.0232	2.35E-01	0.0308
Cask 500 m (Radial) Accident Condition(6)	6.20E-04	1.1638	5.34E-05	0.0341	6.74E-04	0.0279

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location therefore the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 730 mrem/hr gamma, 61 mrem/hr neutron for a total average dose rate of 791 mrem/hr.

(3) The Side dose rates do not need to be scaled when loading 0.380 MTU FAs.

(4) The Top dose rates increase by 18% when loading 0.380 MTU FAs.

(5) The Bottom dose rates increase by 19% when loading 0.380 MTU FAs.

(6) The Accident dose rates increase by 13% when loading 0.380 MTU FAs.

Table P.5-21
Surface Average Dose Rates on HSM-Model 102 with 24PTH-S-LC DSC

Surfaces	Dose Components	Dose Rate from Table N.5-4 of Appendix N (mrem/hr)	Scaling factors	Dose Rate (mrem/hr)
Back(3)	Gamma	1.3	0.6	0.8
	Neutron	0.1	0.1	0.01
Front (excluding bird screen) (1)(2)	Gamma	4.9	6.8	33.5
	Neutron	2.0	0.4	0.9
Roof (excluding bird screen) (1) (2)	Gamma	25.4	1.1	27.1
	Neutron	0.5	0.2	0.1
Side(3)	Gamma	29.6	1.1	31.7
	Neutron	0.5	0.2	0.1
Front Bird Screen(2)	Gamma	261.3	1.1	279.6
	Neutron	6.2	0.2	1.1
Roof Bird Screen(2)	Gamma	408.9	1.1	437.5
	Neutron	9.0	0.2	1.5

AMD 15

Notes: 72.48

(1) If the front average dose rate includes the contribution from the front birdscreen, the dose rates are 45.6 mrem/hr for gammas and 0.9 mrem/hr for neutron radiation. Likewise, if the roof average dose rate includes the contribution from the roof birdscreen, the dose rate is 47.3 mrem/hr for gammas and 0.2 mrem/hr for neutron radiation.

(2) These dose rates increase by 13% when loading 0.380 MTU FAs. AMD 15

(3) These dose rates increase by 30% when loading 0.380 MTU FAs.

Table P.5-22
Maximum Dose Rates on HSM-Model 102 with 24PTH-S-LC DSC

Surfaces	Dose Components	Dose Rate from Table N.5-4 of Appendix N (mrem/hr)	Scaling factors	Dose Rate (mrem/hr)
Back(3)	Gamma	4	0.6	2.5
	Neutron	0.1	0.1	0.02
Front(3)	Gamma	9	6.8	60.6
	Neutron	4	0.4	1.6
Roof(2)	Gamma	55	1.1	59.3
	Neutron	1.0	0.2	0.2
Side(3)	Gamma	250	1.1	266.9
	Neutron	2.0	0.2	0.3
Front Bird Screen(2)	Gamma	458	1.1	489.6
	Neutron	14(1)	0.2	2.5
Roof Bird Screen(2)	Gamma	913	1.1	976.5
	Neutron	18	0.2	3.2

AMD 15

Notes:

72.48

(1) Not calculated in appendix N.5. Estimated here as approximately twice the average dose rate.

(2) These dose rates increase by 13% when loading 0.380 MTU FAs.

(3) These dose rates increase by 30% when loading 0.380 MTU FAs. AMD 15

Table P.5-23 *OS197 FC* TC Total Dose Rates (mrem/hr) at Cask Centerline for 2.0 kW Case[(1)]

Burn-Up, GWD/MTU	Maximum Assembly Average Initial U-235 Enrichment, wt %																														
	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	258.4	235.0	231.2	227.6	224.3	221.1	218.2	215.4	212.8	210.3	208.0	205.7	203.6	201.6	199.7	197.9	196.2	194.6	193.0	191.5	190.1	188.7	187.3	186.1	184.9	183.7	182.6	181.5	180.4	179.4	178.4	177.5
15	409.4	373.3	367.4	361.7	356.4	351.4	346.7	342.2	338.0	333.9	330.1	326.5	323.1	319.8	316.7	313.7	310.9	308.2	305.6	303.1	300.7	298.5	296.3	294.2	292.2	290.2	288.3	286.5	284.8	283.1	281.5	279.9
20	580.7	529.6	521.0	512.8	505.1	497.8	490.9	484.4	478.2	472.2	466.6	461.3	456.3	451.4	446.8	442.4	438.2	434.2	430.4	426.7	423.2	419.8	416.5	413.4	410.4	407.5	404.8	402.1	399.5	397.0	394.6	392.2
25			695.5	684.4	673.8	663.6	654.0	644.9	636.2	627.9	620.1	612.6	605.5	598.7	592.2	586.0	580.1	574.4	569.1	563.9	558.9	554.2	549.6	545.2	540.9	536.9	533.0	529.2	525.6	522.1	518.7	515.4
28				798.7	786.1	773.8	762.5	751.4	741.0	730.9	721.6	712.5	703.9	695.7	687.9	680.4	673.2	666.4	659.9	653.6	647.6	641.8	636.3	631.0	625.9	621.0	616.3	611.7	607.3	603.1	599.0	595.1
30							839.5	827.3	815.4	804.1	793.4	783.3	773.8	764.5	755.6	747.1	739.0	731.3	723.9	716.8	710.0	703.5	697.3	691.3	685.5	680.0	674.7	669.5	664.6	659.8	655.2	650.8
32								906.8	893.8	881.3	869.2	858.0	847.0	836.6	826.5	817.0	807.9	799.2	790.9	782.9	775.4	768.0	761.0	754.3	747.8	741.5	735.5	729.8	724.2	718.9	713.7	708.7
34									976.4	962.3	949.0	936.1	924.0	912.2	901.2	890.6	880.5	870.6	861.3	852.2	843.6	835.5	827.7	820.0	812.8	805.8	799.0	792.6	786.3	780.3	774.5	769.0
36										1047.6	1032.7	1018.3	1004.9	992.0	979.6	967.6	956.4	945.4	935.0	924.8	915.2	906.1	897.2	888.8	880.5	872.8	865.3	857.9	851.0	844.3	837.9	831.6
38												1105.1	1089.9	1075.7	1061.7	1048.5	1035.9	1023.6	1012.1	1000.9	990.1	980.0	970.0	960.6	951.6	942.9	934.3	926.3	918.7	911.2	904.0	896.9
39												1149.8	1134.1	1119.0	1104.4	1090.4	1077.0	1064.4	1052.1	1040.4	1029.1	1018.1	1008.0	998.0	988.2	979.0	970.2	961.6	953.6	945.6	937.9	930.6
40												1195.9	1179.1	1163.2	1148.1	1133.3	1119.3	1105.9	1093.1	1080.5	1068.7	1057.2	1046.5	1036.0	1025.7	1016.0	1006.7	997.6	989.0	980.6	972.7	964.9
41												1195.3	1178.3	1161.7	1192.6	1177.3	1162.5	1148.6	1134.9	1121.8	1109.5	1097.4	1085.9	1074.6	1064.1	1053.9	1044.1	1034.5	1025.4	1016.8	1008.1	1000.1
42												1196.2	1178.6	1161.4	1190.4	1174.4	1159.1	1144.6	1130.6	1117.1	1150.9	1138.3	1126.3	1114.5	1103.4	1092.6	1082.1	1072.3	1062.7	1053.3	1044.3	1035.9
43												1198.2	1180.0	1206.4	1189.4	1172.9	1157.2	1142.1	1127.5	1158.8	1145.4	1132.2	1119.8	1107.7	1096.1	1085.0	1121.2	1110.6	1100.5	1090.9	1081.5	1072.2
44												1202.0	1182.9	1207.2	1189.6	1172.8	1156.6	1140.9	1126.0	1155.0	1141.0	1127.5	1114.6	1102.1	1135.8	1123.9	1112.8	1101.8	1091.4	1081.5	1071.7	1062.4
45														1168.8	1191.4	1174.1	1157.3	1141.3	1167.8	1152.9	1138.3	1124.5	1110.9	1098.2	1085.6	1117.3	1105.8	1094.6	1083.9	1073.3	1063.5	1053.6
46														1213.3	1194.6	1176.4	1159.4	1142.6	1167.5	1151.8	1136.9	1122.5	1108.7	1137.7	1124.6	1112.1	1100.0	1088.5	1077.4	1066.8	1056.4	1090.1
47														1180.4	1199.1	1180.6	1162.7	1145.6	1168.2	1152.3	1136.8	1122.1	1107.7	1094.0	1121.5	1108.3	1096.0	1084.1	1072.4	1061.5	1092.8	1082.4
48														1188.0	1204.9	1185.9	1167.5	1149.8	1132.7	1154.3	1138.2	1123.0	1108.2	1133.2	1119.3	1105.9	1093.2	1080.8	1068.8	1057.6	1086.9	1076.2
49														1197.0	1212.1	1192.3	1173.7	1155.1	1174.3	1157.2	1140.8	1125.0	1109.7	1095.2	1118.8	1104.8	1091.6	1078.8	1066.5	1093.9	1082.5	1071.1
50																1200.4	1180.9	1162.2	1143.9	1161.8	1144.7	1128.6	1112.7	1134.1	1119.3	1105.3	1091.4	1078.4	1065.5	1091.2	1079.1	1067.4
51																1209.5	1189.5	1169.9	1151.4	1167.6	1149.9	1133.3	1116.8	1101.1	1121.3	1106.7	1092.6	1078.9	1065.8	1089.5	1077.0	1065.4
52																1187.6	1199.1	1179.3	1160.0	1174.5	1156.5	1139.0	1122.3	1106.1	1124.7	1109.6	1094.9	1081.0	1067.3	1089.3	1076.6	1064.4
53																1200.3	1210.3	1189.7	1169.8	1150.7	1164.2	1146.2	1129.0	1112.4	1129.1	1113.7	1098.4	1083.9	1069.9	1056.6	1077.2	1064.7
54																1213.6	1192.2	1201.3	1180.9	1161.0	1142.2	1154.7	1136.9	1119.6	1103.0	1118.9	1103.3	1088.3	1074.0	1059.8	1078.9	1065.9
55																1228.0	1205.0	1185.1	1193.0	1172.9	1153.2	1134.2	1145.9	1128.0	1111.1	1094.5	1109.1	1093.8	1078.9	1064.5	1050.8	1037.3
56																1192.1	1169.4	1146.9	1124.5	1105.6	1086.1	1067.5	1049.2	1031.7	1014.9	998.9	983.3	968.2	1085.1	1070.2	1055.9	1072.3
57																1235.5	1212.8	1190.2	1167.7	1145.4	1126.2	1106.2	1087.3	1069.1	1051.6	1034.6	1018.3	1002.7	987.6	972.9	959.0	945.5
58																1254.4	1228.7	1206.3	1208.0	1188.6	1166.3	1147.1	1125.1	1107.4	1089.0	1071.5	1054.5	1038.0	1022.2	1007.1	992.2	978.2
59																1225.9	1225.3	1225.5	1200.2	1204.3	1182.1	1160.0	1165.1	1146.1	1127.6	1109.2	1091.5	1074.2	1057.7	1042.0	1026.6	1011.7
60																1182.2	1178.9	1154.2	1198.0	1173.1	1198.2	1176.2	1157.3	1135.5	1143.0	1121.3	1104.3	1111.5	1094.3	1077.8	1061.7	1046.2
61																1269.6	1244.6	1219.7	1194.9	1173.2	1148.6	1127.1	1105.7	1087.5	1066.3	1048.2	1030.3	1013.1	996.2	980.2	964.5	949.2
62																1270.8	1268.2	1240.4	1238.5	1213.7	1192.0	1167.4	1146.0	1124.7	1106.5	1085.3	1067.3	1049.4	1031.6	1014.6	998.2	982.3

Note(1): Maximum value for burnup of 62 GWd/MTU, 3.4 wt.% U-235, used only to determine the design basis source term.

Table P.5-24 HSM-H Total Dose Rates (mrem/hr) at Roof for 2.0 kW Case(1)

Burn-Up, GWD/MT	Maximum Assembly Average Initial U-235 Enrichment, wt %																														
	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	1.8	1.7	1.7	1.7	1.7	1.7	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.4	1.4	1.4	1.4	1.4
15	2.9	2.7	2.7	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.4	2.4	2.4	2.4	2.4	2.4	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.2	2.2	2.2	2.2	2.2
20	4.0	3.7	3.7	3.7	3.6	3.6	3.6	3.5	3.5	3.5	3.4	3.4	3.4	3.4	3.3	3.3	3.3	3.3	3.3	3.2	3.2	3.2	3.2	3.2	3.2	3.1	3.1	3.1	3.1	3.1	3.1	3.1
25			4.8	4.7	4.7	4.7	4.6	4.6	4.5	4.5	4.5	4.4	4.4	4.4	4.3	4.3	4.3	4.2	4.2	4.2	4.2	4.1	4.1	4.1	4.1	4.1	4.0	4.0	4.0	4.0	4.0	4.0
28				5.4	5.4	5.3	5.3	5.2	5.2	5.1	5.1	5.1	5.0	5.0	5.0	4.9	4.9	4.9	4.8	4.8	4.8	4.7	4.7	4.7	4.7	4.6	4.6	4.6	4.6	4.6	4.5	4.5
30							5.7	5.7	5.6	5.6	5.5	5.5	5.5	5.4	5.4	5.3	5.3	5.3	5.2	5.2	5.2	5.1	5.1	5.1	5.1	5.0	5.0	5.0	5.0	4.9	4.9	4.9
32								6.1	6.1	6.0	6.0	5.9	5.9	5.9	5.8	5.8	5.7	5.7	5.7	5.6	5.6	5.6	5.5	5.5	5.5	5.4	5.4	5.4	5.4	5.3	5.3	5.3
34									6.6	6.5	6.5	6.4	6.4	6.3	6.3	6.2	6.2	6.1	6.1	6.1	6.0	6.0	6.0	5.9	5.9	5.9	5.8	5.8	5.8	5.7	5.7	5.7
36										7.0	6.9	6.9	6.8	6.8	6.7	6.7	6.6	6.6	6.5	6.5	6.5	6.4	6.4	6.3	6.3	6.3	6.2	6.2	6.2	6.2	6.1	6.1
38												7.3	7.3	7.2	7.2	7.1	7.1	7.0	7.0	6.9	6.9	6.9	6.8	6.8	6.7	6.7	6.7	6.6	6.6	6.6	6.5	6.5
39												7.6	7.5	7.5	7.4	7.4	7.3	7.3	7.2	7.2	7.1	7.1	7.0	7.0	7.0	6.9	6.9	6.9	6.8	6.8	6.8	6.7
40												7.8	7.8	7.7	7.7	7.6	7.5	7.5	7.4	7.4	7.4	7.3	7.3	7.2	7.2	7.1	7.1	7.1	7.0	7.0	7.0	6.9
41												7.6	7.6	7.5	**7.9**	7.8	7.8	7.7	7.7	7.6	7.6	7.5	7.5	7.5	7.4	7.4	7.3	7.3	7.3	7.2	7.2	7.2
42												7.5	7.4	7.4	7.7	7.7	7.6	7.5	7.5	7.4	7.8	7.8	7.7	7.7	7.6	7.6	7.6	7.5	7.5	7.4	7.4	7.4
43												7.3	7.3	7.6	7.5	7.5	7.4	7.4	7.3	7.7	7.6	7.6	7.5	7.5	7.4	7.4	7.8	7.7	7.7	7.7	7.6	7.6
44												7.2	7.1	7.4	7.4	7.3	7.3	7.2	7.2	7.5	7.4	7.4	7.3	7.3	7.7	7.6	7.6	7.5	7.5	7.5	7.4	7.4
45														6.9	7.2	7.2	7.1	7.0	7.4	7.3	7.3	7.2	7.2	7.1	7.1	7.4	7.4	7.3	7.3	7.3	7.2	7.2
46														7.1	7.1	7.0	7.0	6.9	7.2	7.1	7.1	7.0	7.0	7.3	7.3	7.2	7.2	7.2	7.1	7.1	7.0	7.4
47														6.7	6.9	6.9	6.8	6.8	7.0	7.0	6.9	6.9	6.8	6.8	7.1	7.1	7.0	7.0	6.9	6.9	7.2	7.2
48														6.5	6.8	6.7	6.7	6.6	6.6	6.8	6.8	6.7	6.7	7.0	6.9	6.9	6.9	6.8	6.8	6.7	7.0	7.0
49														6.4	6.7	6.6	6.6	6.5	6.8	6.7	6.7	6.6	6.6	6.5	6.8	6.7	6.7	6.7	6.6	6.9	6.9	6.8
50																6.5	6.4	6.4	6.3	6.6	6.5	6.5	6.4	6.7	6.6	6.6	6.6	6.5	6.5	6.8	6.7	6.7
51																6.4	6.3	6.3	6.2	6.5	6.4	6.4	6.3	6.3	6.5	6.5	6.4	6.4	6.3	6.6	6.6	6.5
52																6.0	6.2	6.2	6.1	6.3	6.3	6.2	6.2	6.1	6.4	6.3	6.3	6.3	6.2	6.5	6.4	6.4
53																5.9	6.1	6.1	6.0	6.0	6.2	6.1	6.1	6.0	6.3	6.2	6.2	6.1	6.1	6.0	6.3	6.2
54																5.9	5.8	6.0	5.9	5.9	5.8	6.0	6.0	5.9	5.9	6.1	6.1	6.0	6.0	5.9	6.2	6.1
55																5.8	5.7	5.7	5.9	5.8	5.8	5.7	5.9	5.8	5.8	5.8	6.0	5.9	5.9	5.8	5.8	5.8
56																5.3	5.3	5.2	5.2	5.1	5.1	5.0	5.0	5.0	4.9	4.9	4.8	4.8	5.8	5.7	5.7	5.9
57																5.4	5.4	5.4	5.3	5.3	5.2	5.2	5.1	5.1	5.0	5.0	5.0	4.9	4.9	4.8	4.8	4.8
58																5.4	5.3	5.3	5.4	5.4	5.3	5.3	5.2	5.2	5.2	5.1	5.1	5.0	5.0	5.0	4.9	4.9
59																5.0	5.1	5.2	5.2	5.3	5.3	5.2	5.4	5.3	5.3	5.2	5.2	5.2	5.1	5.1	5.0	5.0
60																4.6	4.6	4.6	5.0	4.9	5.2	5.2	5.1	5.1	5.2	5.2	5.1	5.3	5.2	5.2	5.2	5.1
61																4.9	4.9	4.8	4.8	4.8	4.7	4.7	4.6	4.6	4.5	4.5	4.5	4.4	4.4	4.4	4.3	4.3
62																4.8	4.9	4.8	4.9	4.9	4.8	4.8	4.7	4.7	4.7	4.6	4.6	4.5	4.5	4.5	4.4	4.4

Notes:

(1) Maximum value for burnup of 41 GWd/MTU, 3.3 wt.% U-235, used only to determine the design basis source term.

(2) The side dose rates increase by 13% when loading 0.380 MTU FAs

Table P.5-25 OS197FC TC Total Dose Rates (mrem/hr) at Cask Centerline for 1.5 kW Case(1)

Burn-Up, GWD/MTU	Maximum Assembly Average Initial U-235 Enrichment, wt %																															
	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	258.4	235.0	231.2	227.6	224.3	221.1	218.2	215.4	212.8	210.3	208.0	205.7	203.6	201.6	199.7	197.9	196.2	194.6	193.0	191.5	190.1	188.7	187.3	186.1	184.9	183.7	182.6	181.5	180.4	179.4	178.4	177.5
15	409.4	373.3	367.4	361.7	356.4	351.4	346.7	342.2	338.0	333.9	330.1	326.5	323.1	319.8	316.7	313.7	310.9	308.2	305.6	303.1	300.7	298.5	296.3	294.2	292.2	290.2	288.3	286.5	284.8	283.1	281.5	279.9
20	580.7	529.6	521.0	512.8	505.1	497.8	490.9	484.4	478.2	472.2	466.6	461.3	456.3	451.4	446.8	442.4	438.2	434.2	430.4	426.7	423.2	419.8	416.5	413.4	410.4	407.5	404.8	402.1	399.5	397.0	394.6	392.2
25			695.5	684.4	673.8	663.6	654.0	644.9	636.2	627.9	620.1	612.6	605.5	598.7	592.2	586.0	580.1	574.4	569.1	563.9	558.9	554.2	549.6	545.2	540.9	536.9	533.0	529.2	525.6	522.1	518.7	515.4
28				798.7	786.1	773.8	762.5	751.4	741.0	730.9	721.6	712.5	703.9	695.7	687.9	680.4	673.2	666.4	659.9	653.6	647.6	641.8	636.3	631.0	625.9	621.0	616.3	611.7	607.3	603.1	599.0	595.1
30							839.5	827.3	815.4	804.1	793.4	783.3	773.8	764.5	755.6	747.1	739.0	731.3	723.9	716.8	710.0	703.5	697.3	691.3	685.5	680.0	674.7	669.5	664.6	659.8	655.2	650.8
32								906.8	893.8	881.3	869.2	858.0	847.0	836.6	826.5	817.0	807.9	799.2	790.9	782.9	775.4	768.0	761.0	754.3	747.8	741.5	735.5	729.8	724.2	718.9	713.7	708.7
34									901.6	888.2	875.4	862.8	886.3	875.0	864.0	853.5	843.5	833.8	824.7	816.0	843.6	835.5	827.7	820.0	812.8	805.8	799.0	792.6	786.3	780.3	774.5	769.0
36										898.3	884.5	871.3	858.6	846.5	867.3	856.1	845.4	835.0	825.1	815.7	840.4	831.6	823.0	814.9	806.9	799.5	792.2	785.1	813.6	807.0	800.5	794.4
38												882.9	869.3	856.2	843.7	831.6	850.4	839.3	828.7	818.3	808.8	799.4	821.6	812.7	804.4	796.2	788.5	781.0	773.6	799.3	792.4	785.7
39												861.6	875.8	862.4	849.2	836.9	825.1	842.7	831.7	821.0	811.1	801.0	822.1	813.0	804.3	795.7	787.7	779.8	772.4	796.6	789.3	782.5
40												870.2	855.9	869.4	855.9	842.9	830.5	818.9	835.6	824.5	814.0	803.9	794.3	784.9	804.8	796.2	787.8	779.6	771.8	764.5	757.4	780.4
41												879.9	865.0	850.6	863.3	849.8	837.0	824.8	813.0	828.9	818.0	807.3	797.4	787.7	778.5	797.4	788.6	780.4	772.3	764.6	757.1	778.8
42												865.7	874.9	860.2	845.7	832.3	844.3	831.8	819.4	807.9	822.6	811.9	801.4	791.3	781.8	799.4	790.4	781.8	773.3	765.3	757.5	750.1
43												877.8	861.9	870.4	855.7	841.4	827.9	839.2	826.6	814.4	803.0	817.1	806.2	795.8	785.6	776.2	766.9	758.1	775.3	767.0	758.8	751.1
44												890.5	874.1	858.4	843.2	851.5	837.4	824.1	811.0	822.1	810.1	798.6	787.2	800.9	790.7	780.6	771.0	761.9	777.9	769.3	761.0	753.0
45														871.0	855.2	840.2	825.7	834.1	820.6	807.7	817.9	806.1	794.4	783.6	772.9	785.8	775.9	766.5	757.2	748.3	739.9	755.5
46														863.2	847.3	852.6	837.5	823.2	830.7	817.5	804.5	792.2	802.4	790.9	779.9	769.3	781.7	771.6	762.3	753.1	744.1	758.9
47														877.8	861.4	845.4	829.9	835.4	820.9	807.3	814.7	802.1	789.7	799.0	787.7	776.6	766.2	755.9	746.1	758.4	749.1	740.5
48														874.0	857.2	859.6	843.8	828.5	833.2	819.3	805.5	792.5	799.7	787.5	775.7	784.8	773.8	763.2	753.1	743.4	754.9	745.7
49														872.5	855.2	856.2	840.0	842.7	827.6	812.9	817.6	804.0	791.1	778.4	785.5	773.9	762.7	751.9	760.9	750.9	741.1	731.6
50																854.8	838.2	839.5	823.9	826.6	812.1	798.3	803.0	790.0	777.4	765.6	753.9	761.0	750.4	739.9	748.7	739.0
51																855.7	854.9	838.5	822.4	823.8	808.9	811.5	797.9	784.5	789.1	776.7	764.7	753.1	742.3	749.0	738.9	729.0
52																858.3	856.2	839.3	838.6	822.8	807.7	809.1	794.7	797.7	784.4	771.5	759.4	764.3	752.9	741.9	731.0	720.9
53																862.6	859.2	841.9	840.2	823.9	808.3	793.1	793.8	779.7	781.8	768.8	756.1	759.5	747.9	736.4	741.7	730.9
54																868.4	850.0	846.1	829.0	826.6	810.6	809.6	794.6	780.1	780.8	767.3	754.4	756.7	744.6	748.3	737.0	726.0
55																874.1	856.8	852.0	834.2	817.3	814.5	798.7	796.8	796.0	781.5	767.8	768.7	755.7	743.4	745.7	734.2	723.1
56																870.1	863.9	846.5	841.0	823.4	819.4	803.5	800.8	785.5	784.1	769.9	756.1	756.4	743.6	744.9	732.9	721.5
57																871.1	861.2	854.7	837.6	831.0	813.7	809.3	793.3	790.1	775.1	773.2	759.2	758.2	745.3	732.7	733.4	721.6
58																880.4	860.4	843.5	833.3	826.5	822.9	804.9	800.1	796.2	780.7	765.9	763.6	749.5	748.4	735.3	734.9	722.8
59																871.7	864.0	853.4	846.1	826.3	819.2	802.6	786.1	791.8	776.2	772.2	768.9	754.6	741.0	739.2	738.0	714.2
60																877.0	865.9	854.9	847.1	827.6	820.0	812.6	796.2	789.1	773.7	768.8	753.8	749.9	746.8	733.1	730.9	718.2
61																880.3	868.8	849.5	841.2	830.1	822.1	802.9	795.1	790.5	783.1	767.3	752.2	737.4	743.2	739.5	725.9	713.1
62																877.2	865.3	853.6	845.1	826.0	814.5	806.3	798.1	790.1	774.2	766.4	761.5	746.2	741.3	727.3	723.1	710.1

Notes*:*
(1) Maximum value for burnup of 32 GWd/MTU, 2.6 wt.% U-235, used only to determine the design basis source term.

All changes on this page are Amd 15.

Table P.5-26 HSM-H Total Dose Rates (mrem/hr) at Roof for 1.5 kW Case[(1)]

Burn-Up, GWD/MT	Maximum Assembly Average Initial U-235 Enrichment, wt %																														
	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	1.8	1.7	1.7	1.7	1.7	1.7	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.4	1.4	1.4	1.4	1.4
15	2.9	2.7	2.7	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.4	2.4	2.4	2.4	2.4	2.4	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.2	2.2	2.2	2.2	2.2
20	4.0	3.7	3.7	3.7	3.6	3.6	3.6	3.5	3.5	3.5	3.4	3.4	3.4	3.4	3.3	3.3	3.3	3.3	3.3	3.2	3.2	3.2	3.2	3.2	3.2	3.1	3.1	3.1	3.1	3.1	3.1	3.1
25			4.8	4.7	4.7	4.7	4.6	4.6	4.5	4.5	4.5	4.4	4.4	4.4	4.3	4.3	4.3	4.2	4.2	4.2	4.2	4.1	4.1	4.1	4.1	4.1	4.0	4.0	4.0	4.0	4.0	4.0
28				5.4	5.4	5.3	5.3	5.2	5.2	5.1	5.1	5.1	5.0	5.0	5.0	4.9	4.9	4.9	4.8	4.8	4.8	4.7	4.7	4.7	4.7	4.6	4.6	4.6	4.6	4.6	4.5	4.5
30							5.7	5.7	5.6	5.6	5.5	5.5	5.5	5.4	5.4	5.3	5.3	5.3	5.2	5.2	5.2	5.1	5.1	5.1	5.1	5.0	5.0	5.0	5.0	4.9	4.9	4.9
32								**6.1**	6.1	6.0	6.0	5.9	5.9	5.9	5.8	5.8	5.7	5.7	5.7	5.6	5.6	5.6	5.5	5.5	5.5	5.4	5.4	5.4	5.4	5.3	5.3	5.3
34									5.9	5.8	5.8	5.8	6.0	6.0	5.9	5.9	5.8	5.8	5.8	5.7	6.0	6.0	6.0	5.9	5.9	5.9	5.8	5.8	5.8	5.7	5.7	5.7
36										5.7	5.6	5.6	5.5	5.5	5.7	5.7	5.6	5.6	5.6	5.5	5.8	5.7	5.7	5.7	5.6	5.6	5.6	5.5	5.8	5.8	5.8	5.8
38												5.4	5.4	5.3	5.3	5.2	5.4	5.4	5.4	5.3	5.3	5.2	5.5	5.5	5.4	5.4	5.4	5.3	5.3	5.6	5.5	5.5
39												5.1	5.3	5.2	5.2	5.1	5.1	5.3	5.3	5.2	5.2	5.2	5.4	5.4	5.3	5.3	5.3	5.2	5.2	5.4	5.4	5.4
40												5.0	5.0	5.2	5.1	5.1	5.0	5.0	5.2	5.2	5.1	5.1	5.0	5.0	5.2	5.2	5.2	5.1	5.1	5.1	5.0	5.3
41												5.0	4.9	4.9	5.1	5.0	5.0	4.9	4.9	5.1	5.0	5.0	5.0	4.9	4.9	5.1	5.1	5.0	5.0	5.0	4.9	5.2
42												4.7	4.9	4.8	4.8	4.7	4.9	4.9	4.8	4.8	5.0	4.9	4.9	4.8	4.8	5.0	5.0	4.9	4.9	4.9	4.9	4.8
43												4.7	4.6	4.8	4.7	4.7	4.6	4.8	4.8	4.7	4.7	4.9	4.8	4.8	4.7	4.7	4.7	4.6	4.8	4.8	4.8	4.7
44												4.6	4.6	4.5	4.5	4.6	4.6	4.6	4.5	4.7	4.6	4.6	4.5	4.7	4.7	4.6	4.6	4.6	4.8	4.7	4.7	4.7
45														4.5	4.5	4.4	4.4	4.5	4.5	4.4	4.6	4.5	4.5	4.5	4.4	4.6	4.5	4.5	4.5	4.5	4.4	4.6
46														4.3	4.3	4.4	4.3	4.3	4.4	4.4	4.3	4.3	4.4	4.4	4.4	4.3	4.5	4.5	4.4	4.4	4.4	4.5
47														4.3	4.2	4.2	4.2	4.3	4.2	4.2	4.3	4.3	4.2	4.4	4.3	4.3	4.3	4.2	4.2	4.3	4.3	4.3
48														4.1	4.1	4.2	4.1	4.1	4.2	4.2	4.1	4.1	4.2	4.2	4.1	4.2	4.2	4.2	4.2	4.1	4.3	4.2
49														4.0	3.9	4.0	4.0	4.1	4.0	4.0	4.1	4.0	4.0	4.0	4.1	4.1	4.0	4.0	4.1	4.1	4.0	4.0
50																3.9	3.8	3.9	3.9	4.0	3.9	3.9	4.0	4.0	3.9	3.9	3.9	4.0	3.9	3.9	4.0	4.0
51																3.7	3.8	3.8	3.7	3.8	3.8	3.9	3.8	3.8	3.9	3.9	3.8	3.8	3.8	3.9	3.8	3.8
52																3.6	3.7	3.7	3.7	3.7	3.7	3.7	3.7	3.8	3.7	3.7	3.7	3.8	3.7	3.7	3.7	3.7
53																3.5	3.6	3.6	3.6	3.6	3.5	3.5	3.6	3.5	3.6	3.6	3.6	3.6	3.6	3.6	3.7	3.6
54																3.4	3.4	3.5	3.4	3.5	3.4	3.5	3.5	3.4	3.5	3.5	3.4	3.5	3.5	3.6	3.5	3.5
55																3.4	3.3	3.4	3.3	3.3	3.4	3.3	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4
56																3.2	3.3	3.2	3.3	3.2	3.3	3.2	3.3	3.3	3.3	3.3	3.3	3.3	3.3	3.3	3.3	3.3
57																3.1	3.1	3.2	3.1	3.2	3.1	3.2	3.1	3.2	3.2	3.2	3.2	3.2	3.2	3.2	3.2	3.2
58																3.0	3.0	3.0	3.0	3.0	3.1	3.0	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1
59																2.9	2.9	2.9	3.0	2.9	2.9	2.9	2.9	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.1	3.0
60																2.8	2.8	2.8	2.9	2.8	2.8	2.9	2.8	2.9	2.8	2.9	2.8	2.9	2.9	2.9	2.9	2.9
61																2.7	2.7	2.7	2.7	2.7	2.8	2.7	2.7	2.8	2.8	2.8	2.7	2.7	2.8	2.8	2.8	2.8
62																2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.7

Notes:

(1) Maximum value for burnup of 32 GWd/MTU, 2.6 wt.% U-235, used only to determine the design basis source term.

(2) The side dose rates increase by 13% when loading 0.380 MTU FAs.

Table P.5-27
Summary of Experimental Samples as a Function of Burnup Range

		Low Burnup (B < 45 GWd/MTU)		*High Burnup (B > 45 GWd/MTU)*	
Power Plant	***Reference***	***No of Samples***	***Range (GWd/MTU)***	***No of Samples***	***Range (GWd/MTU)***
Takahama-3	*NUREG/CR-6968 Reference [5.16]*	*14*	*8.55 – 42.16*	*2*	*47.03 – 47.25*
Three Mile Island - 1		*10*	*22.80 – 44.8*	*9*	*50.10 – 55.70*
Calvert Cliffs		*3*	*27.35 – 44.34*	-	-
Vandellós II	*NUREG/CR-7013 Reference [5.18]*	*1*	*42.50*	*5*	*54.85 – 78.30*
Gösgen	*NUREG/CR-7012 Reference [5.17]*	*1*	*31.10*	*5*	*46.00 –70.30*
GKN II		-	-	*1*	*54.10*
Total		*29*	-	*22*	-

Number of Samples Used for Each Isotope

	Cm-242	***Cm-244***	***Cs-134***	***Cs-137***	***Ce-144***	***Eu-154***	***Ru-106***	***Sm-147***	***Sr-90***
Gosgen	*4*	*6*	*6*	*6*	*6*	*6*	*6*	*6*	*6*
GKN II	*1*	*1*	-	-	*1*	*1*	-	*1*	-
Takahama 3	*16*	*16*	*16*	*16*	*16*	*16*	*16*	*6*	-
Vandellos II	-	*6*	*6*	*6*	*5*	*5*	*5*	*5*	-
TMI	*8*	*8*	*8*	*19*	-	-	-	*19*	-
Calvert Cliffs	-	-	*3*	*3*	-	*3*	-	*3*	*3*

Table P.5-28
HLZC#2 2.0 kW Design Basis HSM Source Term

Bounding Source at 380 kgU/FA: 29 GWD/MTU, 1.1 wt. %, after 2.0 years of cooling						
E_{min} MeV	***to***	***E_{max} MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	*to*	*5.00e-02*	*7.979E+11*	*3.367E+15*	*1.030E+12*	*2.258E+11*
5.00e-02	*to*	*1.00e-01*	*6.588E+10*	*1.098E+15*	*1.424E+11*	*4.455E+10*
1.00e-01	*to*	*2.00e-01*	*3.658E+10*	*1.030E+15*	*4.780E+10*	*1.085E+10*
2.00e-01	*to*	*3.00e-01*	*2.459E+09*	*2.819E+14*	*2.810E+09*	*5.416E+08*
3.00e-01	*to*	*4.00e-01*	*1.118E+10*	*2.198E+14*	*8.852E+09*	*7.031E+08*
4.00e-01	*to*	*6.00e-01*	*1.114E+11*	*1.651E+15*	*7.460E+10*	*9.290E+08*
6.00e-01	*to*	*8.00e-01*	*7.863E+10*	*2.429E+15*	*5.581E+10*	*8.950E+08*
8.00e-01	*to*	*1.00e+00*	*9.535E+11*	*5.177E+14*	*1.754E+11*	*5.430E+11*
1.00e+00	*to*	*1.33e+00*	*1.864E+13*	*3.495E+14*	*4.118E+13*	*1.295E+13*
1.33e+00	*to*	*1.66e+00*	*5.263E+12*	*1.126E+14*	*1.163E+13*	*3.658E+12*
1.66e+00	*to*	*2.00e+00*	*2.094E+07*	*7.543E+12*	*4.424E+07*	*1.358E+07*
2.00e+00	*to*	*2.50e+00*	*1.261E+08*	*1.869E+13*	*2.783E+08*	*8.753E+07*
2.50e+00	*to*	*3.00e+00*	*1.077E+05*	*5.670E+11*	*2.378E+05*	*7.478E+04*
3.00e+00	*to*	*4.00e+00*	*6.601E-06*	*5.213E+10*	*3.379E-05*	*5.438E-06*
4.00e+00	*to*	*5.00e+00*	*0.0*	*1.222E+07*	*0.0*	*0.0*
5.00e+00	*to*	*6.50e+00*	*0.0*	*4.905E+06*	*0.0*	*0.0*
6.50e+00	*to*	*8.00e+00*	*0.0*	*9.622E+05*	*0.0*	*0.0*
8.00e+00	*to*	*1.00e+01*	*0.0*	*2.043E+05*	*0.0*	*0.0*
*Total Gamma, g/(sec*FA)*			*2.596E+13*	*1.108E+16*	*5.435E+13*	*1.744E+13*
*Total Neutrons, n/(sec*FA)*			*3.505E+8*			
This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for subcritical multiplication and an axial variation of burn-up profile in active fuel region, where the dry k_{eff}= 0.29663 and bpf=1.289.						

Table P.5-29
HLZC#2 2.0 kW Design Basis TC Source Term

Bounding Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 2.9 years of cooling						
E_{min}, MeV	***to***	***E_{max}, MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	*to*	*5.00e-02*	*7.245E+11*	*2.449E+15*	*1.067E+12*	*2.630E+11*
5.00e-02	*to*	*1.00e-01*	*7.597E+10*	*7.568E+14*	*1.647E+11*	*5.170E+10*
1.00e-01	*to*	*2.00e-01*	*4.032E+10*	*6.771E+14*	*5.400E+10*	*1.255E+10*
2.00e-01	*to*	*3.00e-01*	*2.499E+09*	*1.911E+14*	*3.012E+09*	*6.208E+08*
3.00e-01	*to*	*4.00e-01*	*7.126E+09*	*1.436E+14*	*6.453E+09*	*8.113E+08*
4.00e-01	*to*	*6.00e-01*	*1.176E+11*	*1.737E+15*	*7.677E+10*	*9.104E+07*
6.00e-01	*to*	*8.00e-01*	*6.347E+10*	*3.128E+15*	*4.825E+10*	*1.339E+09*
8.00e-01	*to*	*1.00e+00*	*5.207E+11*	*6.585E+14*	*9.766E+10*	*2.967E+11*
1.00e+00	*to*	*1.33e+00*	*2.166E+13*	*3.890E+14*	*4.776E+13*	*1.505E+13*
1.33e+00	*to*	*1.66e+00*	*6.117E+12*	*1.238E+14*	*1.349E+13*	*4.249E+12*
1.66e+00	*to*	*2.00e+00*	*8.849E+05*	*4.851E+12*	*1.874E+06*	*5.758E+05*
2.00e+00	*to*	*2.50e+00*	*1.464E+08*	*9.446E+12*	*3.227E+08*	*1.017E+08*
2.50e+00	*to*	*3.00e+00*	*1.251E+05*	*3.785E+11*	*2.757E+05*	*8.687E+04*
3.00e+00	*to*	*4.00e+00*	*1.162E-05*	*3.516E+10*	*5.946E-05*	*9.570E-06*
4.00e+00	*to*	*5.00e+00*	*0.0*	*4.486E+07*	*0.0*	*0.0*
5.00e+00	*to*	*6.50e+00*	*0.0*	*1.800E+07*	*0.0*	*0.0*
6.50e+00	*to*	*8.00e+00*	*0.0*	*3.532E+06*	*0.0*	*0.0*
8.00e+00	*to*	*1.00e+01*	*0.0*	*7.500E+05*	*0.0*	*0.0*
*Total Gamma, g/(sec*FA)*			*2.933E+13*	*1.027E+16*	*6.276E+13*	*1.992E+13*
*(1)Total Neutrons, n/(sec*FA)*			*1.300E+9*			

(1) This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for subcritical multiplication and an axial variation of burn-up profile in active fuel region, where the dry k_{eff}= 0.25189 (wet k_{eff} = 0.9412) and bpf=1.152.

Table P.5-30
HLZC#6 0.6 kW (Inner) Design Basis HSM and TC Source Term

Bounding Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 18.2 years of cooling						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle (g/s)	In-core (g/s)	Plenum (g/s)	Top Nozzle (g/s)
0.00e+00	to	5.00e-02	5.740E+10	4.802E+14	1.251E+11	3.717E+10
5.00e-02	to	1.00e-01	1.002E+10	1.283E+14	2.209E+10	6.944E+09
1.00e-01	to	2.00e-01	2.877E+09	8.598E+13	5.638E+09	1.679E+09
2.00e-01	to	3.00e-01	1.542E+08	2.526E+13	2.915E+08	8.450E+07
3.00e-01	to	4.00e-01	2.665E+08	1.578E+13	4.195E+08	1.099E+08
4.00e-01	to	6.00e-01	2.452E+09	2.152E+13	1.615E+09	8.754E+06
6.00e-01	to	8.00e-01	2.834E+09	1.075E+15	8.752E+09	1.323E+09
8.00e-01	to	1.00e+00	1.635E+09	1.622E+13	7.925E+09	1.365E+09
1.00e+00	to	1.33e+00	2.910E+12	5.198E+13	6.416E+12	2.022E+12
1.33e+00	to	1.66e+00	8.218E+11	1.067E+13	1.812E+12	5.709E+11
1.66e+00	to	2.00e+00	7.753E+01	4.080E+10	5.053E+01	9.583E-03
2.00e+00	to	2.50e+00	1.966E+07	2.441E+09	4.335E+07	1.366E+07
2.50e+00	to	3.00e+00	1.680E+04	1.938E+08	3.704E+04	1.167E+04
3.00e+00	to	4.00e+00	8.411E-06	7.505E+07	4.305E-05	6.928E-06
4.00e+00	to	5.00e+00	0.0	2.498E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	1.002E+07	0.0	0.0
6.50e+00	to	8.00e+00	0.0	1.967E+06	0.0	0.0
8.00e+00	to	1.00e+01	0.0	4.176E+05	0.0	0.0
Total Gamma, g/(sec*FA)			3.810E+12	1.911E+15	8.399E+12	2.641E+12
(1)Total Neutrons, n/(sec*FA)			7.219E+8			

(1) *This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for subcritical multiplication and an axial variation of burn-up profile in active fuel region, where the dry k_{eff}= 0.25189 and bpf=1.152.*

Table P.5-31
HLZC#6 0.6 kW (Outer) Design Basis HSM Source Term

Bounding Source at 380 kgU/FA: 10 GWD/MTU, 0.7 wt. %, after 2.6 years of cooling						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle (g/s)	In-core (g/s)	Plenum (g/s)	Top Nozzle (g/s)
0.00e+00	to	5.00e-02	3.353E+11	1.155E+15	4.650E+11	1.092E+11
5.00e-02	to	1.00e-01	3.167E+10	3.754E+14	6.894E+10	2.158E+10
1.00e-01	to	2.00e-01	1.573E+10	3.562E+14	2.187E+10	5.249E+09
2.00e-01	to	3.00e-01	9.954E+08	9.515E+13	1.230E+09	2.596E+08
3.00e-01	to	4.00e-01	3.636E+09	7.480E+13	3.128E+09	3.388E+08
4.00e-01	to	6.00e-01	4.322E+10	3.657E+14	2.836E+10	1.019E+08
6.00e-01	to	8.00e-01	2.461E+10	6.218E+14	1.791E+10	3.673E+08
8.00e-01	to	1.00e+00	3.522E+11	8.571E+13	6.303E+10	2.005E+11
1.00e+00	to	1.33e+00	9.027E+12	1.339E+14	2.000E+13	6.279E+12
1.33e+00	to	1.66e+00	2.549E+12	3.967E+13	5.647E+12	1.773E+12
1.66e+00	to	2.00e+00	1.810E+06	2.395E+12	3.939E+06	1.225E+06
2.00e+00	to	2.50e+00	6.100E+07	6.913E+12	1.351E+08	4.242E+07
2.50e+00	to	3.00e+00	5.212E+04	1.736E+11	1.154E+05	3.625E+04
3.00e+00	to	4.00e+00	1.581E-06	1.585E+10	8.093E-06	1.302E-06
4.00e+00	to	5.00e+00	0.0	3.253E+05	0.0	0.0
5.00e+00	to	6.50e+00	0.0	1.305E+05	0.0	0.0
6.50e+00	to	8.00e+00	0.0	2.558E+04	0.0	0.0
8.00e+00	to	1.00e+01	0.0	5.430E+03	0.0	0.0
Total Gamma, g/(sec*FA)			1.238E+13	3.312E+15	2.631E+13	8.389E+12
(1)Total Neutrons, n/(sec*FA)			9.414E+6			

(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for subcritical multiplication and an axial variation of burn-up profile in active fuel region, where the dry k_{eff}= 0.39328 and bpf=1.414

Table P.5-32
HLZC#6 1.3 kW Design Basis HSM Source Term

Bounding Source at 380 kgU/FA: 16 GWD/MTU, 0.8 wt. %, after 2.0 years of cooling						
E_{min}, MeV	***to***	***E_{max}, MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	*to*	*5.00e-02*	*5.902E+11*	*2.485E+15*	*7.482E+11*	*1.611E+11*
5.00e-02	*to*	*1.00e-01*	*4.707E+10*	*8.179E+14*	*1.019E+11*	*3.184E+10*
1.00e-01	*to*	*2.00e-01*	*2.519E+10*	*7.833E+14*	*3.357E+10*	*7.759E+09*
2.00e-01	*to*	*3.00e-01*	*1.739E+09*	*2.098E+14*	*2.000E+09*	*3.884E+08*
3.00e-01	*to*	*4.00e-01*	*9.149E+09*	*1.653E+14*	*7.084E+09*	*5.033E+08*
4.00e-01	*to*	*6.00e-01*	*7.461E+10*	*8.956E+14*	*5.043E+10*	*8.321E+08*
6.00e-01	*to*	*8.00e-01*	*5.937E+10*	*1.276E+15*	*4.148E+10*	*5.499E+08*
8.00e-01	*to*	*1.00e+00*	*7.623E+11*	*2.279E+14*	*1.400E+11*	*4.341E+11*
1.00e+00	*to*	*1.33e+00*	*1.331E+13*	*2.294E+14*	*2.946E+13*	*9.255E+12*
1.33e+00	*to*	*1.66e+00*	*3.758E+12*	*7.094E+13*	*8.318E+12*	*2.613E+12*
1.66e+00	*to*	*2.00e+00*	*1.849E+07*	*5.509E+12*	*3.975E+07*	*1.230E+07*
2.00e+00	*to*	*2.50e+00*	*8.999E+07*	*1.549E+13*	*1.991E+08*	*6.253E+07*
2.50e+00	*to*	*3.00e+00*	*7.688E+04*	*4.031E+11*	*1.701E+05*	*5.343E+04*
3.00e+00	*to*	*4.00e+00*	*3.224E-06*	*3.684E+10*	*1.650E-05*	*2.656E-06*
4.00e+00	*to*	*5.00e+00*	*0.0*	*1.805E+06*	*0.0*	*0.0*
5.00e+00	*to*	*6.50e+00*	*0.0*	*7.241E+05*	*0.0*	*0.0*
6.50e+00	*to*	*8.00e+00*	*0.0*	*1.420E+05*	*0.0*	*0.0*
8.00e+00	*to*	*1.00e+01*	*0.0*	*3.015E+04*	*0.0*	*0.0*
*Total Gamma, g/(sec*FA)*			*1.864E+13*	*7.183E+15*	*3.890E+13*	*1.251E+13*
*(1)Total Neutrons, n/(sec*FA)*			*5.189E+7*			
(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for a subcritical multiplication and an axial variation of burn-up profile in active fuel region, where the dry k_{eff}= 0.35650 and bpf=1.414.						

All changes on this page are Amd 15.

Table P.5-33
HLZC#6 2.5 kW Design Basis HSM and TC Source Term

Bounding Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 2.3 years of cooling						
E_{min}, MeV	***to***	***E_{max}, MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	*to*	*5.00e-02*	*8.819E+11*	*3.355E+15*	*1.215E+12*	*2.836E+11*
5.00e-02	*to*	*1.00e-01*	*8.229E+10*	*1.065E+15*	*1.779E+11*	*5.581E+10*
1.00e-01	*to*	*2.00e-01*	*4.538E+10*	*9.708E+14*	*5.950E+10*	*1.356E+10*
2.00e-01	*to*	*3.00e-01*	*2.898E+09*	*2.722E+14*	*3.385E+09*	*6.722E+08*
3.00e-01	*to*	*4.00e-01*	*9.961E+09*	*2.082E+14*	*8.445E+09*	*8.770E+08*
4.00e-01	*to*	*6.00e-01*	*1.367E+11*	*2.229E+15*	*8.976E+10*	*3.549E+08*
6.00e-01	*to*	*8.00e-01*	*7.832E+10*	*3.565E+15*	*5.792E+10*	*1.341E+09*
8.00e-01	*to*	*1.00e+00*	*8.332E+11*	*8.069E+14*	*1.529E+11*	*4.745E+11*
1.00e+00	*to*	*1.33e+00*	*2.338E+13*	*4.443E+14*	*5.154E+13*	*1.624E+13*
1.33e+00	*to*	*1.66e+00*	*6.602E+12*	*1.460E+14*	*1.456E+13*	*4.586E+12*
1.66e+00	*to*	*2.00e+00*	*7.166E+06*	*7.276E+12*	*1.496E+07*	*4.567E+06*
2.00e+00	*to*	*2.50e+00*	*1.580E+08*	*1.527E+13*	*3.483E+08*	*1.097E+08*
2.50e+00	*to*	*3.00e+00*	*1.350E+05*	*5.631E+11*	*2.976E+05*	*9.376E+04*
3.00e+00	*to*	*4.00e+00*	*1.177E-05*	*5.215E+10*	*6.023E-05*	*9.694E-06*
4.00e+00	*to*	*5.00e+00*	*0.0*	*4.604E+07*	*0.0*	*0.0*
5.00e+00	*to*	*6.50e+00*	*0.0*	*1.848E+07*	*0.0*	*0.0*
6.50e+00	*to*	*8.00e+00*	*0.0*	*3.625E+06*	*0.0*	*0.0*
8.00e+00	*to*	*1.00e+01*	*0.0*	*7.697E+05*	*0.0*	*0.0*
*Total Gamma, g/(sec*FA)*			*3.205E+13*	*1.309E+16*	*6.786E+13*	*2.166E+13*
*(1)Total Neutrons, n/(sec*FA)*			*1.335E+9*			
(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-keff) to account for a subcritical multiplication and an axial variation of burn-up profile in active fuel region, where the dry keff= 0.25189 and bpf=1.152						

Table P.5-34
HLZC#6 0.6 kW (Outer) Design Basis TC Source Term

Bounding Source at 380 kgU/FA: 31 GWD/MTU, 1.1 wt. %, after 6.6 years of cooling						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle (g/s)**	**In-core (g/s)**	**Plenum (g/s)**	**Top Nozzle (g/s)**
0.00e+00	to	5.00e-02	2.545E+11	6.054E+14	4.619E+11	1.301E+11
5.00e-02	to	1.00e-01	3.709E+10	1.628E+14	8.114E+10	2.547E+10
1.00e-01	to	2.00e-01	1.572E+10	1.282E+14	2.397E+10	6.172E+09
2.00e-01	to	3.00e-01	9.254E+08	3.622E+13	1.289E+09	3.061E+08
3.00e-01	to	4.00e-01	2.181E+09	2.393E+13	2.317E+09	4.002E+08
4.00e-01	to	6.00e-01	3.619E+10	2.738E+14	2.361E+10	2.671E+07
6.00e-01	to	8.00e-01	1.997E+10	1.244E+15	1.791E+10	9.437E+08
8.00e-01	to	1.00e+00	2.524E+10	1.236E+14	1.046E+10	1.471E+10
1.00e+00	to	1.33e+00	1.068E+13	1.568E+14	2.359E+13	7.420E+12
1.33e+00	to	1.66e+00	3.015E+12	4.243E+13	6.660E+12	2.095E+12
1.66e+00	to	2.00e+00	5.068E+01	3.544E+11	3.546E+01	1.084E+00
2.00e+00	to	2.50e+00	7.214E+07	4.472E+11	1.594E+08	5.014E+07
2.50e+00	to	3.00e+00	6.163E+04	2.582E+10	1.362E+05	4.284E+04
3.00e+00	to	4.00e+00	6.589E-06	2.438E+09	3.373E-05	5.428E-06
4.00e+00	to	5.00e+00	0.0	1.257E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	5.047E+06	0.0	0.0
6.50e+00	to	8.00e+00	0.0	9.900E+05	0.0	0.0
8.00e+00	to	1.00e+01	0.0	2.102E+05	0.0	0.0
Total Gamma, g/(sec*FA)			1.408E+13	2.798E+15	3.087E+13	9.694E+12
(1)Total Neutrons, n/(sec*FA)			3.605E+8			
(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for a subcritical multiplication and an axial variation of burn-up profile in active fuel region, where the dry k_{eff}= 0.28950 and bpf=1.266						

Table P.5-35
HLZC#6 1.3 kW Design Basis TC Source Term

Bounding Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 4.6 years of cooling						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle (g/s)	In-core (g/s)	Plenum (g/s)	Top Nozzle (g/s)
0.00e+00	to	5.00e-02	4.723E+11	1.250E+15	7.888E+11	2.121E+11
5.00e-02	to	1.00e-01	6.083E+10	3.513E+14	1.324E+11	4.159E+10
1.00e-01	to	2.00e-01	2.914E+10	2.984E+14	4.132E+10	1.009E+10
2.00e-01	to	3.00e-01	1.757E+09	8.428E+13	2.260E+09	4.995E+08
3.00e-01	to	4.00e-01	4.391E+09	5.889E+13	4.319E+09	6.530E+08
4.00e-01	to	6.00e-01	7.735E+10	8.964E+14	5.044E+10	4.328E+07
6.00e-01	to	8.00e-01	4.190E+10	2.331E+15	3.421E+10	1.335E+09
8.00e-01	to	1.00e+00	1.382E+11	3.795E+14	3.231E+10	7.909E+10
1.00e+00	to	1.33e+00	1.744E+13	2.845E+14	3.844E+13	1.211E+13
1.33e+00	to	1.66e+00	4.924E+12	8.354E+13	1.086E+13	3.420E+12
1.66e+00	to	2.00e+00	2.479E+03	1.565E+12	5.201E+03	1.592E+03
2.00e+00	to	2.50e+00	1.178E+08	2.450E+12	2.597E+08	8.184E+07
2.50e+00	to	3.00e+00	1.007E+05	1.225E+11	2.219E+05	6.992E+04
3.00e+00	to	4.00e+00	1.121E-05	1.150E+10	5.738E-05	9.234E-06
4.00e+00	to	5.00e+00	0.0	4.195E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	1.684E+07	0.0	0.0
6.50e+00	to	8.00e+00	0.0	3.303E+06	0.0	0.0
8.00e+00	to	1.00e+01	0.0	7.013E+05	0.0	0.0
Total Gamma, g/(sec*FA)			2.319E+13	6.022E+15	5.038E+13	1.588E+13
(1)Total Neutrons, n/(sec*FA)			1.214E+9			

(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by $bpf/(1-k_{eff})$ to account for a subcritical multiplication and an axial variation of burn-up profile in active fuel region, where the dry k_{eff}= 0.25189 and bpf=1.152

P.9 Acceptance Tests and Maintenance Program

Background for this particular UFSAR chapter:

Beginning with CoC 1004 Amendment 10, which was incorporated into UFSAR Revision 11, Chapter P.9, "Acceptance Tests and Maintenance Program," contained information which was incorporated by reference into the Technical Specifications (TS) associated with a particular amendment. It is known that certain general licensees reconcile the CoC 1004 UFSAR revisions provided to them to their loaded systems, pursuant to 10 CFR 72.48 and 10 CFR 72.212. In doing so they sometimes find the changed UFSAR portions incorporated by reference into the TS to be impossible to reconcile because the 10 CFR 72.48 regulation does not allow proposed activities which involve changes to the TS.

In order to facilitate this reconciliation process by general licensees, the following statements are provided, addressing the licensing basis for certain amendments, as they relate to certain UFSAR chapters which contain TS incorporated by reference. Additionally, so that the actual information is contained in the current CoC 1004 UFSAR, to facilitate the reconciliation by general licensees, the UFSAR Revision 11, 12, 13, and 14 versions of Chapter P.9 are inserted and annotated in this part of the UFSAR. For clarity, this includes annotating the version of Chapter P.9 directly associated with the latest UFSAR revision in which a change to Chapter P.9 occurred.

- Systems loaded to CoC 1004 Amendment 10 have Technical Specifications incorporated by reference from UFSAR Revisions 11 and 12 in Chapter P.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to systems loaded to Amendment 10.
- Systems loaded to CoC 1004 Amendment 11 have Technical Specifications incorporated by reference from UFSAR Revision 13 Chapter P.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 11.
- Note that CoC 1004 Amendment 12 was submitted and docketed, associated with a U.S. Department of Energy project, but due to a lack of review funding the NRC returned it without a review.
- Systems loaded to CoC 1004 Amendment 13 have Technical Specifications incorporated by reference from UFSAR Revisions 14 and 15 Chapter P.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 13.
- Systems loaded to CoC 1004 Amendment 14 have Technical Specifications incorporated by reference by FCN 721004-1575, which will be incorporated into UFSAR Revisions 16 and 17 Chapter P.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 14.
- *Systems loaded to CoC 1004 Amendment 15 have Technical Specifications incorporated by reference from UFSAR Revision 18 Chapter P.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 15.*

This version of Chapter P.9 is associated with CoC 1004 Amendment 15 and is added from UFSAR Revision 18. Please see Page P.9 Introduction - 1 for a discussion as to why certain versions of Chapter P.9 are being maintained in the UFSAR.

P.9 Acceptance Tests and Maintenance Program

P.9.1 Acceptance Tests

The acceptance requirements for the NUHOMS®-24PTH system are given in the UFSAR except as described in the following sections. The NUHOMS®-24PTH DSC has been enhanced to provide leaktight confinement and the basket includes an updated poison plate design. The requirements for the poison plate material acceptance tests and the NUHOMS®-24PTH DSC welds for the 24PTH system are described.

P.9.1.1 Visual Inspection

Visual examinations are performed at the fabricator's facility to ensure that the NUHOMS®-24PTH system components conform to the fabrication specifications and drawings.

P.9.1.2 Structural Tests

The NUHOMS®-24PTH DSC confinement welds are designed, fabricated, tested and inspected in accordance with ASME B&PV Code Section III, Subsection NB [9.1] with exceptions as listed in Section P.3.1. The following requirements are unique to the NUHOMS®-24PTH DSC:

- The inner bottom cover weld is inspected in accordance with Article NB-5231 when the weld joint design is per Figure NB-4243-1,
- The outer bottom cover weld is penetrant tested, and
- The outer top cover plate weld root and cover are penetrant tested.

The NUHOMS®-24PTH DSC basket is designed, fabricated, and inspected in accordance with ASME B&PV Code Section III, Subsection NG [9.1] with exceptions as listed in Section P.3.1.

P.9.1.3 Leak Tests

The NUHOMS®-24PTH DSC confinement boundary is leak tested to verify that it is leaktight in accordance with the criteria of ANSI N14.5 [9.2]. The personnel performing the leak test are qualified in accordance with SNT-TC-1A [9.8].

The leak tests are typically performed using the helium mass spectrometer method. Alternative methods are acceptable, provided that the required sensitivity is achieved.

P.9.1.4 Component Tests

The NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. No other components of the NUHOMS® system require testing, except as discussed in this chapter.

All changes on this page are AMD 11

P.9.1.5 Shielding Integrity Tests

The transfer cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6"x 6" grid, the detector will encompass a 6"x 6" square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed.

The gamma and neutron shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

P.9.1.6 Thermal Acceptance Tests

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section P.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section P.9.1.7.6.

P.9.1.7 Poison Acceptance

CAUTION

Sections P.9.1.7.1 through P.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated *a*luminum

(b) Boron carbide/*a*luminum metal matrix composite (MMC)

(c) BORAL®

This version of Chapter P.9 is associated with CoC 1004 Amendment 15 and is added from UFSAR Revision 18. Please see Page P.9 Introduction - 1 for a discussion as to why certain versions of Chapter P.9 are being maintained in the UFSAR.

All changes on this page are AMD 13

The 24PTH DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content for these materials is given in Table P.9-1.

References to metal matrix composites throughout this chapter are not intended to refer to BORAL®, which is described later in this section.

P.9.1.7.1 Borated Aluminum

See the Caution in Section P.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating primarily as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy (other boron compounds, such as AlB_{12}, can also occur). For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section P.9.1.7.7. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

P.9.1.7.2 Boron Carbide / Aluminum Metal Matrix Composites (MMC)

See the Caution in Section P.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, *molten metal infiltration,* or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding *or produced by molten metal infiltration* shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than

0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B_4C particles in MMCs shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 24PTH DSC, MMCs shall pass the qualification testing specified in Section P.9.1.7.8, and shall subsequently be subject to the process controls specified in Section P.9.1.7.9.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section P.9.1.7.7. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

All changes on this page are AMD 13

P.9.1.7.3 BORAL®

See the Caution in Section P.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns. The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of BORAL®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken from the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

P.9.1.7.4 Visual Inspections of Neutron Absorbers

See the Caution in Section P.9.1.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. *Blisters shall be treated as non-conforming. For clad MMCs and for BORAL®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet.* Material that does not meet *these* acceptance criteria shall be reworked, repaired, or scrapped.

P.9.1.7.5 Other Visual Inspections Criteria (non-Technical Specifications)

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

P.9.1.7.6 Thermal Conductivity Testing of Poison Plates

*Acceptance t*esting shall conform to ASTM E1225[1], ASTM E1461[2], or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, and may be reduced if the first five tests meet the specified minimum thermal conductivity. *For cast products, the lot shall be defined by the heat or ingot. For other products, the lot shall be defined as material produced in a single production campaign using the same heat or lots of aluminum and boron carbide feed materials.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After *twenty five* tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The measured thermal conductivity values shall satisfy the minimum required conductivities as specified in Section P.4.3.

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section P.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

[1] ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"

[2] ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"

P.9.1.7.7 Specification for Acceptance Testing of Neutron Absorber *Content*

Acceptance testing for neutron absorber content shall be performed by either neutron transmission or by B-10 volume density measurement.

CAUTION

Portions of Section P.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

P.9.1.7.7.1 *Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission*

a) Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.1 inch diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard. Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.

This version of Chapter P.9 is associated with CoC 1004 Amendment 15 and is added from UFSAR Revision 18. Please see Page P.9 Introduction - 1 for a discussion as to why certain versions of Chapter P.9 are being maintained in the UFSAR.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided

tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor for a normal distribution with 95% probability and 95% confidence [9.6].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than the statistically derived minimum thickness from P.9.1.7.7 a) or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

P.9.1.7.7.2 *Specification for Acceptance Testing of Neutron Absorbers by B10 Volume Density Measurement*

a) B-10 volume density measurement acceptance testing procedures shall be subject to approval by the certificate holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, as long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for B-10 volume density measurements shall be such that there is at least one density measurement for each 2000 square inches of final product in each lot.

Areal density is determined by measuring the B-10 volume density in test samples and converting the measured values to B-10 areal density. The method of measurement of B-10 volume density as well as the conversion method shall be subject to approval by the certificate holder. The method of conversion shall be qualified by performing benchmarking with both neutron transmission and B-10 volume density measurement to confirm transmissibility between methods. The B-10 volume density shall be checked against the minimum areal density by multiplying the volume realized by the coupon thickness.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B-10 areal densities are determined by volume density as described above. The lower tolerance limit of B-10 volume density is then determined, defined as the mean value of B-10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6]. Finally, the minimum specified value of B-10 areal density is divided by the lower tolerance limit of B-10 volume density to arrive at the minimum plate thickness that provides the specified B-10 areal density.

Any plate that is thinner than the statistically derived minimum thickness from P.9.1.7.7.2 a) or the minimum design thickness, whichever is greater, shall be treated as nonconforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

P.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

CAUTION

Portions of Section P.9.1.7.8.4 and Section P.9.1.7.8.5 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

This version of Chapter P.9 is associated with CoC 1004 Amendment 15 and is added from UFSAR Revision 18. Please see Page P.9 Introduction - 1 for a discussion as to why certain versions of Chapter P.9 are being maintained in the UFSAR.

P.9.1.7.8.1 Applicability and Scope

Metal matrix composites (MMCs) acceptable for use in the 24PTH DSC are described in Section P.9.1.7.2.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section P.9.1.7.9 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the Certificate Holder.

P.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in SectionP.9.1.7.8.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section P.9.1.7.8.5.

P.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for unclad MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for full density MMCs (clad or unclad) consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

P.9.1.7.8.4 **Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity**

At least three samples, one each from approximately the two ends and middle of the qualification material run shall be subject to:

a) **room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:**
   - **Minimum yield strength, 0.2% offset: 1.5 ksi**
   - **Minimum ultimate strength: 5 ksi**

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[3] Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998.

[4] Boralyn testing submitted to the NRC under docket 71-1027, 1998.

[5] ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.

- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

b) Testing to verify more than 98% of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a means to be approved by the Certificate Holder. The maximum interconnected porosity is 0.5 volume %.

c) Delamination Testing of Clad MMC

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage. An example of such a test would be: (1) immerse a specimen at least 6 x 6 inches in water under pressure ≥30 psig for at least 24 hours, (2) place the specimen in a vacuum furnace preheated to at least 300°F and evacuate the furnace. Acceptance criterion: no blistering or delamination of the cladding.

P.9.1.7.8.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

Uniformity of the boron distribution shall be verified either by:

a) Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

b) Quantitative testing for the B10 areal density, B10 density, the boron carbide weight fraction, *or the boron weight fraction,* on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section P.9.1.7.7, or by chemical analysis for boron carbide *or boron* content in the composite.

P.9.1.7.8.6 Qualification Report

Qualification report shall be prepared by, or subject to approval by the Certificate Holder.

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6 ASTM E290, Standard Methods for Bend Testing of Materials for Ductility
7 ASTM E94, Recommended Practice for Radiographic Testing
8 ASTM E142, Controlling Quality of Radiographic Testing
9 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

P.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION

Sections P.9.1.7.9.1 and P.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

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P.9.1.7.9.1 **Applicability and Scope**

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section P.9.1.7.8 is required, depending on the characteristics of the material that could be affected by the process change.

P.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, reduce corrosion resistance, reduce the mechanical strength or ductility of the MMC.

P.9.1.7.9.3 Identification and Control of Key Process Changes

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section P.9.1.7.9.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that are established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average *(d50)* particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product,

e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a magnesium-alloyed aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, and

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

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P.9.2 Maintenance Program

NUHOMS®-24PTH system is a totally passive system and therefore requires little, if any, maintenance over the lifetime of the ISFSI. Typical NUHOMS®-24PTH system maintenance tasks are performed in accordance with the *U*FSAR.

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This version of Chapter P.9 is associated with CoC 1004 Amendment 15 and is added from UFSAR Revision 18. Please see Page P.9 Introduction - 1 for a discussion as to why certain versions of Chapter P.9 are being maintained in the UFSAR.

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P.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition including 2000 addenda.

9.2 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.3 *Deleted.*

9.4 *Deleted.*

9.5 "Aluminum Standards and Data, 2003" The Aluminum Association.

9.6 Natrella, "Experimental Statistics," Dover, 2005.

9.7 *Deleted.*

9.8 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

9.9 *Deleted.*

9.10 *Deleted.*

This version of Chapter P.9 is associated with CoC 1004 Amendment 15 and is added from UFSAR Revision 18. Please see Page P.9 Introduction - 1 for a discussion as to why certain versions of Chapter P.9 are being maintained in the UFSAR.

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Table P.9-1
B10 Specification for the NUHOMS®-24PTH Poison Plates

Poison Type	24PTH Basket Type	Minimum Poison Loading (B10 mg/cm^2)	% Credit Used in Criticality Analysis
Borated Aluminum /MMC	1A or 2A	7	90
	1B or 2B	15	
	1C or 2C	32	
BORAL®	1A or 2A	9	75
	1B or 2B	19	
	1C or 2C	40	

P.10.1 Occupational Exposure

The occupational exposure results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section P.5.4.11. Loading 0.380 MTU fuel results in an increase in occupational exposure of 10%.

The expected occupational dose for placing a canister of spent fuel into dry storage is based on the operational steps outlined in Table 7.4-1. The total exposure for the occupational dose due to placing a single NUHOMS®-24PTH-L DSC into storage is conservatively estimated to be 4.4 person-rem. This value bounds the exposure for loading either a 24PTH-S or 24PTH-S-LC DSC into storage. This is a very conservative estimate because the dose rates on and around the 24PTH DSC's used in these calculations are based on very conservative assumptions for the design-basis source terms and analyses models (Configuration 2 from Section P.2). The calculated exposures are due mainly to the expected gamma dose rate during preparation for welding.

The NUHOMS®-24PTH System loading operations, the number of workers required for each operation, and the amount of time required for each operation are presented in Table P.10-1. This information is used as the basis for estimating the total occupational exposure associated with one fuel load. This evaluation is performed for the storage of one design-basis NUHOMS®-24PTH-L DSC in an HSM-H. The loading operations are identical for the 24PTH-S and 24PTH-S-LC DSC. The dose rates applicable for each operation are based on the results presented in Section P.5.4 for loading operations. Engineering judgment and operational experience are used to estimate dose rates that were not explicitly evaluated. This evaluation assumes that a transfer trailer/skid with an integral ram is used for the DSC transfer operations. Licensees may elect to use different equipment and/or different procedures. Each Licensee must evaluate any such changes in accordance with its ALARA program.

Unique steps are sometimes necessary at the individual site to load the canister, complete closure operations and place the canister in the HSM. Specifically, the licensee may choose to modify the sequence of operations in order to achieve reduced dose rates for a larger number of steps, with the end result of reduced total exposure. The only requirement is that the licensee practice ALARA with respect to the total exposure received for a loading campaign. These estimated durations, manloading and dose rates are not limits.

The amount of time required to complete some operations as identified in Table P.10-1 may be greater than the actual amount of time spent in a radiation field. The process of vacuum drying the DSC includes setting up the vacuum drying system (VDS), verifying that the VDS is operating correctly, evacuating the DSC cavity, monitoring the DSC pressure, and disconnecting the VDS from the DSC. Of these tasks, only setup and removal of the VDS require a worker to spend time near the DSC. The most time consuming task, evacuating the DSC, does not require anyone to be present near DSC at all. The total exposure calculated for each task is therefore not necessarily equal to the number of workers multiplied by the total time required, multiplied by a dose rate. The exposure estimation for each task correctly accounts for cases such as vacuum drying assumes that good ALARA practices are followed.

The results of the evaluations of the 24PTH-L are presented in Table P.10-1.

P.10.2.2 Dose Rates

The dose rate and annual dose results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section P.5.4.11. Loading 0.380 MTU fuel results in an increase in annual dose of 13%.

Dose rates are calculated for distances of 6.1 meters (20 feet) to 600 meters from the edges of the two ISFSI designs. The HSM is modeled in MCNP as a box, representing the HSM arrays.

Neutron and gamma-ray sources are placed on each HSM, with shield walls, surface using the spectra and activities determined above. The angular distribution of source particles is modeled as a cosine distribution. The contribution of capture gamma-rays has been neglected, as has the contribution of bremsstrahlung electrons. The inclusion of coherent scattering greatly increases the variance in a problem with point detector tallies without improving the accuracy of the calculation. Thus, coherent scattering of photons is ignored.

The MCNP models of the ISFSI layouts are described herein. For the 2x10 back-to-back array of HSM-Hs with end shield walls, the "box" dimensions are as follows. The total width is 1260 cm. The length of the "box" is 3129 cm and the height of the "box" is 564 cm.

For the two 1x10 front-to-front arrays of HSM-Hs with end and back shield walls, the "box" dimensions for each array are as follows. The total width is 721 cm. The length of the "box" is 3129 cm and the height of the "box" is 564 cm. The two 1x10 arrays are 1026 cm (34 feet) apart.

For the 2x10 back-to-back array of HSM-Model 102s with end shield walls, the "box" dimensions are as follows. The total width is 1158 cm. The length of the "box" is 3220 cm and the height of the "box" is 457 cm.

For the two 1x10 front-to-front arrays of HSM-Model 102s with end and back shield walls, the "box" dimensions for each array are as follows. The total width is 640 cm. The length of the "box" is 3220 cm and the height of the "box" is 457 cm. The two 1x10 arrays are 1066 cm (35 feet) apart.

Point detectors are placed at the following locations as measured from each face of the "box": 6.095 m (20 feet), 10 m, 20 m, 30 m, 40 m, 50 m, 60 m, 70 m, 80 m, 90 m, 100 m, 200 m, 300 m, 400 m, 500 m, and 600 m. Each point detector is placed 91.4 cm (3 feet) above the ground.

The MCNP results for each detector from the front of 2x10 back-to-back array are summarized in Table P.10-8 and Table P.10-11 for the HSM-H and HSM-Model 102, respectively. The MCNP results as a function of distance from the back of the two 1x10 front-to-front arrays are summarized in Table P.10-9 and Table P.10-12 for the HSM-H and HSM-Model 102, respectively. The MCNP results as a function of distance from the side of the 2x10 back-to-back array and the two 1x10 front-to-front arrays are summarized in Table P.10-10 and Table P.10-13 for the HSM-H and HSM-Model 102, respectively. The results from Table P.10-8, Table P.10-9 and Table P.10-10 are plotted in Figure P.10-1 for the HSM-H. The results from Table P.10-11, Table P.10-12, and Table P.10-13 are plotted in Figure P.10-2 for the HSM-Model 102.

The preceding analyses and the results provided in Figure P.10-1 and Figure P.10-2 are intended to provide typical dose rates for the generic ISFSI layouts described in Section P.10.2. They may not be applicable to an actual ISFSI. The written evaluations performed by a licensee for an

P.11.2 Postulated Accidents

P.11.2.1 Reduced HSM Air Inlet and Outlet Shielding

P.11.2.1.1 Cause of Accident

No change to the cause for HSM Model 102 described in Section 8.2.1.1.

For the HSM-H/HSM-HS, this accident is not credible since the array of HSM-Hs/HSM-HSs is designed with the elimination of 6-inch gaps between the adjacent HSM-Hs. The HSM-Hs/HSM-HSs are placed next to each other and even in the unlikely event of large settlement of the ISFSI foundation, shifting of adjacent HSM-Hs/HSM-HSs occurring and causing the HSM-Hs/HSM-HSs to separate is not credible.

P.11.2.1.2 Accident Analysis

There are no structural consequences that affect the safe operation of the NUHOMS®-24PTH *System* resulting from the separation of the HSM Model 102. The thermal effects of this accident result from the blockage of HSM Model 102 air inlet and outlet openings. However, the effect on the NUHOMS®-24PTH-S-LC DSC, HSM Model 102 and fuel temperatures is bounded by the complete blockage of air inlet and outlet openings described in Section P.11.2.7. The radiological consequences of this accident are described in the paragraph below.

P.11.2.1.3 Accident Dose Calculations

The off-site radiological effects that result from a partial loss of adjacent HSM Model 102 shielding is an increase in the air scattered (skyshine) and direct doses from the assumed *12*-inch gap between the separated HSMs. The air scattered (skyshine) and direct doses are reduced from the gap between the HSM Model 102s that are in contact with each other. On-site radiological effects result from an increase in the direct radiation during recovery operations and increased skyshine radiation. Table 8.2-2 shows the comparisons of the increased dose rate as a function of distance due to the reduced shielding effects of the adjacent HSMs for the 24P DSC with 5-year cooled design basis fuel. Table P.11-1 provides a similar table for 24PTH-S-LC DSC of the NUHOMS®-24PTH System. For the NUHOMS®-24PTH System, the dose received by a person located 100 meters away from the NUHOMS® installation for eight hours a day for five days (estimated recovery time) would be *less than 10* mrem. The increased dose to an off-site person for 24 hours a day for five days located 600 meters away would be *less than 0.1* mrem. Thus, the 10 CFR *Part* 72 requirements for this postulated event are met.

P.11.2.1.4 Corrective Actions

No change to Section 8.2.1.4.

$\cos(24.65+\alpha-90) = 0.00433 + 0.907411 = 0.911741$

$90-\alpha = 24.85-24.25 = 0.60$

Therefore, a loaded HSM-H rotates a maximum of 0.60° from vertical. The loaded HSM-H is stable against overturning as tip-over does not occur until the CG rotates past the edge point (point B, Figure P.11-1) to an angle of more than 24.65° [= $\tan^{-1}(52.0/118.77)$].

P.11.2.3.3 Accident Dose Calculations

The NUHOMS®-24PTH DSC is designed and tested as a leak-tight containment boundary according to the criteria of ANSI N14.5. As shown in Section P.11.1.1.1, the tornado wind and tornado missiles do not breach the containment boundary. Therefore, there is no increase in site boundary dose *as a result of leakage from the DSC* due to this accident event. *However, a small increase in the site dose rate may occur due to damage to the HSM-H.*

*The increase in the dose rates at the localized impact location following the missile impact accident is expected to be bounded by the dose rates at the HSM-H vents, calculated to be 1400 mrem/hour in Table P.5-1 (1237 mrem/hr*1.13 ~ 1400 mrem/hr), since the structural analysis results demonstrate that there is no full penetration. This represents an increase in the roof centerline dose rate by a factor greater than 20 and is conservative.*

For the purpose of this calculation, it is conservatively assumed that the affected area is twice the area of impact ~ 1.6 ft². The approximate surface areas at the HSM-H front is 140 ft², at the HSM-H roof is 200 ft² and that at the HSM-H side is 280 ft². The impact area, therefore, represents approximately 0.6% to 1.2% of the surface area of the HSM-H, and the average dose rate on the surface of the impacted HSM will not increase appreciably. This increase does not significantly affect the ISFSI site dose rates and the results from Section P.10.2 for a 2x10 array of undamaged HSMs (specifically Table P.10-11) can be utilized to determine the exposure from a damaged HSM. This method is conservative because the missile impact will affect at most a single HSM, while a 2x10 array has approximately 20 front and 20 roof vents. The total dose rate is then the dose rate of the damaged HSM summed with the dose rate of the undamaged HSMs in the array, or twice the dose rate of the undamaged array using the conservative assumptions outlined above.

*The dose received by a person located 100 meters away from the ISFSI for the assumed 8-hour duration would be less than 5 mrem (2*8 hours*Model 102 dose rate at 100m, 1.09E-01 mrem/hour*1.13 scaling factor) with a 2x10 array of HSMs. As an additional conservatism, Model 102 dose rates are used because these dose rates bound HSM-H dose rates. The dose to an offsite person located 500 meters away for the assumed 8-hour duration would be less than 0.02 mrem (2*8 hours*Model 102 dose rate at 500m, 7.43E-04 mrem/hour*1.13 scaling factor) with a 2x10 array of HSMs.*

P.11.2.4 Flood

P.11.2.4.1 Cause of Accident

No change to Section 8.2.4.1.

P.11.2.4.2 Accident Analysis

No change to the HSM Model 102 analysis presented in Section 8.2.4.2.

The HSM-H and DSCs are evaluated for flooding in Section P.3.7.3. The DSC is designed and tested to be leak tight to the criteria of ANSI N14.5. The stresses in the DSC due to the design basis flood are well below the allowable stresses for Service Level C of the ASME Code Subsection NB [11.5]. Therefore, the NUHOMS®-24PTH DSC will withstand the design basis flood without breach of the confinement boundary.

The evaluation of the HSM-H/HSM-HS for flooding, presented in Section U.3.7.3, is not changed.

P.11.2.4.3 Accident Dose Calculations

The radiation dose due to flooding of the HSMs (HSM-H or HSM Model 102) is negligible. The NUHOMS®-24PTH DSC is designed and tested as a leak-tight containment boundary. Flooding does not breach the containment boundary. Therefore radioactive material inside the DSC will

The section below describes the additional analyses performed to demonstrate the acceptability of the system with the NUHOMS®-24PTH DSC.

P.11.2.7.3 Accident Dose Calculations

There are no off-site dose consequences as a result of this accident. The only significant dose increase is that related to the recovery operation. Based on the results presented in Appendix P.5, Table P.5-1 and Table P.5-2, the bounding average dose on HSM front or roof is *36.4* mrem/hr and *53.7* mrem/hr for the 24PTH-S or –L DSC in HSM-H and 24PTH-S-LC DSC in HSM Model 102 respectively.

It is conservatively estimated that the on-site workers will receive an additional dose of no more than *430* mrem during the eight hour period it is estimated may be required for removal of debris from the inlet and outlet vent openings.

Table P.11-1
Comparison of Total Dose Rates for HSM-H Loaded with 24PTH-S-LC, with and without Adjacent HSM Shielding Effects

Distance from Nearest HSM Wall, 2x10 Array (meters)	Normal Case Dose Rate[(1)] (mrem/hr)	Accident Case Dose Rate[(1)] (mrem/hr)
10	6.4	12.7
100	0.1	0.2
500	7.4×10^{-4}	1.5×10^{-3}
600	3.1×10^{-4}	6.2×10^{-4}

(1) Air scattered plus direct radiation. *Dose rates to be scaled by a factor of 1.13 to account for the effect of 0.380 MTU loading.*

Table P.11-2
Calculated Accident Dose Rates on the Side of the OS197FC TC and Standardized TC

Distance	Neutron(1)		Total Gamma(1)		Total(1)	
	mRem/hr	1σ error	mRem/hr	1σ error	mRem/hr	1σ error
OS197FC TC						
1 meter	3.19E+03	0.0124	3.10E+02	0.0011	3.505E+03	0.0128
100 meter	5.10E-01	0.0134	1.50E-01	0.0366	6.61E-01	0.0124
500 meter	4.10E-04	0.0305	4.95E-04	0.8299	9.05E-04	0.0194
Standardized TC						
1 meter	4.18E+02	0.0219	3.44E+02	0.0018	7.62E+02	0.0258
100 meter	6.76E-02	0.0232	1.67E-01	0.0872	2.35E-01	0.0308
500 meter	5.34E-05	0.0341	6.20E-04	1.1638	6.74E-04	0.0279

(1) Dose rates to be scaled by a factor of 1.13 to account for the effect of 0.380 MTU loading.

APPENDIX T
NUHOMS® 61BTH DSC

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T.1.1 Introduction

The NUHOMS®-61BTH *System* is designed to store up to 61 intact (including reconstituted) or up to 16 damaged with up to 4 *failed fuel cans* (FFCs) loaded with failed fuel with the remainder intact BWR fuel assemblies with or without fuel channels. Alternatively, 61 damaged fuels can be stored in the NUHOMS®-61BTH DSC as shown in Figure T.2-9. The fuel to be stored is limited to a maximum initial lattice average initial enrichment of 5.0 wt. %, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of 3.0 years. The design characteristics, including physical and radiological parameters of the payload, are described in Appendix T.2.

Reconstituted assemblies containing up to 10 replacement stainless steel rods per assembly or up to 61 lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods are acceptable for storage in 61BTH DSC as intact fuel assemblies with a slightly longer cooling time than that required for a standard assembly. The maximum number of reconstituted fuel assemblies per DSC is four with stainless steel rods or 61 with UO_2 rods.

Provisions have been made for storage of up to 61 damaged fuel assemblies in *lieu of an equal number of intact assemblies in cells located at the outer edge of* the 61BTH basket. Damaged BWR fuel assemblies are assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. *The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means and the retrievability is ensured following the normal and off-normal conditions. The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is ensured following normal and off-normal conditions.* The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to ensure retrievability.

Provisions have also been made for storage of up to four failed fuel assemblies in the corner cells, along with up to 12 damaged fuel assemblies in the cells located at the outer periphery of the 61BTH basket and balance intact as described in Appendix T.2.

The NUHOMS®-61BTH System consists of the following new or modified components:

- A 61BTH DSC, with two alternate configurations, designated as Type 1 61BTH DSC or Type 2 61BTH, is described in detail in Section T.1.2. It provides confinement, an inert environment, structural support, heat rejection, and criticality control for the 61 BWR fuel assemblies,
- A modified HSM-H module, as described in Section T.1.2, or HSM Model 80/102/152/202, with no modifications to the configuration as described in UFSAR Chapter 1, is provided for environmental protection, shielding and heat rejection during storage,
- An OS197 or OS197H TC with no modifications to the configuration as described in UFSAR Chapter 1, or a modified version of the OS197FC TC, designated as OS197FC-B, described in Section T.1.2, is provided for onsite transfer of the 61BTH DSCs,
- An upgraded version of the HSM-H, designated as HSM-HS, is provided to allow storage of the NUHOMS®-61BTH DSC in locations where higher seismic levels exist. The HSM-HS design configuration, described in Appendix U.1, is modified to accommodate the smaller diameter of the NUHOMS®-61BTH DSC, and

T.2.1 Spent Fuel To Be Stored

As described in Appendix T.1, there are two alternate design configurations for the NUHOMS®-61BTH DSC; Type 1 and Type 2. Each of the DSC configurations is designed to store intact (including reconstituted) and/or damaged BWR fuel assemblies as specified in Table T.2-1 and Table T.2-2. The fuel to be stored is limited to a maximum lattice average initial enrichment of 5.0 wt. % ^{235}U. The maximum allowable fuel assembly average burnup is limited to 62 GWd/MTU. The minimum required cooling time for fuel to be stored with 170 kgU/FA and 198 kgU/FA is explicitly specified as a function of burnup and enrichment in Tables T.2-5 through T.2-10 and T.2-16 *through T.2-18*. For fuel with a kgU/FA loading between these two values, the minimum required cooling time for fuel to be stored as a function of burnup and enrichment is determined by using the interpolation methodology specified in the notes and examples following Table *T.2-18. Cooling times for burnup / enrichment combinations in the Not-Analyzed area of the Fuel Qualification Tables can be determined using the methodology specified in the notes and examples following Table T.2-18.*

The NUHOMS®-61BTH DSC is also authorized to store fuel assemblies containing blended low enriched uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Reconstituted fuel assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or 61 lower enrichment UO_2 rods instead of zircaloy clad enriched UO_2 rods are acceptable for storage in 61BTH DSCs as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel rods or 61 with UO_2 rods or Zr rods or Zr pellets or unirradiated stainless steel rods.

The NUHOMS®-61BTH DSCs can also accommodate up to a maximum of 61 damaged fuel assemblies placed in the fuel compartments located in accordance with Figure T.2-9. When loaded with failed fuel, the damaged fuel assemblies are to be loaded as shown in Figure T.2-9. Damaged BWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. *The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means.* The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is assured following normal and off-normal conditions. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

The NUHOMS®-61BTH Type 2 DSC, when used with the top grid assembly (Alternate 1) design, is also able to accommodate up to a maximum of four failed fuel assemblies encapsulated in individual failed fuel cans and placed in cells located at the outer edge of the DSC as shown in Figure T.2-9. Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Failed fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means.

Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a rod storage basket are also considered as failed fuel. Loose fuel debris, not contained in a rod storage basket may also be placed in a failed fuel can for storage, provided the size of the debris is larger than the failed fuel can screen mesh opening and it is located at least 10 in. above the top of the bottom shield plug of the DSC.

Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its contents shall be less than 705 lbs.

A 61BTH DSC containing less than 61 fuel assemblies may contain dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly.

The NUHOMS®-61BTH Type 1 DSC may store up to 61 BWR fuel assemblies arranged in any of the five alternate heat load zoning configurations shown in Figure T.2-1 through Figure T.2-4 and Figure T.2-10 with a maximum decay heat of 0.54 kW per assembly while restricting the maximum canister heat load to 22.0 kW. The NUHOMS®-61BTH Type 2 DSCs may store up to 61 BWR fuel assemblies arranged in any of the ten alternate heat load zoning configurations shown in Figure T.2-1 through Figure T.2-8, Figure T.2-10 and Figure T.2-11 with a maximum decay heat of *1.2* kW per assembly and a maximum heat load of 31.2 kW per canister.

The NUHOMS®-61BTH DSC is designed with six alternate basket configurations based on the boron content in the poison plates as listed in Table T.2-3 or Table T.2-4 or Table T.2-4a (designated as "A" for the poison plates with the lowest B-10 loading to "F" for the highest B-10 loading). Three alternate poison materials are allowed: (a) borated aluminum alloy, (b) a boron carbide/aluminum metal matrix composite (MMC), or (c) Boral®. For criticality analysis, 90% of the B-10 content present in the borated aluminum alloy and MMC is credited, while only 75% of the B-10 content in Boral® is credited.

A summary of the minimum B-10 loadings required in the poison plates as a function of the maximum lattice average enrichment level of the fuel assembly to be stored in a given 61BTH basket type is presented in Table T.2-3 for intact fuel. Table T.2-4 for damaged fuel, and in Table T.2-4a for failed and damaged fuel.

Table T.2-5 through Table T.2-10 and Table T.2-16 and Table T.2-17 define the minimum required cooling time after reactor discharge for a fuel assembly for a given assembly heat load, assembly average burnup, and maximum initial lattice average enrichment parameters. These tables ensure that the fuel assembly decay heat load is less than that specified for each table and that the corresponding radiation source term is bounded by that analyzed in Appendix T.5.

The NUHOMS®-61BTH DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight allowed is 705 lbs for fuel assemblies with channels and 640 lbs for fuel assemblies without channels.

The maximum fuel cladding temperature limit of 400 °C (752 °F) is applicable to normal conditions of storage and all short term operations from the spent fuel pool to the ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-61BTH DSC per NUREG-1536 [2.1]. In addition, NUREG-1536 [2.1] does not permit repeated thermal cycling of the fuel

cladding (limited to less than 10 cycles) with cladding temperature differences greater than 65°C (117°F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570°C (1058°F) is applicable to accidents or off-normal storage thermal transients [2.1].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses, including shielding, criticality, thermal and confinement. These evaluations are described in Appendices T.5, T.6, T.4 and T.7 respectively. The fuel assembly classes considered are listed in Table T.2-2. It was determined that the GE 7x7 is the enveloping fuel design for the shielding source term calculation because of its total assembly weight and highest initial heavy metal loading. For criticality safety, the GE 10x10 fuel assembly is the most reactive assembly type for a given enrichment. This assembly is used to determine the most reactive configuration in the DSC. Using this most reactive configuration, criticality analyses for all other fuel assembly classes, *except for GNF2 and ATRIUM 11 fuel assembly classes,* are performed to determine the maximum enrichment allowed as a function of the fixed poison loading. *The GNF2 and ATRIUM 11 fuel assembly classes are evaluated individually.* For thermal analysis, the FANP 9x9-2 fuel assembly is limiting, since it has the lowest effective thermal conductivity. The confinement analysis is based on GE 7x7 fuel assembly, since it results in the least free volume inside the DSC cavity.

For calculating the maximum internal pressure in the NUHOMS®-61BTH DSC, it is assumed that 1% of the fuel rods are damaged for normal conditions, up to 10% of the fuel rods are damaged for off normal conditions, and 100% of the fuel rods will be damaged following a design basis accident event. A minimum of 100% of the fill gas and 30% of the fission gases within the ruptured fuel rods are assumed to be available for release into the DSC cavity, consistent with NUREG-1536 [2.1].

The maximum internal pressures used in the structural analysis for the NUHOMS®-61BTH Type 1 DSC are 10, 20, and 65 psig for normal, off-normal and accident conditions, respectively, during storage and transfer operations. The maximum internal pressures for the 61BTH Type 2 DSC are 15, 20, and 120 psig for normal, off-normal and accident conditions, respectively during storage and transfer operations.

T.2.1.1 General Operating Functions

No change to Section 3.1.2.

T.2.5 Summary of NUHOMS®-61BTH System Design Criteria

T.2.5.1 61BTH DSC Design Criteria

The NUHOMS®-61BTH DSC is designed to store intact and/or damaged BWR fuel assemblies with assembly average burnup, lattice average initial enrichment and cooling time as described in Table T.2-1 and Table T.2-2. The maximum total heat generation rate of the stored fuel is limited to *1.2* kW per fuel assembly for the Type 2 DSC and 0.54 kW per fuel assembly for the Type 1 DSC. The maximum heat load per canister is limited to 31.2 kW for the Type 2 DSC and 22.0 kW for Type 1 DSC, in order to keep the maximum fuel cladding temperature below the limit [2.4] necessary to ensure cladding integrity. The fuel cladding integrity is assured by the NUHOMS®-61BTH DSC and basket design, which limits fuel cladding temperature and maintains a non-oxidizing environment in the DSC cavity as described in Chapters T.4 and T.7.

The NUHOMS®-61BTH DSC is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. A combination of fixed neutron absorbers and favorable geometry are employed to maintain the upper subcritical limit of 0.9415. The fixed neutron absorbers are in the form of plates made from either Borated Aluminum alloy or MMC or Boral®.

The NUHOMS®-61BTH DSC (shell and closure) is designed and fabricated as a Class 1 component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [2.2], and the alternative provisions to the ASME Code as described in Table T.3.1-2.

The basket is designed and fabricated in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG-3200 [2.2] and the alternative provisions to the ASME Code as described in Table T.3.1-2.

The principal design loadings for the NUHOMS®-61BTH DSC are provided in Table T.2-14. The applicable load combinations for the NUHOMS®-61BTH DSC are presented in Table T.2-11 and the corresponding stress criteria are presented in Table T.2-12 and Table T.2-13.

The NUHOMS®-61BTH system is designed to withstand the effects of severe environmental conditions and natural phenomena such as earthquakes, tornadoes, lightning and floods. Chapter T.11 describes the NUHOMS®-61BTH DSC behavior under these accident conditions.

The NUHOMS®-61BTH DSC design, fabrication and testing are covered by Transnuclear's Quality Assurance Program, which conforms to the criteria in Subpart G of 10CFR72.

T.2.5.2 HSM-H Models 80, 102, 152, 202 and HSM-H Design Criteria

There is no change to the HSM Models 80, 102, 152, 202 design criteria as presented in Chapter 3 of the UFSAR for Models 80/102 and the appropriate appendix for Models 152/202. The maximum heat load allowed for storage of a 61BTH in these HSMs remains at 22 kW.

There is no change to the HSM-H design criteria presented in Appendix P.2 except for accommodating a payload (61BTH DSC) with a maximum decay heat load of 31.2 kW.

Table T.2-1

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-1t.

Table T.2-2
BWR Fuel Assembly Design Characteristics[1] for the NUHOMS®-61BTH DSC

Transnuclear ID	**7x7 49/0**	**8x8 63/1**	**8x8 62/2**	**8x8 60/4**	**8x8 60/1**	**9x9 74/2**	**10x10 92/2**	**7x7 49/0**	**7x7 48/1Z**	**8x8 60/4Z**	**8x8 62/2**	**9x9 79/2**	**Siemens QFA**	**10x10 91/1**	***11x11***
Initial Design or Reload Fuel Designation	GE1 GE2 GE3	GE4	GE-5 GE-Pres GE-Barrier GE8 Type I	GE8 Type II	GE9 GE10	GE11 GE13	GE12 GE14 *GNF2*	ENC-IIIA	ENC-III[2]	ENC Va ENC Vb	FANP 8x8-2	FANP9 9x9-2	9x9	ATRIUM-10	*ATRIUM-11*
Maximum Length (in) (Unirradiated)	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.2	176.51	176.51	*176.51*
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	*UO_2*
Maximum Number of Fuel Rods	49	63	62	60	60	74	92	49	48	60	62	79	72	91	*112*

(1) Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.
(2) Includes ENC-IIIE and ENC-IIIF.

Table T.2-3
Maximum Fuel Assembly Lattice Average Initial Enrichment v/s Minimum B-10 Requirements for the NUHOMS®-61BTH DSC Poison Plates (Intact Fuel)

61BTH DSC Type	Basket Type	Maximum Lattice Average Enrichment (wt. % U-235)[(1)]	Minimum B-10 Areal Density, (grams/cm²)	
			Borated Aluminum/MMC	Boral®
1	A	3.7	0.021	0.025
	B	4.1	0.032	0.038
	C	4.4	0.040	0.048
	D	4.6	0.048	0.058
	E	4.8	0.055	0.066
	F	5.0	0.062	0.075
2	A	3.7	0.022	0.027
	B	4.1	0.032	0.038
	C	4.4	0.042	0.050
	D	4.6	0.048	0.058
	E	4.8	0.055	0.066
	F	5.0	0.062	0.075

Note:

1. *For ATRIUM 11 fuel assemblies, the U-235 wt. % enrichment is reduced by 0.55%.*

Table T.2-4
Maximum Fuel Assembly Lattice Average Initial Enrichment v/s Minimum B-10 Requirements for the NUHOMS®-61BTH DSC Poison Plates (Damaged Fuel)

61BTH DSC Type	Basket Type	Maximum Lattice Average Enrichment (wt. % U-235)		Minimum B-10 Areal Density, (grams/cm^2)	
		Up to 4 Damaged Assemblies(1)	Five or More Damaged Assemblies(1) (16 Maximum)	Borated Aluminum/MMC	Boral®
1	A	3.7	2.80	0.021	0.025
	B	4.1	3.10	0.032	0.038
	C	4.4	3.20	0.040	0.048
	D	4.6	3.40	0.048	0.058
	E	4.8	3.50	0.055	0.066
	F	5.0	3.60	0.062	0.075
2	A	3.7	2.80	0.022	0.027
	B	4.1	3.10	0.032	0.038
	C	4.4	3.20	0.042	0.050
	D	4.6	3.40	0.048	0.058
	E	4.8	3.50	0.055	0.066
	F	5.0(2)(3)	3.60	0.062	0.075

Notes
1. See Figure T.2-9 for the location of damaged fuel assemblies within the 61BTH DSC.
2. *ATRIUM 11 fuel assemblies are authorized for storage only in the Type 2F basket with a maximum of 4 damaged intact fuel assemblies.*
3. *For ATRIUM 11 fuel assemblies, the U-235 wt. % enrichment is reduced by 0.55%.*

Table T.2-4a
BWR Fuel Assembly Initial Lattice Average Enrichment v/s Minimum B-10 Requirements for the NUHOMS®-61BTH DSC Poison Plates (Failed and Damaged Fuel)(2)

61BTH DSC Type	Basket Type	Maximum Lattice Average Enrichment (wt% U-235)		Minimum B-10 Areal Density, gram/cm^2	
		Up to 4 Failed Assemblies (Corner Locations)(1)	Up to 4 Failed Assemblies (Corner Locations) and up to 12 Damaged Assemblies (Interior Locations)(1)	Borated Aluminum/MMC	Boral®
2	A	3.7	2.8	0.022	0.027
	B	4.0	3.1	0.032	0.038
	C	4.4	3.2	0.042	0.050
	D	4.6	3.4	0.048	0.058
	E	4.8	3.4	0.055	0.066
	F	5.0	3.5	0.062	0.075

Notes
1. See Figure T.2-9 for the locations of the failed and damaged assemblies within the 61BTH DSC.
2. *ATRIUM 11 fuel not authorized for storage.*

Table T.2-4b

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-1x.

Table T.2-5

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-4a.

Table T.2-6

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-4b.

Table T.2-7

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-4c.

Table T.2-8

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-4d.

Table T.2-9

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-4e.

Table T.2-10

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-4f.

The explanatory notes associated with Tables T.2-5 through T.2-10 and *Tables* T.2-16 *through T.2-18* can be found in CoC 1004 *Amendment 15* Technical Specifications following *Table 1-4i.* 72.48 AMD 15

Table T.2-16

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-4g.

Table T.2-17

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-4h.

Table T.2-18

The detailed information associated with this table can be found in CoC 1004 Amendment 15 Technical Specifications Table 1-4i.

The detailed information associated with this figure can be found in CoC 1004 *Amendment 15* Technical Specifications Figure 1-25.

Figure T.2-9

The detailed information associated with this figure can be found in CoC 1004 *Amendment 15* Technical Specifications Figure 1-25a.

Figure T.2-10

The detailed information associated with this figure can be found in CoC 1004 *Amendment 15* Technical Specifications Figure 1-25b.

Figure T.2-11

Table T.3.1-2

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, ASME Code Alternatives for the NUHOMS-61BTH DSC Confinement Boundary.

Table T.3.1-3

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, ASME Code Alternatives for the NUHOMS-61BTH DSC Basket.

The change on this page is a result of the issuance of Amendment 15.

Proprietary Information on Pages T.3.5-1 through T.3.5-15
Withheld Pursuant to 10 CFR 2.390.

Table T.3.5-5
Summary of Stress Results for 75g Side Drop

	7x7	8x8	8x8	8x8	8x8	9x9	10x10	7x7	8x8	8x8	8x8	8x8	9x9	10x10	8x8	8x8	*10x10*	*11x11*
GE Designation	GE1, GE2, GE3	GE4	GE5	GE8	GE9, GE10	GE11, GE13	GE12, GE14	ENC-IIIA	ENC-III	ENC-VA, ENC-VB	FANP 8x8-2	FANP 9x9-2	Siemens-QFA	Atrium-10 Atrium-10XM (5)	XXX-RCN	STD GE-4	*GNF2*	*ATRIUM 11*
Max Bending Stress, S_b (psi)	60,010	58,763	65,319	65,319	65,319	62,888	62,225	56,478	56,478	55,968	63,374	68,882	73,184	67,565	58,763	58,763	*65,154*	*68,768*
Internal Pressure (psi)	2365	2369	3178	3273	3230	2396	2310	2365	2411	3276	3180	2278	2448	2330	3326	3135	*1400(6)*	*1400(6)*
Axial $Stress_{press}$ (psi)(4)	10,661	8634	12,156	12,520	12,355	9691	9302	9586	9773	11,382	10,985	8163	10,524	10,142	12,122	11,426	*6,325*	*6,116*
Combined Stress(1) (psi)	70,671	67,397	77,475	77,839	77,674	72,579	71,527	66,064	66,251	67,350	74,359	77,045	83,708(4)	77,707	70,885	70,189	*71,479*	*74,884*
Yield Stress at Temperature(2) (psi)	93,834	93,834	93,834	93,834	93,834	93,834	93,834	93,834	93,834	93,834	93,834	93,834	93,834	93,834	93,834	93,834	*93,834*	*93,834*

Notes:

(1) Includes 0.0027 inch reduction in cladding thickness to account for oxidation.
(2) Temperature of 495.2°F used for stress allowable at the location where maximum stress occurs.
(3) Axial Stress due to pressure = p x D_{avg} / 4t
(4) Dynamic analysis performed in Appendix T.3.5.4 calculates maximum combined stress of 76,768 psi.
(5) Atrium-10XM has a larger diameter and larger cladding thickness than Atrium-10 and, therefore, is bounded by Atrium-10.
(6) The internal pressure of 70 bar (1,015 psi) is taken from Reference [3.53] and conservatively increased to 1,400 psi.

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Withheld Pursuant to 10 CFR 2.390

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Proprietary Information on Pages T.3.5-31 through T.3.5-35
Withheld Pursuant to 10 CFR 2.390.

Proprietary Information on Pages T.3.6-22 through T.3.6-33a
Withheld Pursuant to 10 CFR 2.390.

T.3.8 References

3.1 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Subsections NB, NC, NF, NG, and Appendices 1998 Edition through 2000 Addenda.

3.2 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section II, Part D, 1998 with 2000 Addenda.

3.3 NUREG/CR-0497, Rev. 2, "MATPRO-Version 11, A Handbook of Materials Properties for Use in the analysis of light Water Reactor Fuel Rod Behavior."

3.4 Brooks and Perkins, "Boral® Product Performance Report 624."

3.5 Pacific Northwest Laboratory Annual Report – FY 1979, "Spent Fuel and Fuel Pool Component Integrity," May 1980.

3.6 G. Wranglen, "An Introduction to Corrosion and Protection of Metals," Chapman and Hall, 1985, pp. 109-112.

3.7 A. J. McEvily, Jr., ed., "Atlas of Stress Corrosion and Corrosion Fatigue Curves," ASM Int'l, 1995, p. 185.

3.8 Baratta, et al. "Evaluation of Dimensional Stability and Corrosion Resistance of Borated Aluminum," Final Report submitted to Eagle-Pitcher Industries, Inc. by the Nuclear Engineering Department, Pennsylvania State University.

3.9 TNW letter 31-B9604-97-003 dated December 19, 1997 from Dave Dawson to Tim McGinty, NRC (Docket No. 72-1004).

3.10 "Hydrogen Generation Analysis Report for TN-68 Cask Materials," Test Report No. 61123-99N, Rev. 0, Oct 23, 1998, National Technical Systems.

3.11 ANSYS Engineering Analysis System, Users Manual for ANSYS Rev. 8.1, and Rev. 10.0A1, *and Revision 14.0.3,* Swanson Analysis Systems, Inc., Houston, PA.

3.12 Roark, 5th Edition, "Formulas for Stress and Strain."

3.13 U.S. Nuclear Regulatory Commission (U.S. NRC), "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)," Regulatory Guide 3.48 (Task FP-029-4), (October 1981).

3.14 American National Standard, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)," ANSI/ANS 57.9-1984, American Nuclear Society, La Grange Park, Illinois (1984).

3.15 U.S. Atomic Energy Commission, "Damping Values for Seismic Design of Nuclear Power Plants," Regulatory Guide 1.61, (October 1973).

3.44 Kuroda, M., et al., "Influence of Precipitated Hydride on the Fracture Behavior of Zircaloy Fuel Cladding Tube," Journal of Nuclear Science and Technology, Vol. 37, No. 8, August 2000.

3.45 "The Stress Analysis of Cracks Handbook" Third Edition by Hiroshi Tada et al., ASME press.

3.46 A. Machiels, EPRI Report, "Fracture Toughness Data for Zirconium Alloys Application to Spent Fuel Cladding in Dry Storage", 1001281, January 2001.

3.47 R. W. Hertzberg," Deformation and Fracture Mechanics of Engineering Material" John Wiley & Sons, New York, 1976.

3.48 Geelhood, K.J. Beyer, C.E., "Mechanical Properties for Irradiated Zircaloy, Transactions of ANS Conference (Vol. 93, PP, 707-708), November 2005.

3.49 NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," March, 2007.

3.50 Barsoum, R.S., "on the Use of Isoparametric Finite Elements in Linear Elastic Fracture Mechanics," International Journal for Numerical Methods in Engineering, Vol. 10, 1976.

3.51 Safety Analysis Report, "NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel," NRC Docket No. 72-1030.

3.52 LSDYNA Version 970 *and 971d R2, Revision 7600.1224* Keyword User's Manual, "NonLinear Dynamic Analysis."

3.53 Fuel Material Technology Report Volume II, A.N.T International, August 2007.

T.4.6.10 Thermal Evaluation of NUHOMS® 61BTH DSC with HLZCs #9 and #10

This section presents the thermal evaluation of 61BTH DSC with HLZC #9 and HLZC #10. The HLZCs are described as follows:

1. HLZC #9 allows a maximum heat load of 22.0 kW per DSC as shown in Figure T.2-10. This HLZC is applicable to both the 61BTH Type 1 and Type 2 DSCs.

2. HLZC #10 allows a maximum heat load of 31.2 kW per DSC and accommodates FAs with a maximum heat load of *1.2* kW as shown in Figure T.2-11. This HLZC is only applicable to 61BTH Type 2 DSC. *As shown in Figure T.2-11, Zone 3 in HLZC #10 can either be loaded with FAs with a maximum decay heat of 0.90 kW or 1.20 kW. Section T.4.6.10.2.1 presents the thermal evaluation for HLZC #10 wherein the FA in Zone 3 is loaded with a maximum decay heat of 0.9 kW and Section T.4.6.10.2.2 presents the thermal evaluation of HLZC #10 wherein the FA in Zone 3 is loaded with a maximum decay heat of 1.2 kW.*

The maximum heat load of 22 kW for HLZC #9 is equal to the maximum allowable heat load specified for HLZCs #1 and #2 as shown in Figures T.2-1 and T.2-2. Similarly, the maximum heat load of 31.2 kW for HLZC #10 is equal to the maximum allowable heat load of HLZCs #5, #6 and #7 as shown in Figures T.2-5 to T.2-7. The thermal evaluations presented in Section T.4.4 for storage conditions and Section T.4.5 for transfer operations are performed by considering the maximum heat load either as a heat flux on the radial inner surface of the DSC or as a volumetric heat generation rate applied over a homogenized basket. Because of this approach, the thermal evaluations presented in in Section T.4.4 for storage conditions and Section T.4.5 for transfer operations, are not dependent on the HLZC but are dependent only on the maximum heat load. Since the maximum heat loads considered for HLZCs #9 and #10 remain bounded by those previously evaluated, the DSC shell temperature profiles from these evaluations remain applicable for HLZCs #9 and #10.

Thermal evaluation of the 61BTH DSC with the HLZCs #9 and #10 are presented in Section T.4.6.10.1 and T.4.6.10.2, respectively. These evaluations are performed using the thermal model described in Section T.4.6. The only change considered to this thermal model is the updated HLZC.

T.4.6.10.1 Thermal Evaluation of 61BTH Type 1 and Type 2 DSCs with HLZC #9

The maximum heat load for HLZC #9 is 22 kW for both 61BTH Type 1 and Type 2 DSCs. This corresponds to the maximum allowable heat load for the 61BTH Type 1 DSC among all HLZCs and is lower than the maximum allowable heat load of 31.2 kW for 61BTH Type 2. Since the 61BTH Type 2 DSC is thermally more efficient than the 61BTH Type 1 DSC, the thermal evaluation for HLZC #9 is based on 61BTH Type 1 DSC.

Since no other changes are considered to the 61BTH DSC except for the HLZC, the thermal evaluation of 61BTH DSC Type 1 with HLZC #9 is based on a sensitivity study of the vertical transfer operation with the DSC/TC in a 120 °F ambient. The maximum heat load of 22 kW for HLZC 9 is identical to the maximum heat load considered for 61BTH Type 1 DSC in Section T.4.5. Therefore based on the discussion in the main body of Section T.4.6.10, the DSC shell temperature profiles from the thermal evaluation results presented for 61BTH DSC at 22 kW in Section T.4.5 are used as boundary conditions to evaluate the thermal performance of the 61BTH DSC with HLZC #9.

T.4.6.10.2 Thermal Evaluation of 61BTH Type 2 DSC with HLZC #10

As shown in Figure T.2-11, HLZC #10 has a maximum heat load of 31.2 kW. Because HLZC #10 has a higher maximum allowable heat load than that for the 61BTH Type 1 DSC of 22 kW, it is only applicable to the 61BTH Type 2 DSC.

Since no other changes are considered to the 61BTH DSC except for the HLZC, the thermal evaluation of 61BTH DSC Type 2 with HLZC #10 is based on a sensitivity study of the normal hot storage condition with 100 °F ambient and the vertical transfer condition with 120 °F ambient. The maximum heat load of 31.2 kW for HLZC 10 is identical to the maximum heat load considered for 61BTH Type 2 DSC in Section T.4.4 and Section T.4.5. Therefore, based on the discussion in the main body of Section T.4.6.10, the DSC shell temperature profiles from the thermal evaluation results, presented for 61BTH DSC at 31.2 kW in Section T.4.4 and Section T.4.5, are used as boundary conditions to evaluate the thermal performance of the 61BTH Type 2 DSC with HLZC #10.

T.4.6.10.2.1 HLZC #10 with a Maximum Decay Heat of 0.9 kW per FA

The following table compares the maximum fuel cladding and DSC component temperatures for the 61BTH Type 2 DSC with HLZC #10 to the design basis values presented in Table T.4-12 and Table T.4-14 for normal hot storage condition (DSC in HSM, 100 °F ambient). The design basis values presented for 61BTH Type 2 DSC are based on the bounding temperatures determined for HLZCs #5 through #8 with a maximum heat load of 31.2 kW for the normal hot storage condition (DSC in HSM, 100 °F ambient).

Maximum Component Temperatures for 61BTH Type 2 DSC for Normal Storage in 100 °F Ambient

HLZC	Fuel Cladding (°F)	Fuel Compartment (°F)	Neutron Absorber (°F)	R45 & R90 Rails (°F)	Top Grid (°F)	DSC Shell (°F)
61BTH Type 2 DSC (Design Basis) [Table T.4-12 and Table T.4-14]	719	690	689	514	496	434
61BTH Type 2 DSC HLZC #10	711	687	686	515	497	434
ΔT ($T_{HLZC\ \#10} - T_{Design\ Basis}$)	-8	-3	-3	1	1	0

As shown in the above table, the maximum temperatures of fuel cladding and fuel compartment for 61BTH Type 2 DSC with HLZC #10 are bounded by design basis values listed in Table T.4-12 and Table T.4-14. The 1 °F temperature increase observed for the basket rails and top grid is insignificant and does not affect the thermal or structural performance of the 61BTH Type 2 DSC. Based on this evaluation, the maximum fuel cladding temperature listed for the various storage conditions of a 61BTH Type 2 DSC in Table T.4-12, Table T.4-17 and Table T.4-21 remain bounding for HLZC #10.

A review of the time limits for transfer operations of a 61BTH Type 2 DSC with heat loads greater than 22 kW in Section T.4.5.4 shows that there are two separate time limits for 61BTH Type 2 DSC depending on the HLZC. HLZCs #5, #6 and #8 have transfer time limit of 26 hours, whereas HLZC #7 has a transfer time limit of 13 hours.

To determine the most appropriate time limit for transfer operation with HLZC #10, the 61BTH Type 2 DSC model with HLZC #7 (transfer time limit of 13 hours) was re-evaluated using HLZC #10. No other changes are considered in this evaluation. The following table presents a comparison of the maximum fuel cladding and DSC component temperatures determined previously for HLZC # 7 to those determined in this sensitivity study with HLZC # 10 for the vertical transfer condition (DSC in TC, 120 °F ambient) with a time limit of 15 hours (13 hours excluding the two hour allowance to initiate recovery actions).

Maximum Component Temperatures for 61BTH Type 2 DSC for Vertical Transfer, 120 °F Ambient, 15 hours

HLZC	Fuel Cladding (°F)	Fuel Compartment (°F)	Neutron Absorber (°F)	R45 & R90 Rails (°F)	Top Grid (°F)	DSC Shell (°F)
61BTH Type 2 DSC HLZC # 7	730	701	701	522	493	408
61BTH Type 2 DSC HLZC #10	722	698	698	524	494	408
ΔT ($T_{HLZC \#10} - T_{HLZC \#7}$)	-8	-3	-3	2	1	0

The results of this evaluation show that the maximum temperature of the fuel cladding and fuel compartment are lowered by 8 °F and 3 °F similar to the storage evaluation presented above. The temperature increase of *2* °F observed for the basket rails and *1* °F observed for the top grid is insignificant and does not affect the thermal or structural performance of the 61BTH Type 2 DSC. Furthermore, the maximum fuel cladding and DSC component temperatures for the 61BTH Type 2 DSC with HLZC # 10 are also below the bounding temperatures listed in Table T.4-12 and Table T.4-14 for the DSC in TC at 120 °F ambient (vertical transfer condition).

Based on this evaluation, the maximum fuel cladding temperatures listed for the various transfer conditions of a 61BTH Type 2 DSC in Table T.4-12, Table T.4-17, and Table T.4-21 remain bounding for HLZC #10. Therefore, the time limits for transfer operations determined for a 61BTH Type 2 DSC with HLZC 7 in Section T.4.5.4 are applicable to a 61BTH Type 2 DSC with HLZC #10.

In addition, the average helium temperature determined for HLZC #10 is 3 °F higher than that used in determining the maximum internal pressure in Table T.4-16. This small change in the average helium temperature does not affect the maximum internal pressure.

Based on this discussion, no further evaluations are required for a 61BTH Type 2 DSC with HLZC #10 and all design criteria described in Section T.4.1 are satisfied.

T.4.6.10.2.2 HLZC # 10 with a Maximum Decay Heat of 1.2 kW per FA

An alternate option is being proposed for HLZC #10, wherein the maximum decay heat per FA in Zone 3 can be increased to 1.2 kW while the maximum decay heat per FA in all other zones is limited to 0.393 kW as shown in Figure T.2-11. This section evaluates the thermal performance of 61BTH Type 2 DSC for this alternate option of HLZC #10 wherein Zone 3 is loaded with 1.2 kW per FA using the same methodology described in Section T.4.6.10.2.1.

Because there is no change to the total heat load of the DSC, the same limiting cases for transfer operation and storage conditions identified in Section T.4.6.10.2.1 are re-evaluated with the alternate configuration of HLZC #10.

Maximum fuel cladding and DSC components temperatures for the bounding storage and transfer conditions in comparison to the design basis values are listed below.

Maximum Component Temperatures for 61BTH Type 2 DSC for Normal Storage in 100 °F Ambient

HLZC	*Fuel Cladding (°F)*	*Fuel Compartment (°F)*	*Neutron Absorber (°F)*	*R45 & R90 Rails (°F)*	*Top Grid (°F)*	*DSC Shell (°F)*
Design Basis [Table T.4-12 and Table T.4-14]	*719*	*690*	*689*	*514*	*496*	*434*
Alternate HLZC #10	*712*	*689*	*688*	*517*	*498*	*434*
ΔT ($T_{HLZC \#10\_Alternate}$ - $T_{Design\ basis}$)	*-7*	*-1*	*-1*	*3*	*2*	*0*

Maximum Component Temperatures for 61BTH Type 2 DSC for Vertical Transfer, 120 °F Ambient, 15 hours

HLZC	*Fuel Cladding (°F)*	*Fuel Compartment (°F)*	*Neutron Absorber (°F)*	*R45 & R90 Rails (°F)*	*Top Grid (°F)*	*DSC Shell (°F)*
HLZC #7	*730*	*701*	*701*	*522*	*493*	*408*
Alternate HLZC #10	*724*	*700*	*700*	*525*	*495*	*408*
ΔT ($T_{HLZC \#10\_Alternate}$ - $T_{HLZC \#7}$)	*-6*	*-1*	*-1*	*3*	*2*	*0*

As shown in the above tables, the maximum temperatures of fuel cladding, fuel compartment and neutron absorber for 61BTH Type 2 DSC with the alternate HLZC #10 are bounded by the design basis values for HLZC #7 in Section T.4.6. The temperature increases of 3 °F observed for the basket rails and 2 °F observed for the top grid are insignificant and do not affect the thermal or structural performance of the 61BTH Type 2 DSC.

Based on this evaluation, the maximum fuel cladding temperatures for the 61BTH Type 2 DSC reported in Table T.4-12, Table T.4-17 and Table T.4-21 for normal, off-normal and accident conditions remain valid for the 61BTH Type 2 DSC with the alternate HLZC #10 wherein Zone 3 is loaded with 1.2 kW per FA. Therefore, the time limits for transfer operations determined for a 61BTH Type 2 DSC with HLZC #7 in Section T.4.5.4 are applicable to a 61BTH Type 2 DSC with HLZC #10.

In addition, the average helium temperature determined for the alternate HLZC #10 remains the same as that used in determining the maximum internal pressure in Table T.4-16 for the normal conditions. Therefore, there is no change in the maximum internal pressures.

Based on this discussion, no further evaluations are required for the 61BTH Type 2 DSC with the alternate HLZC #10 and all design criteria described in Section T.4.1 are satisfied.

$$dhl = \frac{Q/N}{n\left(\frac{\pi}{4}{d_p}^2\right)L_a} \qquad (6)$$

dhl = Heat generating boundary condition, Btu/min-in-°F
Q = Total decay heat load, Btu/min
N = Number of assemblies, 61
n = Number of fuel rods
d_p = Pellet outer diameter, in
L_a = Active fuel length, in

The models were run with a series of isothermal boundary conditions applied to the nodes representing the fuel compartment walls. The symmetry lines going through the center of the fuel assembly are kept at the adiabatic boundary conditions.

T.4.8.1.5 Results

The Siemens QFA 9x9 assembly has the minimum (bounding) axial conductivity. Backfill gas property does not have any effect on the axial effective fuel conductivity. Therefore, identical axial effective fuel conductivity values can be used for helium and vacuum conditions.

The calculated transverse conductivities for fuels to store in the 61BTH DSC are presented in Figure T.4-36 for a helium environment. As shown herein, the FANP9 9 x 9-2 assembly has the (bounding) minimum transverse conductivity. The bounding transverse effective conductivity values for fuels to store in the 61BTH DSC are listed in Section T.4.2.

T.4.8.1.6 Thermal Conductivities of 61BTH DSC with New Fuel Types GNF2 and ATRIUM-11

A review of the dimensions of GNF2 FA listed in Table T.2-2 shows that they are very similar to the previously evaluated GE12/GE14 FAs listed in the same table. Therefore, the thermal properties for the GE12/GE14 FAs are also applicable to the GNF2 FA and no further evaluation is necessary for this FA type.

For ATRIUM-11 FA type, the effective thermal properties are determined using the same methodology described in Sections T.4.8.1.3 and T.4.8.1.4. The following table compares the computed effective thermal conductivity values for ATRIUM-11 FA with the bounding values (for FANP 9x9-2 FA) calculated for all FAs except ATRIUM-11 FA.

Comparison of Effective Conductivities of ATRIUM-11 FA with Bounding BWR Fuel with Helium Backfill

Temperature (°F)	***Transverse Conductivity (Btu/min-in-°F)***		***Axial Conductivity (Btu/min-in-°F)***	
	Bounding BWR Fuel (Item 1 in Section T.4.2)	*ATRIUM-11*	*Bounding BWR Fuel (Item 1 in Section T.4.2)*	*ATRIUM-11*
200	*2.618E-04*	*2.997E-04*	*6.7000E-4*	*7.6000E-4*
300	*3.021E-04*	*3.397E-04*		
400	*3.520E-04*	*3.899E-04*		
500	*4.104E-04*	*4.483E-04*		
600	*4.756E-04*	*5.122E-04*		
700	*5.468E-04*	*5.813E-04*		
800	*6.250E-04*	*6.561E-04*		

As shown in the above table, both the transverse and axial effective thermal conductivities of ATRIUM-11 FA are higher than those determined for the bounding FA. The effective thermal properties for the bounding BWR fuel discussed in Section T.4.8.1.5 and listed in Section T.4.2 remain applicable for ATRIUM-11 FA.

T.4.8.2 Calculation of Fuel Effective Specific Heat and Density

This section presents the calculation of the fuel effective specific heat and density used in the transient thermal analyses.

Volume average density and weight average specific heat are calculated to determine the effective density and specific heat for the fuel assembly.

The equations to determine the fuel effective density ρ_{eff} and specific heat $C_{p\ eff}$ are shown below.

$$\rho_{eff} = \frac{\sum \rho_i V_i}{V_{assembly}} = \frac{\rho_{UO_2}\ V_{UO_2} + \rho_{Zr}\ V_{Zr}}{4a^2 L_a}$$

$$C_{p,eff} = \frac{\sum \rho_i\ V_i\ C_{Pi}}{\sum \rho_i\ V_i} = \frac{\rho_{UO_2}\ V_{UO_2}\ C_{P,UO_2} + \rho_{Zr}\ V_{Zr}\ C_{P,Zr}}{\rho_{UO_2}\ V_{UO_2} + \rho_{Zr}\ V_{Zr}}$$

where:

ρ_i, $C_{p,i}$, V_i = density, specific heat, and volume of component,
L_a = active fuel length, and
a = half of compartment width.

The properties of Zircaloy and UO_2 are provided in Section T.4.2.

The calculated minimum (bounding) values of fuel effective specific heat and fuel effective density for fuel to store in the 61BTH DSC are summarized in Section T.4.2.

Following the same methodology, the effective density and specific heat for ATRIUM-11 FA are calculated and compared to the bounding values (Item 1 in Section T.4.2) in the following table.

Comparison of Effective Density and Specific Heat of ATRIUM-11 FA with Bounding BWR Fuel with Helium Backfill

	Bounding BWR Fuel (Item 1 in Section T.4.2)	***ATRIUM-11***
Effective Density (lbm/in^3)	*0.103*	*0.116*
Effective Specific Heat (Btu/lbm-°F)	*0.0575*	*0.0575*

As seen from the above table, the effective density and specific heat of ATRIUM-11 FA are either higher than or equal to the values calculated for the bounding FA in Section T.4.2. The calculated bounding values of fuel effective specific heat and fuel effective density for fuel to store in the 61BTH DSC in Section T.4.2 remain applicable for ATRIUM-11 FA.

T.4.8.3 61BTH DSC Basket Effective Thermal Properties

The 61BTH DSC basket effective density, thermal conductivity and specific heat are calculated for use in the transient analyses of the 61BTH DSC in the OS197/OS197H/OS197FC-B transfer cask and in the HSM or HSM-H. The calculation of these thermal effective properties is based on the DSC component weights.

The 61BTH DSC effective density $\rho_{eff\,DSC\,basket}$, and specific heat $c_{p\,eff\,DSC\,basket}$ are calculated as volumetric and weight average values, respectively.

The effective transverse thermal conductivity is determined by theoretical solution for conduction in an infinite cylinder with uniform heat generation [4.28]:

$$k_{eff-basket} = \frac{Q}{4\pi \cdot L \cdot (T_c - T_s)}$$

where
Q is total heat load, W
L is cylinder (DSC cavity) length, m
T_C is temperature at the cylinder center, °C
T_S is temperature at the cylinder surface, °C

The effective transverse thermal conductivities of the 61BTH DSC basket $k_{eff\text{-}basket}$ are calculated for the Type 1 and Type 2 DSCs, using the corresponding ANSYS models.

The heat generation is applied to the fuel assemblies uniformly without a peaking factor. The temperatures from 100°F to 800°F are applied uniformly to the DSC shell.

An average, $(T_s+T_c)/2$, is used as the reference temperature, for which $k_{eff\text{-}basket}$ is reported.

The bounding radial and axial thermal conductivity values for 61BTH DSCs are shown in Section T.4.2.

T.4.8.4 Effective Air Conductivity in the HSM-H Closed Cavity

During blockage of the inlet and outlet vents, the air within the HSM-H is trapped. The convection heat transfer under these circumstances reduces to free convection in closed cavities. For conservatism, no convection is considered within the HSM-H cavity during blockage of the vents.

T.5 Shielding Evaluation

This chapter specifically addresses the shielding evaluation of the NUHOMS® 61BTH *System* with design basis BWR fuel loaded in a NUHOMS®-61BTH DSC. The radiation shielding evaluation for the Standardized NUHOMS® System (during loading, transfer, and storage) for the other NUHOMS® canisters is discussed in other sections and appendices of the UFSAR. The NUHOMS®-61BTH *System* consists of the NUHOMS® horizontal storage module (HSM) and HSM-H, the OS197 transfer cask (TC), and the 61BTH Type 1 and the Type 2 DSCs.

The 61BTH DSC will be transferred using either the OS197/OS197H or a modified version of the OS197FC TC (OS197FC-B) if air circulation is required. The NUHOMS® 61BTH System will use either the HSM Model 80, Model 102, or the HSM-H module (up to 31.2 kW/DSC) for storage.

The radiation shielding evaluation described below is for the NUHOMS® 61BTH Type 1 and Type 2 DSCs loaded in a NUHOMS® System TC. For heat levels below 22 kW both DSC types can be transferred in any of the 3 transfer casks: 1) OS197, 2) OS197H, and 3) OS197FC-B, and stored in any of these HSMs: 1) HSM Model 80, 2) HSM Model 102, 3) HSM Model 152, 4) HSM Model 202, and 5) HSM-H. For heat loads exceeding 22 kW, the 61BTH Type 2 DSC must be used and can only be stored in HSM-H and transferred in the OS197FC-B TC. HSM Model 80 offers the least amount of shielding and therefore, bounds all other HSMs. With respect to shielding performance, seven possible loading combinations are considered as listed below:

(1) 61BTH Type 1/2 DSC → OS197 (bounded by #3)
(2) 61BTH Type 1/2 DSC → OS197H (bounded by #3)
(3) 61BTH Type 1 DSC → OS197FC-B (bounds OS197/OS197H and Type 2 DSC)
(4) 61BTH Type 2 DSC → OS197FC-B (bounded by #3)
(5) 61BTH Type 1/2 DSC → HSM Model 80 (bounding)
(6) 61BTH Type 1/2 DSC → HSM Model 102 (bounded by #5)
(7) 61BTH Type 1/2 DSC → HSM Model HSM-H (bounded by #5)

These design features of the HSM result in the occupational and site dose rates being as low as reasonably achievable (ALARA).

The NUHOMS® 61BTH DSC can also be stored within an upgraded HSM model, designated as HSM-HS as described in Appendix U of the UFSAR. From a shielding standpoint, the HSM-HS-module is identical to the HSM-H module. Therefore, all calculations performed with the HSM-H are applicable to the HSM-HS.

The NUHOMS® 61BTH DSC is also transferred in a modified version of the OS200 TC as described in Appendix U of the UFSAR. The OS200 TC is fitted with an aluminum sleeve to accommodate the smaller diameter 61BTH DSC. The basket layout for Type 1 and Type 2 DSC configurations is identical except for the basket transition rails. Each DSC configuration is designed to store up to 61 intact BWR fuel assemblies or up to 61 damaged fuel assemblies in accordance with Figure T.2-9. For shielding

purposes radiological sources related to the 61BTH Type 2 bound the 61BTH Type 1 because assemblies with higher neutron and gamma sources are loaded in the outer zones. The presence of solid aluminum rails that fill the space between the peripheral fuel compartments and the DSC shell results in a more effectively shielded configuration for the Type 2 DSC. When such bounding neutron and gamma sources are placed in a Type 1 DSC the resulting shielding configuration, HSM and TC dose rates are bounding for all the shielding configurations. Therefore, the shielding evaluation presented herein is performed for the hypothetical shielding configuration where radiological source terms bounding for Type 2 DSC are analyzed with the Type 1 DSC.

The NUHOMS® 61BTH Type 1 DSC is identical to the NUHOMS® 61BT DSC analyzed in UFSAR Appendix K, except for an optional redesigned basket hold-down ring. Relative to the existing 61BT DSC, the 61BTH Type 1 DSC allows for an increase in heat load from 18.3 kW to 22 kW, increase in maximum burnup from 40,000 MWd/MTU to 62,000 MWd/MTU, and an increase in maximum initial fuel enrichment from 4.4 wt. % U-235 to 5 wt. % U-235.

The 61BTH Type 2 DSC is also based on the basket design for the 61BT DSC with modifications to the shell assembly (cover plate thicknesses are increased to handle higher internal pressures) and to the basket transition rails to allow storage of fuel assemblies with a total heat load of up to 31.2 kW, with burnup of up to 62,000 MWd/MTU and with maximum initial fuel enrichments of up to 5 wt. % U-235. The Type 2 basket also incorporates the redesigned hold down ring.

The OS197FC-B is essentially the same as the OS197FC except that the lid and bottom have been modified to introduce air cooling design features to accommodate a higher decay heat load (>22 kW). The design of the OS197FC TC is identical to the design of OS197/OS197H TC except that the OS197FC TC has a modified top lid. For the shielding analysis, OS197FC-B TC is used to bound the OS197/OS197H TC, also because the design features in the TC radial direction are identical for all three TCs; and OS197FC top axial geometry bounds other TCs.

There are a total of ten possible heat load zoning configurations (HLZCs) for the 61BTH Type 1 and Type 2 DSCs. Five out of ten total DSC HLZCs are for Type 2 DSC only. The remaining five can be used with either DSC type; however, certain restrictions apply for Type 1 DSC in some cases. DSC HLZCs are depicted in Figures T.2-1 through Figure T.2-8, Figure T.2-10, and Figure T.2-11 of Chapter T.2.

The fuel qualification tables shown in Table T.2-5 through Table T.2-10, Table T.2-16 *through* Table *T.2-18* are developed for the design basis heavy metal loading of 0.198 MTU with SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] and for the heavy metal loading of 0.170 MTU with SAS2H/ORIGEN-S modules of SCALE 5.0 [5.20]. Section *7.2.3.2* of Chapter *7* provides the methods for determining minimum required cooling times using fitting equations or linear interpolation for a given MTU between 0.170 MTU and 0.198 MTU. *Also, Section T.5.2 provides the methods for determining the minimum required cooling times for combinations of burn-up and enrichments in "Not Analyzed" domain of the Fuel Qualification Tables (FQTs). Cells relevant to this domain are empty and shaded in the FQTs.*

Radiological source terms for the shielding analysis are calculated using design basis heavy metal loading of 0.198 MTU.

The NUHOMS®-61BTH DSCs are designed to store BWR fuel assemblies with the characteristics described in Table T.2-1 and Table T.2-2, and associated tables and figures of Chapter T.2. The NUHOMS® 61BTH Type 2 DSC is also designed to store up to four failed fuel assemblies in the corner locations of the basket. Each failed fuel assembly is housed inside a failed fuel canister prior to loading in these designated positions within the basket.

Radiological sources related to Type 2 DSC HLZCs #5, #6, and #7 result in nearly the same maximum HSM dose rates and maximum dose rates at the "middle of transfer cask side." Bounding dose rates for the HSM and "middle of transfer cask side" are due to HLZC #6 radiological source terms. However, HLZC #7 source terms are important when calculating dose rates on and around the ends of the TC. Dose rates are also estimated for the HSM Model 80 and 102. The hypothetical DSC shielding configuration where bounding radiological HLZC #6 or #7 source terms are loaded into 61BTH Type 1 DSC is referred to as the "design basis DSC shielding configuration" or simply "design basis DSC" in the discussion that follows.

The design basis BWR fuel source terms are derived from a bounding "generic" fuel assembly at 0.198 MTU. The parameters of the bounding "generic" fuel assembly are selected in such a way that the resulting radiological and decay heat source terms bound those from all other fuel assembly types that are authorized for loading into the NUHOMS® 61BTH DSC. This "generic" fuel assembly shares many common features with the GE-2,3 7x7 Type G2A assembly. It is bounding because it has the highest initial heavy metal loading as compared to the 8x8, 9x9, 10x10, *and 11x11* fuel assemblies which are also authorized for loading into the NUHOMS®-61BT DSC. Its parameters are described in Section T.5.2. In addition, the maximum Co-59 content of each hardware region for the bounding fuel assembly type is used to determine the activation source for each fuel assembly region.

Maximum decay heat allowed for the 61BTH Type 1 DSC is 22 kW and a maximum heat load of 31.2 kW is allowed for the Type 2 DSC. The HLZCs to be used for the 61BTH Type 1 DSC are shown in Figure T.2-1 through Figure T.2-4 and Figure T.2-10 of Chapter T.2. Fuel assemblies loaded in 61BTH Type 2 DSC can have a maximum decay heat of *1.20* kW per assembly. The design basis fuel source terms for this evaluation are defined as the source terms from fuel with the burnup/initial enrichment/cooling time combination that results in a maximum calculated dose rate on the surface of the HSM and/or TC side because the highest source fuel assemblies are on the outer periphery of the basket region where self-shielding due to adjacent assemblies is limited.

The dose rates at the side of the TC are bounded by 61BTH DSC HLZC #6 and in specific areas the dose rates at the top/bottom of the TC are bounded by 61BTH Type 2 DSC HLZC #7. The bounding burnup, minimum initial enrichment, and cooling time combinations for the fuel assemblies used in the shielding analysis of the 61BTH design basis DSC in the OS197FC-B are as follows:

- Spent fuel parameters for HLZC #6, in OS197FC-B:
    - Central zone (0.22 kW per assembly): 31 GWd/MTU, 0.9 wt. % U-235, 11.7-year cooled fuel

T.5.2 Source Specification

Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the design basis heavy metal weight of 0.198 MTU. The SAS2H/ORIGEN-S results are used to develop the fuel qualification tables listed in Table T.2-5 through Table T.2-10, Table T.2-16 *through* Table *T.2-18* of Chapter T.2 and the design basis fuel source terms suitable for use in the shielding calculations.

The GE-2,3 7x7 Type G2A assembly is the bounding fuel assembly design for shielding purposes because it has the highest initial heavy metal loading in the fuel and Co-59 content in the hardware regions as compared to the 8x8, other 9x9, 10x10, *and 11x11* fuel assemblies which are also authorized contents of the NUHOMS®-61BTH DSC. The neutron flux during reactor operation is peaked in the active fuel region of the fuel assembly and drops off rapidly outside the active fuel region. Much of the fuel assembly hardware is outside of the active fuel or in-core region of the fuel assembly. To account for this reduction in neutron flux, the fuel assembly is divided into four exposure "regions." The four axial regions used in the source term calculation are: the bottom (nozzle) region, the active fuel region, the (gas) plenum region, and the top (nozzle) region. The GE 7x7 fuel assembly masses for each irradiation region are listed in Table T.5-6. The light elements that make up the various materials for the various fuel assembly materials are taken from Reference [5.6] and are listed in Table T.5-7. The design basis heavy metal loading is 0.198 MTU. These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. To account for the reduction in neutron flux outside the active fuel regions, neutron flux (fluence) correction factors are applied to the light element composition for each region. The neutron flux correction factors which are from Reference [5.19] are given in Table T.5-8.

Evaluations of the existing light water reactor (LWR) fuel data with SAS2H and the 44-group ENDF/B-V library used in the calculation of the design basis source terms are documented in References [5.17] and [5.18]. These comparisons all show generally good agreement between the calculations and measurements, and show no trend as a function of burnup in the data that would suggest that the isotopic predictions, and therefore neutron and gamma source terms, would not be in good agreement. A similar conclusion is also reached by the results documented in JAERI report [5.14]. In fact, for the case with 46,460 MWd/MTU burnup, the isotopic predictions are all within 2% of those measured. There are ongoing efforts, some of which are documented in Reference [5.12], to obtain more data for burnups above 45 GWd/MTU.

There are cross-section data on about 1600 isotopes in the cross section libraries available for SAS2H. Only about 20 isotopes are primary concern when dealing with high burnup spent fuel [5.13, 5.15]. According to Reference [5.15] 95 % of the decay heat is dominated by fewer than 10 nuclides for LWR assemblies at five years of cooling. Eighty-five percent of the decay heat would be contributed by only four isotopes after 100 years.

Applicability of SAS2H for prediction of isotopic content in BWR assemblies was analyzed in [5.13]. A UO_2 sample was burned to 57 GWd/MTU in a BWR reactor. The sample U-235 enrichment was 4.97 wt. %. Also, the isotopic content of the discharged sample was measured experimentally. Measured content was reported for actinides and fission products. Among concentrations of 16 nuclides investigated, five agreed with the measured values to within ±5%.

The fuel qualification tables are generated based on the decay heat limits for the various heat load zoning configurations shown in Figure T.2-1 through Figure T.2-8 of Chapter T.2. SAS2H is used to calculate the minimum required cooling time to the nearest 0.1 year (0.3 years at burnups greater than ~50 GWd/MTU when considering low, less than 0.35 kW/FA, decay heat powers) as a function of fuel assembly initial enrichment and burnup for each decay heat limit. These cooling times are rounded up to the nearest 0.5 year increment in the final fuel qualification tables. Because the decay heat generally increases slightly with decreasing enrichment for a given burnup, it is conservative to assume that the required cooling time for a higher enrichment assembly is the same as that for a lower enrichment assembly with the same burnup. The required cooling time for initial enrichments that fall between any two SAS2H runs are assumed to be that of the lower enrichment case results.

Parameters that influence the source term calculations are fuel assembly specific power (expressed in MW/Fuel Assembly (MW/FA)) and the total time between cycles. Other depletion parameters like cycle length and number of cycles are derived from the target burnup, MTU loading and specific power. The most important parameter for the calculation of source terms is the specific power. Specific power for typical US-BWR fuel assemblies are ≤ 5 MW/FA. The source terms for this evaluation are calculated using a specific power that ranges from 7 MW/FA (at lower burnups) to 14 MW/FA (at higher burnups) and results in a conservative estimation of the source terms. The time between cycles utilized is 73 days and represents a typical downtime for US BWRs (60 to 90 days).

FQTs are developed for two different uranium loadings: 0.170 MTU and 0.198 MTU. Because cooling times are selected to target specific decay heat values and decay heat is proportional to the uranium loading, the FQT cooling times decrease with decreasing uranium loading to maintain the same heat load. In most cases, the uranium loading of a fuel assembly will fall between the 0.170 MTU and 0.198 MTU values. In such cases, the cooling time interpolation methodology described in Section M.5.6 may be employed.

Each FQT contains an unanalyzed zone marked in the FQTs as gray. Limited extrapolation of FQT cooling times into the unanalyzed regions is allowed. The extrapolation may be performed for a maximum difference of 4 GWd/MTU in burnup or 0.4 wt.% in enrichment. The extrapolation may be performed for either fixed enrichment (variable burnup, fixed FQT column) or fixed burnup (variable enrichment, fixed FQT row). The methodology is:

1. *Perform a regression analysis on the FQT cooling times and associated variable (either burnup or enrichment). Note: All FQT cooling times in either the row or column of data being extrapolated shall be used, even if many of the cooling times are the same.*

2. *Develop a fitting equation for the data. A fourth-order polynomial with parameters having at least six significant digits to avoid rounding errors is recommended.*

3. *Use the fitting equation to compute the extrapolated cooling time at the desired enrichment or burnup.*

4. *Add 0.2 years as additional margin.*

Because extrapolation may be performed on either an FQT row or column of data, in some cases extrapolation to the same FQT cell could be achieved using either data set. It is possible that the extrapolating equations with two alternative sets of parameters may result in slightly different predictions for the cooling times. However, either of the predicted values are legitimate to use because they are both conservative.

An example is provided for extrapolating the 170 kgU FQT for a 1.2 kW fuel assembly at a fixed burnup of 50 GWd/MTU. Note that only built-in capabilities of MS Excel® are utilized in this example. The cooling times are extracted from Table T.2-18 and summarized in the table below. The minimum enrichment in the analyzed region is 2.6 wt.%, and the cooling time for an enrichment of 2.2 wt.% is desired.

Enrichment (wt.%)	*Cooling time (years)*	*Enrichment (wt.%)*	*Cooling time (years)*
2.60	2.60	4.00	2.40
2.70	2.60	4.10	2.40
2.80	2.60	4.20	2.40
2.90	2.60	4.30	2.40
3.00	2.60	4.40	2.40
3.10	2.50	4.50	2.40
3.20	2.50	4.60	2.40
3.30	2.50	4.70	2.30
3.40	2.50	4.80	2.30
3.50	2.50	4.90	2.30
3.60	2.50	5.00	2.30
3.70	2.50		
3.80	2.50		
3.90	2.40		

The fitting equation Cooling Time as function of Enrichment using 4th order polynomial is:

CT = -0.016105 enr4 + 0.231001* enr3 - 1.219261* enr2 + 2.676368* enr + 0.572562*

For enr = 2.2 wt.%, cooling time = 2.6 years. An additional 0.2 years is added to this value for a final extrapolated cooling time of 2.8 years.

Additional examples are shown in Table T.5-30.

The design basis source terms are defined as the burnup/initial enrichment/cooling time combination given in the fuel qualification tables that result in the maximum dose rate on the surface of the HSM or TC. Note that for a given DSC design, the design basis HSM source will not necessarily be the same as the corresponding design basis TC source. The 1-D discrete ordinates code ANISN [5.7] and the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.5] is used to determine the relative HSM and TC dose rate for each entry in the fuel qualification tables and thereby determine the design basis source. As ANISN is a 1-D code, a single dose location must be selected for both the HSM and TC for analysis purposes. For the HSM, the roof can be selected as the dose location, and for the TC the cask side is selected as the dose location. This approach, described in detail in Section T.5.2.4, is consistent with the method used to determine the fuel qualification tables for the Standardized NUHOMS® 24PTH described in Appendix P, Chapter P.5. The radiological source terms generated in the SAS2H/ORIGEN-S runs are used in the ANISN evaluations to calculate the surface dose rates.

The ANISN models are similar to the appropriate MCNP models for the locations of interest. Note that ANISN code is not used to calculate any design basis dose rates. MCNP code models are used for calculating design basis dose rates.

A sample SAS2H/ORIGEN-S input file for the Active Fuel Region of 0.70 kW/FA assembly is listed in Section T.5.5.1. This case corresponds to 62 GWd/MTU, 2.6 wt. % U-235 and 7.1 -years cooling case.

T.5.4.6 Assumptions

The following general assumptions are used in the analyses. *Shielding analysis models using radiological sources described earlier and based on the assumptions described next result in bounding dose rates for the arrangements of the fuel assemblies in the DSCs as prescribed by HLZCs depicted on Figures T.2-1 through Figure T.2-8, Figure T.2-10, and Figure T.2-11 of Chapter T.2.*

T.5.4.6.1 Source Term Assumptions

- The primary neutron source in LWR spent fuel is the spontaneous fission of ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 90% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein and is assumed to follow the ^{244}Cm fission spectrum provided in Section T.5.2.2.
- The BWR heavy metal weight is assumed to be 0.198 MTU per fuel assembly. *Radiological sources from such an assembly result in bounding dose rates for the NUHOMS® 61BTH system loaded with its authorized content if used in the shielding analysis models described next.*

T.5.4.6.2 HSM-H Dose Rate Analysis Assumptions

- The 61BTH DSC and fuel assemblies are positioned as close to the HSM-H front door as possible to maximize the HSM-H front wall dose rates.
- Planes of reflection are used to simulate adjacent HSM-Hs.
- Embedment and rebar in the HSM-H concrete are conservatively neglected.
- Penetrations on the exterior of the HSM-H modules for instrumentation and ease of installation are not modeled since they do not result in any significant change in the dose rate distribution and are covered by other modeling conservatisms.
- The borated neutron absorber sheets in the 61BTH DSC are modeled as aluminum.
- An axial source distribution is discussed in Section T.5.2.3 is utilized.
- Fuel is homogenized within the fuel compartment, although the 61BTH DSC basket is modeled explicitly.

T.5.4.6.3 HSM Model 102 Dose Rate Analysis Assumptions

The dose rates for HSM Model 102 were also calculated using MCNP. Those dose rates are due to bounding 61BTH DSC Type 1 loading configuration sources. This configuration includes 0.393 kWt/FA assemblies in the central 25 fuel compartments. The next layer of 24 fuel compartments holds 0.54 kWt/FA assemblies, the outer compartments admit assemblies generating 0.54 kWt/FA. Note that this is also a fictitious loading configuration because more than 22.0 kWt/DSC heat load is not allowed for 61BTH DSC Type 1 and HSM Model 102 configuration. This "fictitious" configuration is depicted in Figure T.5-1. The same set of assumptions listed in Section T.5.4.6.2 applies to MCNP model of HSM Model 102.

Cask Decontamination. The 61BTH DSC and the OS197FC-B TC are assumed to be completely filled with water, including the region between 61BTH DSC and cask, which is referred to as the "TC/61BTH DSC annulus." The 61BTH DSC top shield plug and inner cover plate are assumed to be in place and the temporary shielding has not yet been installed (configuration prior to placement of the automated welding system (AWS)). Results for this case are provided in Table T.5-5.

Welding and 61BTH DSC Draining. Before the start of welding operation, approximately 60% of the water in the DSC cavity is removed due to hydrogen generation considerations. A dry DSC cavity is assumed in all welding models to be conservative. Temporary shielding consisting of three inches of NS-3 and one inch of steel is assumed to cover the 61BTH DSC top shield plug. In addition, the DSC outer top cover plate is not present. The cask/61BTH DSC annulus is assumed to remain completely filled with water. Results for this case are provided in Table T.5-5.

T.5.4.10 Impact on Dose Rates due to Reduced Density Concrete and Gaps between HSMs

A bounding analysis is performed by employing a minimum concrete density of 140 pounds per cubic foot (pcf) in the HSM-H MCNP model combined with a maximum gap of 1.5 inches between adjacent HSM-H modules and shield walls to determine the effect on maximum and average dose rates due to a fully loaded 32PTH1 DSC. These calculations are documented in Appendix U.5, Section U.5.4.10. The ratios shown in Appendix U.5, Table U.5-*18* and Table U.5-*19* can be used as scaling factors to increase the maximum and surface-average dose rates of the 61BTH in the HSM-H to account for low density concrete and 1.5" gaps. Note that the HSM-H concrete contains high density rebar which is not credited in the MCNP models. Further, the modules are installed adjacent to each other such that there will not be a "uniform" gap of 1.5 inches. Ignoring the effect due to increased vent dose rates, the increase in the average dose rates caused by both the maximum postulated uniform gaps and the minimum postulated concrete density is expected to be less than 20% at the front and roof surfaces of the HSM-H module. Dose reduction hardware may be installed to further reduce these dose rates. 72.48

Table T.5-30
Minimum Cooling Time by 4th Order Polynomials Vs Minimum Cooling Time determined by SAS2H – 1.2 kW

Burnup / Enrichment (MWd/MTU / wt%)	*Cooling time (Yrs)*	
45 GWd/MTU / 2.2 wt%	*2.4*	*4th order polynomials:* $(-0.049917*enr^4 + 0.757046*enr^3 - 4.214617*enr^2 + 10.085886*enr - 6.454616)+0.2$
	2.4	*SAS2H*
50 GWd/MTU – 2.2 wt%	*2.8*	*4th order polynomials:* $(-0.016105*enr^4 + 0.231001*enr^3 - 1.219261*enr^2 + 2.676368*enr + 0.572562)+0.2$
	2.7	*SAS2H*
55 GWd/MTU - 2.2 wt%	*3.1*	*4th order polynomials:* $(-0.012465*enr^4 + 0.201286*enr^3 - 1.203539*enr^2 + 3.009657*enr + 0.252701)+0.2$
	3.1	*SAS2H*
62 GWd/MTU - 2.2 wt%	*3.6*	*4th order polynomials:* $(-0.026123*enr^4 + 0.397919*enr^3 - 2.232819*enr^2 + 5.291467*enr - 1.057387)+0.2$
	3.6	*SAS2H*
70 GWd/MTU – 2.2 wt%	*4.2*	*4th order polynomials:* $(-0.034190*enr^4 + 0.525662*enr^3 - 2.987048*enr^2 + 7.214849*enr - 2.236293)+0.2$
	4.2	*SAS2H*

T.6.1 Discussion and Results

Figure T.6-1 and Figure T.6-2 show the radial cross section of the NUHOMS®-61BTH Type 1 and Type 2 DSCs. The generic cask consists of an inner stainless steel shell, and lead gamma shield, a stainless steel structural shell and a hydrogenous neutron shield. This analysis is applicable to any licensed cask of similar construction. The NUHOMS®-61BTH DSC/Cask configuration is shown to be subcritical under normal, off-normal and accident conditions.

For all fuel assemblies listed in Table T.6-2 except for the GNF2 and ATRIUM 11 fuel assemblies, the criticality calculations utilize the General Electric (GE) 10x10 fuel assembly as the most reactive fuel. For GNF2 and ATRIUM 11 fuel assemblies, separate criticality evaluations are performed to determine the maximum allowable U-235 wt. % enrichment. The calculations determine k_{eff} with the CSAS25 and CSAS5 control modules of SCALE-4.4 and SCALE 6.0, respectively, [6.1 and 6.7] for various configurations and initial enrichments, including all uncertainties to assure criticality safety under all credible conditions.

The results of the evaluation demonstrate that the maximum k_{eff}–including statistical uncertainty– is less than the Upper Subcritical Limit (USL) determined from a statistical analysis of benchmark criticality experiments. The statistical analysis procedure includes a confidence band with an administrative safety margin of 0.05.

T.6.2 Package Fuel Loading

The NUHOMS®-61BTH DSC is capable of transporting BWR fuel assemblies with or without fuel channels and as intact or damaged fuel assemblies. The NUHOMS®-61BTHF DSC, an alternate version of the NUHOMS®-61BTH Type 2 DSC, is also designed to accommodate up to a maximum of 4 failed fuel assemblies encapsulated in individual failed fuel cans and placed in cells located at the outer edge of the DSC as shown in Figure T.2-9.

The fuel assemblies considered as authorized contents are listed in Table T.6-2.

Table T.6-3 lists the fuel parameters for the BWR fuel assemblies. Reload fuel from other manufacturers, for the same fuel assembly class, with the same parameters are also allowed. The design basis fuel chosen for the NUHOMS®-61BTH system is the GE 10x10 fuel assembly, *except for the GNF2 and ATRIUM 11 fuel assemblies that are evaluated individually*. The GE 10x10 assembly is used because, as demonstrated in Section T.6.4, it is the most reactive assembly of those authorized to be shipped in the NUHOMS®-61BTH DSC System, *except for the GNF and ATRIUM 11 fuel assemblies.*

T.6.4 Criticality Calculation

This section describes the models used for the criticality analysis. The analyses were performed with the CSAS25 module of the SCALE system. A series of calculations were performed to determine the most reactive fuel and configuration. The most reactive fuel, as demonstrated by the analyses, is the GE 10x10 assembly for both the intact, damaged, and failed lattices. *The GNF2 and ATRIUM 11 fuel assemblies are evaluated individually in the criticality analyses to determine the maximum allowable U-235 wt. % enrichment.* The most reactive credible configuration is an infinite array of flooded casks with minimum assembly-to-assembly pitch and the poison plate gaps located near the center of the basket and at the centerline of the active fuel region.

The NUHOMS®-61BTH DSC is analyzed for additional considerations arising from mechanical uncertainties of damaged fuel assemblies after a hypothetical accident. In case of a severe transportation accident, rod breakage may be postulated to occur in rods with known pre-existing gross cladding failure. These models were constructed to evaluate the effects of radial movement of fuel rod pieces (the result of "single-ended" breaks), and axial movement (the result of "double-ended" breaks). Loose fuel pellets or shards may become dislodged if a rod becomes severed, but this will not result in a more reactive state than the cases described below because the fuel assembly is under-moderated by design. The models used to study these limiting breaks are described below.

Single breaks- "Free ends" caused by a break are assumed to move away from the rest of the fuel assembly. Increasing the rod spacing of the broken rods is found to increase k_{eff}. Conversely, k_{eff} is expected to decrease for local decreases in rod pitch. Rods on the exterior of the fuel assembly are displaced in the models and the fuel assembly is assumed to be pressed in the corner of the fuel cell, thus maximizing the potential rod displacement. Since internal rods cannot move as far as rods on the outside of the assembly, they are not limiting. For modeling simplicity, an entire face of 7 rods for the 7x7 array and 8 rods for the 8x8 array are assumed to evenly move away from the remainder of an assembly, as shown in Figure K.6-6 of Appendix K. This overpredicts the effect of single rod breaks since the grid spacers of the fuel will limit radial rod displacement over most of the length of the rod.

Double breaks- The affect of pieces of fuel rod migrating axially was investigated by conservatively adding an entire row of fuel rods in the models. Again, the fuel assembly was assumed to be in the worst case position: pressed in the corner of the fuel compartment as shown in Figure K.6-7 of Appendix K. In addition, total cladding loss was assumed for the damaged rows of rods to simulate the bare fuel rod case. The limiting case was the double-ended break with the damaged rods being modeled without the cladding. This is expected to be the limiting case because the extra row of rods added to the model represents an increase in the fuel loading of the canister.

Rod Pitch Variation- The effect of bending and bowing of rods together with the total loss of grid spacers was investigated by varying the fuel rod pitch for all the fuel assembly classes from a minimum (where the rods are close to each other) to a maximum (bounded by the internal dimension of the rod compartment). This was done to determine the optimum rod pitch where the reactivity of the fuel lattice is maximized. In addition, rods were removed (non

mechanistically) from within the lattice to determine the optimum rod positions (and the number of rods) to bound the expected lattice configurations. This hypothetical accident case is modeled

to maximize the reactivity of the damaged fuel assembly and also to qualify fuel assemblies with damaged grids and missing rods to be loaded in the damaged fuel assembly locations.

The most reactive damaged fuel assembly configuration is based on a 10x10 lattice with optimum pitch and 95 fueled rods.

T.6.4.1 Calculational Method

T.6.4.1.1 Computer Codes

The CSAS25 control module of SCALE-4.4 [6.1] was used to calculate the effective multiplication factor (k_{eff}) of the fuel in the cask. The CSAS25 control module allows simplified data input to the functional modules BONAMI-S, NITAWL-S, and KENO V.a. These modules process the required cross sections and calculate the k_{eff} of the system. BONAMI-S performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL-S applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below 0.0016 for all calculations.

Validation and verification of the SCALE 4.4 computer system were performed. Criticality benchmarking calculations were performed.

The CSAS5 control module of SCALE6 [6.7] is used in the determination of the additional damaged fuel *configurations* performed with the NUHOMS®-61BTH DSC. The CSAS5 control module allows simplified data input to the functional modules BONAMI, NITAWL, and KENO V.a. These modules process the required cross sections and calculate the k_{eff} of the system. BONAMI-S performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below *0.0015* for all calculations.

The CSAS5 control module of SCALE6 [6.7] is also used in the determination of the maximum allowable U-235 wt. % enrichment for the GNF2 and ATRIUM 11 fuel assemblies. The USL determined for SCALE 6.0 of 0.9418 is applicable to the results obtained for these fuel assemblies.

T.6.4.1.2 Physical and Nuclear Data

The physical and nuclear data required for the criticality analysis include the fuel assembly data and cross-section data as described below.

Table T.6-3 lists the pertinent data for criticality analysis with the GE12 10x10 fuel assembly in the NUHOMS®-61BTH DSC as loaded in a generic cask described in Section T.6.1.

The criticality analysis used the 44-group cross-section library built into the SCALE system. ORNL used ENDF/B-V data to develop this broad-group library specifically for criticality analysis of a wide variety of thermal systems.

T.6.4.1.3 Bases and Assumptions

The analytical results reported in Section T.3 demonstrate that the cask containment boundary and canister basket structure do not experience any significant distortion under hypothetical accident conditions. The fuel assembly drop analyses documented in Section T.3-5 also demonstrate that the fuel rods do not experience any deformation significant to cause a change in the fuel geometry. Therefore, for both normal and hypothetical accident conditions the cask geometry is identical except for the neutron shield and skin. As discussed above, the neutron shield and skin are conservatively modeled as water.

The cask was modeled with KENO V.a using the permissible geometry options. These options allow a model to be constructed with regular geometric shapes and define the material

boundaries. No cases have been made to model the fuel assemblies with fission products, burnable absorbers, or radial and axial variations in the initial fuel enrichment. Instead, fuel assemblies have been modeled as unirradiated fuel with a uniform enrichment. This results in a very large margin of conservatism in the calculated k_{eff}.

The following conservative assumptions were also incorporated into the criticality calculations:

1. Omission of grid plates, spacers, and hardware in the fuel assembly.

2. Unirradiated fuel - no credit taken for fissile depletion, fission product poisoning or burnable absorbers.

3. For intact fuel, the pins are modeled assuming a lattice average uniform enrichment everywhere in the lattice. Natural *uranium* blankets, *gadolinia*, *integral fuel burnable absorber* (IFBA), *erbia* or any other burnable absorber rods and axial or radial enrichment zones are modeled as enriched *uranium*, uniform everywhere. AMD 15 AMD 15 & 72.48

4. All fuel rods are assumed to be filled with 100% pure water in the fuel/cladding gap to account for the possibility of water being entrained in the fuel pin and because it has a slight positive effect on reactivity.

5. The fuel pellet stack was conservatively modeled at 96.5% of theoretical density with no allowance for dishing or chamfer. *For ATRIUM 11 fuel assemblies, the value is 98.045% of theoretical density. For GNF2 fuel assemblies, the value is 97% of theoretical density. Although the GNF2 damaged fuel assemblies are modeled with 96.5%. As demonstrated in Table 6-36 the reactivity impacts due to differences in the two values is statistically insignificant.* AMD 15

6. Water density at optimum internal and external moderator density.

7. Only the active fuel length of each assembly type is explicitly modeled. The presence of the plenum, end fittings, and channels above and below the active fuel reduce the k_{eff} of the system; therefore, these regions are modeled as water or the reflective boundary conditions. For the cases with reflective boundary condition, the model is effectively infinitely long. For intact fuel the active fuel region is conservatively assumed to start level with the bottom of the poison plates even though the fixed poison spans the entire length of the basket.

8. For all of the transportation *hypothetical accident conditions* (HAC) cases the neutron shield and stainless steel skin of the cask assumed to be replaced with external moderator. AMD 15

9. The least material condition (LMC) is assumed for the fuel compartment, poison plates and wrappers. This minimizes neutron absorption in the steel sheets and poison plates.

10. The maximum allowed gap between the poison plates in the worst case position is explicitly modeled to maximize k_{eff}.

conclusions about relative reactivity. Above all, the limit of 58 fueled rods (and 6 water rods) was arrived at so that the k_{eff} of this analyzed configuration is at least 3σ below the most reactive GE 10x10 fuel lattice.

A typical input file is included in Appendix K, Section K.6.6.2. The results of these calculations are listed in Table T.6-6. The most reactive fuel lattice evaluated for the canister design is the GE 10x10, without a fuel channel. *This result applies to the entire fuel assembly list in Table T.6-3 except GNF2 and ATRIUM 11. These fuel assemblies are evaluated individually in their most reactive configuration.*

B. Determination of the Most Reactive Configuration – Intact Fuel

The fuel-loading configuration of the canister/cask affects the reactivity of the package. Several series of analyses determined the most reactive configuration for the canister/cask.

For this analysis, the canister/cask is modeled over the active fuel height of the fuel assembly with reflective boundary conditions on all sides of the model. This represents an infinite array in the x-y direction of canister/casks that are infinite in length. The canister/cask model for this evaluation differs from the actual design in the following ways:

- the B-10 absorber loading in the poison plates is lower than specified,
- maximum gaps between poison plates are modeled in their worst case configuration,
- the stainless steel rails which hold the Type 1 DSC basket together are modeled as water,
- the rail structure for the Type 2 DSC basket are modeled using solid aluminum with water holes at the eight corner locations, and
- the neutron shield and the skin of the cask are conservatively modeled as water.

The models are fully described in Section T.6.3.1 except for the additional considerations for paired aluminum/poison plates and the representation of the Type 2 DSC basket. These additional modeling considerations are described in this section. The purpose of these models is to determine the most reactive configuration for intact fuel assemblies.

The first series of analyses determined the most reactive fuel assembly-to-assembly pitch. The GE 10x10 fuel assembly (determined in the previous section as the most reactive fuel assembly) with a lattice average fuel enrichment of 4.4 wt. % U-235 and a poison plate boron-10 loading of 36.0 mg/cm^2 are used in the model. The results in Table T.6-7 show the most reactive configuration occurs with minimum fuel assembly-to-assembly pitch. The model is similar to the model shown in Table K.6-4 and Figure K.6-2 of Appendix K, except that the nominal fuel cell size, nominal poison sheet thickness fuel clad OD are used, and the assemblies are moved within the fuel compartment to vary the fuel assembly-to-assembly pitch.

The second set of analyses evaluates the effect of canister shell thickness on the system reactivity. The model starts with the most reactive fuel assembly-to-assembly pitch (minimum pitch) case above and the canister shell thickness is varied from 0.49 to 0.55 inches. As demonstrated by the results the variation of shell thickness within the tolerance range is statistically insignificant. The nominal shell thickness is used throughout the rest of the analysis except that one additional case is added for the most reactive canister configuration (minimum poison plate thickness and minimum fuel cell size) to demonstrate that the slightly higher result for the maximum shell thickness is indeed a result of the statistics of the calculation.

by changing the Units 15, 17, 18, 19, 21, 22, 24, 25, 27, 29, 31, 50, 53, 55 and 57 such that the 0.300″ poison material is modeled as 0.125″ borated aluminum poison and the remaining as aluminum. The KENO Unit 25 is split into Units 125 and 225 so that the poison and aluminum are modeled separately. Similar treatment is accorded to Units 27, 29 and 31. This parametric evaluation scoping is also extended to include Boral® material as the poison. The Boral® material is modeled based on the specification shown in Table T.6-1 for a B-10 loading of 36 mg/cm^2 and a thickness of 0.064 *inch*. The results of this evaluation demonstrate that the effect of modeling the paired plates is statistically insignificant. The results also show that the effect of the various poison plate material specifications is also statistically insignificant as long as the amount of absorber material present in the model does not change. The statistically insignificant results arising due to the variation in the thickness of the poison plates (0.300-*inch*, 0.125-*inch* and 0.064-*inch*) and hence, the thickness of the aluminum plates indicates that there is no reactivity effect due to the modeling of cladding materials for the poison (like for Boral®). These results are shown in Table T.6-7.

The KENO model implementation of the paired plates has been effected in three ways in the same input model. For Units 15, 17, 18, 19, 21, 22 and 24 the poison in the paired plates is surrounded by aluminum. For Units 25, 27, 29 and 31 the poison and the aluminum are modeled as two distinct plates. For Units 50, 53, 55 and 57 the aluminum in the paired plates is surrounded by poison. Though the geometry in the actual basket for the paired plates is expected to be similar to what is modeled in Unit 25, this representation provides further insight to the results shown in Table T.6-7 and the conclusions herein with regard to variation in the poison plate and aluminum plate thicknesses. Therefore, treatment of paired plates does not result in any significant change in the system reactivity.

The ninth set of calculations determines the effect of basket rail modeling on the system reactivity. In order to obtain an acceptable, yet conservative model for both Type 1 and Type 2 DSC basket designs, the peripheral rails were modeled with solid aluminum and 7.5 cm (approximately 5.9″ diameter, about 15% less than the actual water volume fraction at those locations) water holes in the eight corner positions. The configuration is similar to that shown in Appendix K, Table K.6-4. The water holes (circular cross section) were also modeled using water squares to determine the effect due to the "hole" geometry. The results of this evaluation indicate that the assumption of solid aluminum rails with water holes conservatively bounds the internal moderator rails for the Type 1 DSC basket. It is also clear from the results that the use of solid aluminum as a rail material alone results in an overly conservative model. These results are shown in Table T.6-7. The water area calculations are shown in Section T.6.6.2.

Finally, minimum boron loading in the poison plate as a function of lattice average initial enrichment is evaluated. These models represent the most reactive intact fuel assembly (GE12, 10x10) with a minimum assembly-to-assembly pitch, nominal shell thickness, minimum paired plate thickness, minimum fuel clad OD, minimum fuel cell width with full internal and optimum external moderator density. Moreover, the calculational criticality analysis KENO model is also based on internal moderator gaps, paired plates with minimum poison thickness, solid aluminum rails with water holes that bounds both the Type 1 and Type 2 DSC basket designs.

Note that the criticality results for GNF2 and ATRIUM 11 fuel assemblies are obtained using the subject fuel assemblies instead of a representative model. The criticality analysis for the GNF2

fuel assembly is performed as a sensitivity to the GE12 10x10 fuel assembly on the basis of three DSC basket types – Types 1 and 2 with A, C, and F poison loadings. The results are provided in Table T.6-26 and Table T.6-27.

The analyses performed for the three DSC basket type poison loadings show that the maximum enrichments determined for the GE12 10x10 intact fuels are applicable to the GNF2 fuel assembly. These analyses demonstrate that the GNF2 fuel is similar in reactivity to the GE12 10x10 fuel; hence, the maximum enrichments determined for the GE12 10x10 fuel for the DSC basket type poison loadings B, D, and E are also applicable to the GNF2 fuel.

For the ATRIUM 11 fuel assembly, criticality evaluations are performed using the Type 2F basket, and the results indicate that the maximum allowable enrichment in is reduced by 0.55 wt. % U-235 from that allowed for the GE12 fuel assembly. The results are provided in Table T.6-28.

The *B-10* areal density in the poison plate (and hence thickness of the poison plate) is varied to determine the maximum lattice average fuel assembly enrichment. Thus, these cases can be

used to specify the minimum *B*-10 poison plate loading (and the appropriate thickness) as a function of maximum lattice average assembly enrichment. The results are reported in Table T.6-8. For selected poison plate loadings, the criticality analyses results are reported for Type 1 and Type 2 DSC baskets separately because it was necessary to maintain the same *B*-10 loading and maximum lattice enrichment as determined in Appendix K. In order to ensure that there is no significant change in the results due to poison type (as discussed in earlier evaluations), some of the most reactive cases are re-run with Boral® as the poison material. These cases are modeled with slightly lower poison loading so that appropriate conclusions can be drawn for the applicability of these results for all cases. One case with MMC is also analyzed. It may be noted that since the MMC poison material composition is similar to that of Boral® poison, additional runs with this poison for other thicknesses are not necessary.

The most reactive case is modified to model the paired plates "correctly" to determine if there is any significant variation due to poison material distribution within the basket. A comparison of the results indicates that the paired plate geometry as implemented in the design basis KENO model is adequate.

The Type 1 DSC basket is modeled similarly to the bounding model described above except that the rail material is based on internal moderator and not aluminum with water holes.

The dry case k_{eff} results for intact fuel are shown in Table T.6-8. The dry k_{eff} calculations were not performed for all the enrichment types but only for the most reactive fuel enrichment. This is done due to the assertion that the most reactive optimum moderator density configuration is most likely to yield the most reactive dry configuration and that the differences in reactivity at dry conditions are insignificant. The dry case k_{eff} result for shielding analysis to determine the sub-critical multiplication factor at an enrichment of 2.6 wt. % U-235 is also shown in Table T.6-8.

Note that for ATRIUM 11 fuel assemblies, a separate dry case is presented in Table T.6-8 along with the optimum moderator density configuration that will likely yield the most reactive dry configuration.

The KENO input file for the most reactive intact assembly case is listed in Section T.6.6.4.

C. Determination of the Most Reactive Configuration – Damaged Fuel

This section determines the most reactive configuration for the damaged fuel. This evaluation includes sensitivity calculations to determine the most reactive damaged fuel model with 7x7 and 8x8 lattices and then, subsequently, to determine the most reactive fuel assembly design. All the discussion and results for the 7x7 and 8x8 models (Case 1 through Case 17) are directly from Appendix K, Section K.6.6.3.

Five damaged fuel configurations are evaluated using two assembly arrays, 7x7 and 8x8, to demonstrate that a fuel assembly with up to seven fuel rods with gross cladding damage and a peak pellet enrichment of 4.4 wt. % U-235 and a lattice average of 4.0 wt. % U-235 will remain subcritical under all conditions of transfer and storage. These models evaluate the effects of radial movement of fuel rod pieces (the result of "single-ended" breaks), and axial movement (the result of "double-ended" breaks). The models all include water in the fuel pellet cladding

annulus. Section T.6.6.3 (almost identical to Appendix K, Section K.6.6.3) includes a more detailed description of these models.

GE 7x7 Array: The first two models, Case 1 and Case 2, are used to demonstrate that the difference between reflective and water boundary conditions on the ends has a minimal effect on

One important difference between the design basis intact assembly models and the damaged assembly models is in the treatment of axial boundary conditions. In the intact assembly models, the axial boundary conditions are reflective and therefore, the fuel assembly length was essentially infinite axially (conservative modeling). In the case of the damaged assembly models, a fuel assembly active length of 144″ was utilized with an additional 12.5″ of shifted damaged row of rods and water boundary conditions axially. The active fuel length of the GE 10x10 assembly is 150″ and even though it is expected to contain at least 6″ length of blankets, it is necessary to evaluate the effect of an increase in the active fuel length, if any, on the system reactivity.

Case ID 25 was modified to create the Case ID 26 where the active fuel length was increased to 150″. These results are also shown in Table T.6-11 and indicate that there is no significant change in the system reactivity due to the modeling of shorter active fuel length.

Case ID 23 was modified to create Case ID 27 where the outermost damaged rows of rods was modeled without cladding (clad replaced with internal moderator). This configuration is expected to result in an increase in k_{eff} because of the fact that fuel assemblies, in general, are undermoderated. Therefore, increasing the moderation by replacing the clad with moderator will result in an increase in k_{eff}. Variations to the above case included modeling the two outermost rows of rods with bare fuel, Case ID 28. The results of these cases are also shown in Table T.6-11. The most reactive configuration for the double shear cases is based on the Case ID 28, as expected, since it contains more bare rods.

The optimum rod pitch case, Case ID 29, was based on a modification to Case ID 23 except that there was no "UP" modeling included in the model. The "Dancoff" factor to be used to describe the damaged lattice is obtained from the rod array cases documented in Table T.6-10. A comparison of the k_{eff} obtained from this case, also shown in Table T.6-11, with the design basis double shear case from the previous evaluation (Case ID 28) clearly shows that the worst case damaged assembly configuration is based on the optimum rod pitch model.

Finally, minimum boron loading in the poison plate as a function of lattice average initial enrichment is evaluated. These models represent the DSC with the most reactive damaged fuel assemblies (GE12, 10x10, optimum pitch, 95 rods) for both the 4 and 16 assembly loading configurations with the most reactive configuration determined in the previous analyses. The remaining locations are loaded with the most reactive intact fuel assembly (GE12, 10x10) with the most reactive configuration determined in Section T.6.4.2B. The calculational criticality analysis KENO model is also based on internal moderator gaps, paired plates with minimum poison thickness, solid aluminum rails with water holes that bounds both the Type 1 and Type 2 DSC basket designs. Above all, all the damaged assembly criticality models for the paired plates are based on the "correct" arrangement (as described at the end of Section T.6.4.2B) of these plates. All the damaged assembly calculations are carried out with the borated aluminum poison.

Note that the criticality results for GNF2 and ATRIUM 11 fuel assemblies are obtained using the subject fuel assemblies instead of a representative model. The criticality analysis for the GNF2 fuel assembly is performed as a sensitivity to the GE12 10x10 fuel assembly on the basis of three DSC basket types – Types 1 and 2 with A, C, and F poison loadings. The results for four

damaged fuel assemblies are provided in Table T.6-29 and Table T.6-30, while for those with 16 damaged fuel assemblies are provided in Table T.6-32 and Table T.6-33.

The analyses performed for the three DSC basket type poison loadings show that the maximum enrichments determined for the GE12 10x10 damaged fuel are applicable to the GNF2 fuel assembly. These analyses demonstrate that the GNF2 fuel is similar to the GE12 10x10 fuel in reactivity; hence, the maximum enrichments determined for the GE12 10x10 fuel for the DSC basket type poison loadings B, D, and E are also applicable to the GNF2 fuel.

For the ATRIUM 11 fuel assembly, only up to four damaged fuel assemblies are authorized for storage with the remaining intact. The criticality evaluations are performed using the Type 2F basket and the results indicate that the maximum allowable enrichment is reduced by 0.55 wt. % U-235 from that allowed for the GE12 fuel assembly. The results are provided in Table T.6-31.

A sensitivity calculation was performed to determine the effect of the specification of the second lattice or the "Dancoff" factor in the criticality analyses. This is due to the fact that both the intact and damaged fuel assemblies can be specified as the second lattice. The "Dancoff" factor for the intact fuel assemblies is 2.6461172E-01 while that for the damaged fuel assemblies is 1.4377643E-01.

may slide into and increase the reactivity of the system as this region does not have basket poison plates. The length of 16 inches is conservatively chosen to bound the length of top grid axial length, which for current designs is maximum 14.5 inches (Appendix T.1). The most reactive configuration search includes the variation of assembly position inside the compartment and of the material filling around the failed fuel compartments in the 16 inches axial region outside the basket. This 16-inch axial region variation is evaluated using water, aluminum, or an approximate model of the top grid is used in order to cover a wide range of design variations that might affect the top grid region. The input models for configurations with 4 failed fuel assemblies in Table T.6-14 are constructed by changing the model of the design basis input for loading configuration with 4 damaged assemblies to represent the failed fuel loaded in the designated peripheral compartments in basket. Similarly, the input models for configurations with 4 failed fuel and 12 damaged fuel assemblies in Table T.6-19 are constructed by changing the model of the design basis input for loading configuration with 16 damaged assemblies to represent the failed fuel loaded in the designated peripheral compartments in basket. The KENO model plot of the 4 failed/57 intact assembly configuration is shown in Figure T.6-7. The input file listing for this case is provided in Section T.6.6.6. The KENO model plot of the 4 failed/12 damaged/45 intact assembly configuration is shown in Figure T.6-8. The input file listing for this case is also provided in Section T.6.6.6.

Finally, minimum boron loading in the poison plate as a function of lattice average initial enrichment is evaluated. These models represent the DSC with the most reactive failed fuel compartment configuration (GE, 10x10, optimum pitch, de-cladded rods, no rods removed, as shown in Table T.6-18), most reactive damaged fuel assemblies (GE, 10x10, optimum pitch, 95 rods, as shown in, Table T.6-10) in both loading configurations investigated, with 4 failed fuel compartments, and with 4 failed fuel compartments and 12 damaged assemblies. The remaining locations are loaded with the most reactive intact fuel assembly (GE12, 10x10) with a minimum assembly-to-assembly pitch, nominal shell thickness, minimum fuel clad OD, minimum poison thickness, minimum fuel cell width with full internal and optimum external moderator density. The calculational criticality analysis KENO model is also based on internal moderator gaps and solid aluminum rails with water holes that bounds Type 2 basket design. All the failed fuel calculations are carried out with the borated aluminum poison.

Note that the criticality results for GNF2 fuel assemblies are obtained using the subject fuel assemblies instead of a representative model. The criticality analysis for the GNF2 fuel assembly is performed as a sensitivity to the GE12 10x10 fuel assembly on the basis of three DSC basket types – Types 1 and 2 with A, C, and F poison loadings. The results are provided in Tables T.6-34 and T.6-35.

The analyses performed for the three DSC basket type poison loadings show that the maximum enrichments determined for the GE12 10x10 failed fuels are applicable to the GNF2 fuel assembly. These analyses demonstrate that the GNF2 fuel is similar to the GE12 10x10 fuel in reactivity; hence, the maximum enrichments determined for the GE12 10x10 fuel for the DSC basket type poison loadings B, D, and E are also applicable to the GNF2 fuel.

The *B*-10 areal density in the poison plate is varied to determine the maximum lattice average fuel assembly enrichment. Thus, these cases can be used to specify the minimum *B*-10 poison plate loading (and the appropriate thickness) as a function of maximum lattice average assembly enrichment. The results are reported in Tables T.6-20 and T.6-21.

The dry case calculations are performed for the most reactive initial enrichment/poison plate loading combination. The case selected for performing the dry calculations is based on the most reactive fully flooded case (100% internal and external moderator density). For the dry cases, which include evaluation at different internal moderator densities, the Dancoff factors are obtained from separate KENO runs.

Table T.6-1
Minimum B-10 Content as a Function of Enrichment

Poison ID	Maximum Lattice Average Initial Enrichment (wt. % U-235)			Minimum B-10 Content (gram/cm^2)		
	Intact Assemblies	Up to 4 Damaged Assemblies (Corner Locations)(1)(3)	5 or More Damaged Assemblies (Interior Locations)(1)	Utilized in this Analysis	Specified for 90% Credit	Specified for 75% Credit
Type 1 DSC						
A	3.70	3.70	2.80	0.019	0.021	0.025
B	4.10	4.10	3.10	0.029	0.032	0.038
C	4.40	4.40	3.20	0.036	0.040	0.048
D	4.60	4.60	3.40	0.043	0.048	0.058
E	4.80	4.80	3.50	0.050	0.055	0.066
F	5.00	5.00	3.60	0.056	0.062	0.075
Type 2 DSC						
A	3.70(2)	3.70(2)	2.80	0.020	0.022	0.027
B	4.10	4.10	3.10	0.029	0.032	0.038
C	4.40	4.40	3.20	0.038	0.042	0.050
D	4.60	4.60	3.40	0.043	0.048	0.058
E	4.80	4.80	3.50	0.050	0.055	0.066
F	5.00(2)	5.00(2)	3.60	0.056	0.062	0.075

		Up to 4 Failed Assemblies (Corner Locations)(1)(4)	Up to 4 Failed Assemblies (Corner Locations) and up to 12 Damaged Assemblies (Peripheral Locations)(1)			
A	3.70	3.70	3.00	0.020	0.022	0.027
B	4.10	4.00	3.10	0.029	0.032	0.038
C	4.40	4.40	3.40	0.038	0.042	0.050
D	4.60	4.60	3.40	0.043	0.048	0.058
E	4.80	4.80	3.40	0.050	0.055	0.066
F	5.00(3)	5.00(3)	3.50	0.056	0.062	0.075

Poison ID	57 Damaged Fuel at 3.30 wt. % U-235		Minimum B-10 Content (grams/cm2)		
	With 4 Intact Assemblies (Corner Locations)(1)	With 4 Damaged Assemblies (Corner Locations)(1)	Utilized in this Analysis	Specified for 90% Credit	Specified for 75% Credit
A	-	-	-	-	-
B	-	-	-	-	-
C	-	-	-	-	-
D	5.00	4.20	0.043	0.048	0.058
E	5.00	4.20	0.050	0.055	0.066
F	5.00	4.20	0.056	0.062	0.075

(1) See Figure T.2-9 of Chapter T.2 for the locations of these assemblies in the DSC.

(2) For ATRIUM 11 fuel assemblies, the U-235 wt. % enrichment is reduced by 0.55%. The ATRIUM 11 fuel assemblies are authorized for storage in the Type 2F DSC only.

(3) ATRIUM 11 fuel assemblies are authorized for storage only in the Type 2F basket with a maximum of four damaged and intact fuel assemblies.

(4) ATRIUM 11 failed fuel assemblies are not authorized for storage in the DSC.

Table T.6-2
Authorized Contents for NUHOMS®-61BTH System

General Electric 7x7 /GE1	7x7
General Electric 7x7 /GE2	7x7
General Electric 7x7 /GE3	7x7
Exxon/ANF 7x7 /ENC III-A	7x7
Exxon/ANF 7x7 /ENC III	7x7
General Electric 8x8 /GE4	8x8
General Electric 8x8 /GE5	8x8
General Electric 8x8 /GE-Pres	8x8
General Electric 8x8 /GE-Barrier	8x8
General Electric 8x8 /GE8 Type I	8x8
General Electric 8x8 /GE8 Type II	8x8
General Electric 8x8 /GE9	8x8
General Electric 8x8 /GE10	8x8
Exxon/ANF 8x8 /ENC Va and Vb	8x8
Framatome ANP 8x8-62/2	8x8
General Electric 9x9 /GE11	9x9
General Electric 9x9 / GE13	9x9
Siemens QFA	9x9
Framatome ANP 9x9-79/2 [FANP9]	9x9
General Electric 10x10/GE12	10x10
General Electric 10x10/GE14	10x10
Framatome ANP ATRIUM-10/ATRIUM-10XM	10x10
General Electric 10x10 – 92/2 / GNF2	*10x10*
AREVA ATRIUM 11 (1)	*11x11*

Table T.6-3
Parameters for BWR Assemblies for Shipment
(Part 1 of 2)

Manufacturer[1]	Array	Version	Active Fuel Length (in)	Number Fuel Rods per Assembly	Pitch (in)	Fuel Pellet OD (in)
GE	7x7	GE1	144	49	0.738	0.487
GE	7x7	GE2	144	49	0.738	0.487
GE	7x7	GE3	144	49	0.738	0.487
Exxon/ANF	7x7	ENC III-A	144	49	0.738	Note 2
Exxon/ANF	7x7	ENC III	144	48	0.738	Note 3
GE	8x8	GE4	146	63	0.640	0.416
GE	8x8	GE5	150	62	0.640	0.411[8]
GE	8x8	GE-Pres	150	62	0.640	0.411[8]
GE	8x8	GE-Barrier	150	62	0.640	0.411[8]
GE	8x8	GE8 Type I	150	62	0.640	0.411[8]
GE	8x8	GE8 Type II	150	60	0.640	0.411
GE	8x8	GE9	150	60	0.640	0.411
GE	8x8	GE10	150	60	0.640	0.411
Exxon/ANF	8x8	ENC Va and Vb	144	60	0.642	0.4195
Framatome ANP	8x8	8x8-62/2	150	62	0.641	Note 4
GE	9x9	GE11 GE13	146 - Full 50-146 - Partial	74	0.566	0.376
Framatome ANP	9x9	9x9-79/2	150	79	0.572	0.3565
Siemens	9x9	QFA	145.24	72	0.569	0.3737
GE	10x10	GE12 GE14	150 - Full 50-150 - Partial	92	0.510	0.345
Framatome-ANP	10x10	ATRIUM-10/ ATRIUM-10XM	150	91	0.510	0.3413/ 0.350[10]
GE	*10x10*	*GNF 10x10-92/2*	*150 – Full* *59.4-110.8 – Partial*	*92*	*0.510*	*0.350*
AREVA[12]	*11x11*	*ATRIUM 11*	*150 – Full* *55.9-88 – Partial*	*112*	*0.502*[11]	*0.320*

Table T.6-3
Parameters for BWR Assemblies for Shipment (Concluded)
(Part 2 of 2)

Manufacturer [1]	Array	Version	Clad Thickness (in)	Clad OD (in)	Water Rod OD[5] (in)	Water Rod ID (in)
GE	7x7	GE1	0.032	0.563	NA	NA
GE	7x7	GE2	0.032	0.563	NA	NA
GE	7x7	GE3	0.032	0.563	NA	NA
Exxon/ANF	7x7	ENC III-A	0.0355[6]	0.570	NA	NA
Exxon/ANF	7x7	ENC III	0.0355[6]	0.570	0.572[9]	NA
GE	8x8	GE4	0.034	0.493	0.493	0.425
GE	8x8	GE5	0.032	0.483	0.591	0.531
GE	8x8	GE-Pres	0.032	0.483	0.591	0.531
GE	8x8	GE-Barrier	0.032	0.483	0.591	0.531
GE	8x8	GE8 Type I	0.032	0.483	0.591	0.531
GE	8x8	GE8 Type II	0.032	0.483	2@0.591 2@0.483	2@0.531 2@0.419
GE	8x8	GE9	0.032	0.483	1.34	1.26
GE	8x8	GE10	0.032	0.483	1.34	1.26
Exxon/ANF	8x8	ENC Va and Vb	0.036	0.5015	0.5015[9]	NA
Framatome-ANP	8x8	8x8-62/2	0.035	0.484	0.484	0.414
GE	9x9	GE11	0.028	0.440	0.98	0.92
GE	9x9	GE13	0.028	0.440	0.98	0.92
Framatome-ANP	9x9	9x9-79/2	0.030	0.424	0.425	0.364
Siemens	9x9	QFA	0.0262	0.433	1.516[7]	1.458
GE	10x10	GE12	0.026	0.404	0.98	0.92
GE	10x10	GE14	0.026	0.404	0.98	0.92
Framatome-ANP	10x10	ATRIUM-10/ ATRIUM-10XM	0.0239/ 0.0244[10]	0.3957/ 0.405[10]	1.378[7]	1.321
GE	*10x10*	*GNF 10x10-92/2*	*0.0236*	*0.404*	*0.980*	*0.92*
AREVA [12]	*11x11*	*ATRIUM 11*	*0.0224*	*0.372*	*1.366*[7]	*1.299*

Notes:

(1) Reload fuel from other manufacturers with these parameters are also acceptable.
(2) Variable Fuel Pellet OD – evaluated from 0.468 to 0.488 in same assembly.
(3) Variable Fuel Pellet OD – evaluated from 0.468 to 0.491 in same assembly.
(4) Variable Fuel Pellet OD – evaluated from 0.4045 to 0.4055 in same assembly.
(5) Water rods for some fuel designs occupy more than one lattice position. Therefore, the number of water pin positions can be determined by subtracting the number of fuel rods from the total number of lattice positions for the array. As an example, the GE 10x10 fuel assembly has two water rods that occupy eight pin positions.
(6) Variable Fuel Clad Thickness – Thinnest clad thickness listed and conservatively used in the analysis.
(7) The water rod is more like a water box occupying 9 pin positions (3x3 pin array) and the ID and OD refer to Inside and Outside Dimensions of the box.
(8) Variable Fuel Pellet OD - evaluated from 0.410 to 0.411 in same assembly
(9) Solid Zirc rod(s)
(10) Value is specific to ATRIUM-10XM Fuel
(11) The pitch for the ATRIUM 11 FA class varies axially from 11.75 mm at the bottom spacer to up to 12.75 mm at the top spacer. The pitch reported is the maximum value of 12.75 mm, which is conservative.
(12) ATRIUM 11 fuel assemblies are authorized for storage only in the Type 2F basket with a maximum of 4 failed fuel assemblies.

Table T.6-4
Summary of Criticality Analyses

Description of Evaluation	Summary of Analyses	Reference
Determine the most reactive intact fuel assembly design	Reactivity of various GE, and Exxon (7x7 and 8x8) fuel assemblies are compared.	Appendix K.6
	Reactivity of the Framatome-ANP 8x8, 9x9, Siemens QFA and Atrium 10x10 fuel assemblies are compared. In addition, the number of fuel rods for GE 8x8 assembly is varied from 63 to 54 to allow these fuel assemblies with missing rods to be treated as intact assemblies.	Appendix T.6
Determine the most reactive configuration with intact fuel assemblies	The reactivity effect of the various DSC and fuel assembly geometry and material design parameters like assembly pitch, DSC shell thickness, poison thickness, fuel cladding OD, fuel cell width, internal and external moderator density are evaluated. KENO model described in Figure K.6.2 is developed.	Appendix K.6
	The reactivity effects of additional parameters like paired poison / aluminum plate thicknesses and rail material modeling for Type 2 DSC basket design are evaluated. The design basis criticality analysis KENO model shown in Figure T.6-3 for intact assemblies is developed.	Appendix T.6
Intact assembly criticality analyses	Using the design basis KENO model the lattice average enrichment of intact fuel assemblies as a function of fixed poison loading is determined.	Appendix T.6
Determine the most reactive damaged assembly mechanism and configuration	Reactivity of various postulated damaged assembly mechanisms for 7x7 and 8x8 fuel assemblies are compared using a minor modification to the KENO model described in Appendix K, Section K.6.6.3.	Appendix K.6
	The reactivity of the double shear mechanism for Siemens 9x9 QFA and the GE 10x10 fuel assemblies are compared using the KENO model described in Appendix T, Section T.6.6.3.	Appendix T.6
	The fuel rod pitch is varied for all assembly classes to determine the optimum pitch for each design. Additionally, rods are removed to determine the most reactive rod loading pattern within a lattice. This will determine the most reactive fuel assembly configuration for the rod pitch mechanism.	Appendix T.6
	The design basis criticality analysis KENO model for damaged assemblies is developed by synthesizing the KENO model geometries developed for most reactive intact configuration (basket and intact fuel) and the most reactive damaged configuration (rod pitch).	Appendix T.6
Damaged assembly criticality analyses	Using the design basis KENO model the lattice average enrichment of damaged fuel assemblies as a function of fixed poison loading for the 4 and 16 damaged assembly configurations is determined.	Appendix T.6
Determine most reactive failed assembly configuration	The most reactive damaged configuration is further evaluated for rod pitch and axial movement to determine the most reactive failed configuration.	Appendix T.6
Failed assembly criticality analyses	Using the design basis KENO model, the enrichment of failed assemblies as a function of fixed poison loading is determined.	Appendix T.6
57 damaged fuel assemblies at 3.3 wt. % U-235 with 4 either damaged at 4.2 wt. % U-235 or intact at 5.0 wt. % U-235.	The starting KENO model is the configuration found in Table T.6-12 with intact at 4.6 wt. % U-235 and Damaged at 5.0 wt. U-235, 43.2 mg B-10/cm2, B/Alum 0.153".	Appendix T.6
GNF2 and ATRIUM 11 intact fuel assemblies. *GNF2 and ATRIUM 11 4 damaged fuel assemblies.* *GNF2 16 damaged fuel assemblies.* *GNF2 4 failed & 4 failed + 12 damaged fuel assemblies.*	*The starting KENO model is the most reactive fuel assembly. Compare the reactivity of the GNF2 and ATRIUM 11 fuel assemblies with that of GE12. Utilize the models for the final results in Table T.6-8, T.6-12, and T.6-13.*	*Appendix T.6*

Table T.6-6
Most Reactive Fuel Type
(Continued)

Manufacturer	Array	Version	k_{KENO}	1σ	k_{eff}
GE	8x8	GE9, GE10	0.9043	0.0013	0.9069
GE	8x8 0.120 channel	GE9, GE10	0.9062	0.0013	0.9088
GE	8x8 0.080 channel	GE9, GE10	0.9054	0.0011	0.9076
GE	8x8 0.065 channel	GE9, GE10	0.9052	0.0014	0.9080
Exxon/ANF	8x8	ENC Va and Vb	0.8851	0.0011	0.8873
Exxon/ANF	8x8 0.120 channel	ENC Va and Vb	0.8827	0.0011	0.8849
Exxon/ANF	8x8 0.080 channel	ENC Va and Vb	0.8831	0.0012	0.8855
Exxon/ANF	8x8 0.065 channel	ENC Va and Vb	0.8821	0.0014	0.8849
GE	8x8 w/variable enrichment	GE5	0.8951	0.0011	0.8973
GE	8x8 w/variable enrichment	GE9	0.9008	0.0013	0.9034
GE	9x9	GE11, GE13	0.9042	0.0014	0.9070
GE	9x9 0.120 channel	GE11, GE13	0.9025	0.0014	0.9053
GE	9x9 0.080 channel	GE11, GE13	0.9066	0.0012	0.9090
GE	9x9 0.065 channel	GE11, GE13	0.9040	0.0013	0.9066
GE	**10x10**	**GE12, 14**	**0.9095**	**0.0013**	**0.9121**
GE	10x10 0.120 channel	GE12, 14	0.9094	0.0010	0.9114
GE	10x10 0.080 channel	GE12, 14	0.9092	0.0013	0.9118
GE	10x10 0.065 channel	GE12, 14	0.9076	0.0011	0.9098
The results shown in this part of the table are based on CSAS25 runs.					
Framatome-ANP	8x8(4)	8x8-62/2	0.8991	0.0013	0.9017
Framatome-ANP	8x8 0.120 channel	8x8-62/2	0.8966	0.0012	0.8990
Framatome-ANP	8x8 0.080 channel	8x8-62/2	0.9005	0.0014	0.9033
Framatome-ANP	8x8 0.065 channel	8x8-62/2	0.8973	0.0013	0.8999
Framatome-ANP	9x9	9x9-79/2	0.9072	0.0015	0.9102
Framatome-ANP	9x9 0.120 channel	9x9-79/2	0.9065	0.0013	0.9091
Framatome-ANP	9x9 0.080 channel	9x9-79/2	0.9075	0.0013	0.9101
Framatome-ANP	9x9 0.065 channel	9x9-79/2	0.9054	0.0012	0.9078
Siemens	9x9	QFA	0.9078	0.0013	0.9104
Siemens	9x9 0.120 channel	QFA	0.9084	0.0012	0.9108
Siemens	9x9 0.080 channel	QFA	0.9085	0.0012	0.9109
Siemens	9x9 0.065 channel	QFA	0.9077	0.0012	0.9101
Framatome-ANP	10x10	Atrium-10/ Atrium-10XM(6)	0.9070/ 0.9071	0.0013/ 0.0011	0.9096/ 0.9093
Framatome-ANP	10x10 0.120 channel	Atrium-10/ Atrium-10XM(6)	0.9070/ 0.9059	0.0014/ 0.0011	0.9098/ 0.9081
Framatome-ANP	10x10 0.080 channel	Atrium-10/ Atrium-10XM(6)	0.9081/ 0.9067	0.0012/ 0.0010	0.9105/ 0.9087
Framatome-ANP	10x10 0.065 channel	Atrium-10/ Atrium-10XM(6)	0.9072/ 0.9058	0.0012/ 0.0013	0.9096/ 0.9084
GE	*10x10*	*GNF2 10x10-92/2(7)*	*0.9120*	*0.0011*	*0.9142*
GE	*10x10 0.120 channel*	*GNF2 10x10-92/2(7)*	*0.9097*	*0.0014*	*0.9125*
GE	*10x10 0.080 channel*	*GNF2 10x10-92/2(7)*	*0.9100*	*0.0010*	*0.9120*
GE	*10x10 0.065 channel*	*GNF2 10x10-92/2(7)*	*0.9120*	*0.0011*	*0.9142*
AREVA	*11x11*	*ATRIUM 11(7)*	*0.9452*	*0.0011*	*0.9474*
AREVA	*11x11 0.120 channel*	*ATRIUM 11(7)*	*0.9423*	*0.0012*	*0.9447*
AREVA	*11x11 0.080 channel*	*ATRIUM 11(7)*	*0.9438*	*0.0011*	*0.9460*
AREVA	*11x11 0.065 channel*	*ATRIUM 11(7)*	*0.9443*	*0.0012*	*0.9467*

Table T.6-6
Most Reactive Fuel Type

(Concluded)

Description[(5)]	**k_{KENO}**	**1σ**	**k_{eff}**
GE 4 fuel from 1st page of this table	0.8951	0.0013	0.8977
GE 4, 62 fueled rods	0.8966	0.0012	0.8990
GE 4, 61 fueled rods	0.8973	0.0013	0.8999
GE 4, 60 fueled rods	0.8971	0.0012	0.8995
GE 4, 59 fueled rods, alternate	0.8964	0.0013	0.8990
GE 4, 59 fueled rods, alternate	0.8955	0.0013	0.8981
GE 4, 59 fueled rods, alternate	0.9043	0.0014	0.9071
GE 4, 59 fueled rods	0.9007	0.0011	0.9029
GE 4, 58 fueled rods, alternate	0.9048	0.0013	0.9074
GE 4, 58 fueled rods	0.8965	0.0013	0.8991
GE 12/14 fuel from 2nd page of this table	**0.9095**	**0.0013**	**0.9121**

(1) Small fuel pellet OD (Note large pellet OD identical to GE1 analysis)
(2) Small fuel pellet OD
(3) Large fuel pellet OD
(4) Used maximum pellet OD
(5) For certain fueled rod configurations, alternate arrangements have also been analyzed.
(6) Evaluation performed at 95.5% of the theoretical density using SCALE5.0 [6.6]. Note that the Most Reactive Fuel Type analyses are carried out with 95% of the theoretical density for all the fuel types except for ATRIUM-10XM.
(7) Most reactive fuel type analyses are carried out with 95% of the theoretical density for all the fuel types except for GNF2 and ATRIUM 11.

Table T.6-14
Criticality Results

Model Description	k_{KENO}	1σ	k_{eff}
Regulatory Requirements for Storage			
Dry Storage (Bounded by infinite array of undamaged casks)	0.4633	0.0004	0.4641
Normal Conditions (Wet Loading)	0.9378	0.0011	0.9400
Off-Normal Conditions (damaged transfer cask while fuel still wet)	0.9378	0.0011	0.9400
Design Basis Cases for Intact Fuel			
3.7 wt% U-235; 20.1 mgB-10/cm^2 100 % IMD	0.9368	0.0010	0.9388
4.1 wt% U-235; 28.8 mgB-10/cm^2 100 % IMD, Optimum EMD	0.9376	0.0012	0.9400
Design Basis Cases for Damaged Fuel			
Intact @ 4.4 wt% U-235; 4 Damaged @ 5.0 wt% U-235 37.5 mgB-10/cm^2, Type 2 100 % IMD	0.9371	0.0012	0.9395
Intact @ 4.1 wt% U-235; 16 Damaged @ 3.1 wt% U-235 28.8 mgB-10/cm^2, 100 % IMD, Optimum EMD	0.9375	0.0011	0.9397
Design Basis Cases for Failed Fuel			
Intact @ 5.0 wt% U-235; 4 Damaged @ 5.0 wt% U-235 56.2 mgB-10/cm^2, Type 2 100 % IMD, Optimum EMD	0.9375	0.0008	0.9391
Intact @ 4.1 wt% U-235; 12 Damaged and 4 Failed @ 3.1 wt% U-235, 28.8 mgB-10/cm^2, Type 2 100 % IMD, Optimum EMD	0.9374	0.0011	0.9396
ATRIUM 11 Fuel Assembly			
Dry Storage (Bounded by infinite array of undamaged casks)	*0.4844*	*0.0005*	*0.4854*
Normal Conditions (Wet Loading)	*0.9385*	*0.0011*	*0.9407*
Off-Normal Conditions (damaged transfer cask while fuel still wet)	*0.9385*	*0.0011*	*0.9407*

Table T.6-26
Criticality Analysis Results for GNF2 Intact Fuel – Type 1 DSC

EMD (%)	k_{KENO}	*1σ*	k_{eff}
Type 1 DSC only, intact fuel=3.7%, 18.9 mg B-10, B/Alum 0.102"			
1	*0.9306*	*0.0010*	*0.9326*
10	*0.9294*	*0.0011*	*0.9316*
30	*0.9321*	*0.0011*	*0.9343*
50	*0.9286*	*0.0010*	*0.9306*
70	*0.9301*	*0.0011*	*0.9323*
90	*0.9328*	*0.0013*	*0.9354*
100	*0.9295*	*0.0010*	*0.9315*
Type 1 DSC only, intact fuel=4.4%, 36.0 mg B-10, B/Alum 0.127"			
1	*0.9262*	*0.0010*	*0.9282*
10	*0.9262*	*0.0012*	*0.9286*
30	*0.9257*	*0.0010*	*0.9277*
50	*0.9256*	*0.0010*	*0.9276*
70	*0.9267*	*0.0011*	*0.9289*
90	*0.9269*	*0.0010*	*0.9289*
100	*0.9247*	*0.0012*	*0.9271*
Type 1 DSC only, intact fuel=5.0%, 56.2 mg B-10, B/Alum 0.199"			
1	*0.9283*	*0.0011*	*0.9305*
10	*0.9278*	*0.0011*	*0.9300*
30	*0.9241*	*0.0010*	*0.9261*
50	*0.9259*	*0.0014*	*0.9287*
70	*0.9238*	*0.0011*	*0.9260*
90	*0.9297*	*0.0011*	*0.9319*
100	*0.9269*	*0.0012*	*0.9293*

Table T.6-27
Criticality Analysis Results for GNF2 Intact Fuel – Type 2 DSC

EMD (%)	k_{KENO}	*1σ*	k_{eff}
Type 2 DSC only, Intact fuel=3.7%, 20.1 mg B-10, B/Alum 0.102"			
1	0.9291	0.0010	0.9311
10	0.9273	0.0009	0.9291
30	0.9293	0.0010	0.9312
50	0.9292	0.0009	0.9309
70	0.9292	0.0009	0.9310
90	0.9278	0.0009	0.9296
100	0.9291	0.0009	0.9309
Type 2 DSC only, intact fuel=4.4%, 37.5 mg B-10, B/Alum 0.132"			
1	0.9269	0.0009	0.9287
10	0.9286	0.0009	0.9304
30	0.9263	0.0009	0.9281
50	0.9272	0.0010	0.9291
70	0.9265	0.0009	0.9282
90	0.9267	0.001	0.9286
100	0.9251	0.001	0.9271
Type 2 DSC only, intact fuel=5.0%, 56.2 mg B-10, B/Alum 0.199"			
1	0.9283	0.0011	0.9305
10	0.9278	0.0011	0.9300
30	0.9241	0.0010	0.9261
50	0.9259	0.0014	0.9287
70	0.9238	0.0011	0.9260
90	0.9297	0.0011	0.9319
100	0.9269	0.0012	0.9293

Table T.6-28
Criticality Analysis Results for ATRIUM 11 Intact Fuel – Type 2 DSC

EMD (%)	k_{KENO}	1σ	k_{eff}
Type 2 DSC only, intact fuel=4.45%, 56.2 mg B-10, B/Alum 0.199"			
1	0.9385	0.0011	0.9407
10	0.9375	0.0012	0.9399
30	0.9342	0.0010	0.9362
50	0.9361	0.0010	0.9381
70	0.9375	0.0011	0.9397
90	0.9357	0.0013	0.9383
100	0.9385	0.0011	0.9407

Table T.6-29
Criticality Analysis Results for GNF2 (4 damaged) – Type 1 DSC

EMD (%)	k_{KENO}	1σ	k_{eff}
Type 1 DSC only, intact fuel = 3.7%, damaged=3.7%, 18.9 mg B-10, B/Alum 0.102"			
1	0.9289	0.0011	0.9311
10	0.9285	0.0011	0.9307
30	0.9300	0.0013	0.9326
50	0.9273	0.0010	0.9293
70	0.9310	0.0010	0.9330
90	0.9290	0.0010	0.9310
100	0.9296	0.0012	0.9320
Type 1 DSC only, intact fuel=4.4%, damaged=4.4%, 4 damaged, 36.0 mg B-10, B/Alum 0.127"			
1	0.9263	0.0011	0.9285
10	0.9278	0.0010	0.9298
30	0.9291	0.0014	0.9319
50	0.9262	0.0008	0.9278
70	0.9263	0.0011	0.9285
90	0.9274	0.0013	0.9300
100	0.926	0.0011	0.9282
Type 1 DSC only, intact fuel=5.0%, damaged=5.0%, 4 damaged, 56.2 mg B-10, B/Alum 0.199"			
1	0.9243	0.0012	0.9267
10	0.9264	0.0010	0.9284
30	0.9242	0.0010	0.9262
50	0.9238	0.0011	0.9260
70	0.9228	0.0012	0.9252
90	0.9228	0.0011	0.9250
100	0.9227	0.0010	0.9247

Table T.6-30
Criticality Analysis Results for GNF2 (4 damaged) – Type 2 DSC

EMD (%)	k_{KENO}	1σ	k_{eff}
Type 2 DSC only, Intact fuel=3.7%, Damaged=3.7%, 20.1 mg B-10, B/Alum 0.102"			
1	0.9302	0.0010	0.9322
10	0.9244	0.0010	0.9264
30	0.9298	0.0009	0.9317
50	0.927	0.0010	0.9290
70	0.9292	0.0008	0.9309
90	0.9304	0.0010	0.9324
100	0.929	0.0011	0.9312
Type 2 DSC only, intact fuel=4.4%, damaged=4.4%, 37.5 mg B-10, B/Alum 0.132"			
1	0.9272	0.0012	0.9296
10	0.9275	0.0010	0.9295
30	0.9275	0.0008	0.9291
50	0.9293	0.0011	0.9315
70	0.9291	0.0010	0.9311
90	0.9289	0.0012	0.9313
100	0.927	0.0011	0.9292
Type 2 DSC only, intact fuel=5.0%, damaged=5.0%, 56.2 mg B-10, B/Alum 0.199"			
1	0.927	0.0011	0.9292
10	0.9281	0.0011	0.9303
30	0.9297	0.0011	0.9319
50	0.9287	0.0012	0.9311
70	0.9305	0.0012	0.9329
90	0.9282	0.0012	0.9306
100	0.9297	0.0014	0.9325

All changes on this page are Amd 15.

Table T.6-31
Criticality Analysis Results for ATRIUM 11 (4 damaged) – Type 2 DSC

EMD (%)	k_{KENO}	*1 σ*	k_{eff}
Type 2 DSC only, intact fuel=4.45%, damaged=4.45%, 4 damaged, 56.2 mg B-10, B/Alum 0.199"			
1	*0.9351*	*0.0012*	*0.9375*
10	*0.9371*	*0.0011*	*0.9393*
30	*0.9337*	*0.0011*	*0.9359*
50	*0.9363*	*0.0012*	*0.9387*
70	*0.9335*	*0.0013*	*0.9361*
90	*0.9344*	*0.0010*	*0.9364*
100	*0.9346*	*0.0010*	*0.9366*

Table T.6-32
Criticality Analysis Results for GNF2 (16 damaged) – Type 1 DSC

EMD (%)	k_{KENO}	1 σ	k_{eff}
Type 1 DSC only, intact fuel=3.7%, damaged=2.8%, 16 damaged, 18.9 mg B-10, B/Alum 0.102"			
1	0.9290	0.0010	0.9310
10	0.9291	0.0013	0.9317
30	0.9303	0.0010	0.9323
50	0.9292	0.0011	0.9314
70	0.9302	0.0012	0.9326
90	0.9291	0.0008	0.9308
100	0.9315	0.0012	0.9339
Type 1 DSC only, intact fuel=4.4%, damaged=3.2%, 16 damaged, 36.0 mg B-10, B/Alum 0.127"			
1	0.9266	0.0012	0.9290
10	0.9271	0.0011	0.9293
30	0.9262	0.0010	0.9282
50	0.9251	0.0010	0.9271
70	0.9280	0.0009	0.9297
90	0.9273	0.0011	0.9295
100	0.9292	0.0013	0.9318
Type 1 DSC only, intact fuel=5.0%, damaged=3.6%, 16 damaged, 56.2 mg B-10, B/Alum 0.199"			
1	0.9263	0.0011	0.9285
10	0.9272	0.0013	0.9298
30	0.9267	0.0010	0.9287
50	0.9248	0.0011	0.9270
70	0.9256	0.0010	0.9276
90	0.9249	0.0009	0.9266
100	0.9253	0.0013	0.9279

Table T.6-33
Criticality Analysis Results for GNF2 (16 damaged) – Type 2 DSC

EMD (%)	k_{KENO}	*1 σ*	k_{eff}
Type 2 DSC only, intact fuel=3.7%, damaged=2.8%, 16 damaged, 20.1 mg B-10, B/Alum 0.102"			
1	*0.9311*	*0.0010*	*0.9331*
10	*0.9279*	*0.0011*	*0.9301*
30	*0.9277*	*0.0010*	*0.9297*
50	*0.9277*	*0.0014*	*0.9305*
70	*0.9284*	*0.0010*	*0.9304*
90	*0.9299*	*0.0011*	*0.9321*
100	*0.9280*	*0.0010*	*0.9300*
Type 2 DSC only, intact fuel=4.4%, damaged=3.2%, 16 damaged, 37.5 mg B-10, B/Alum 0.132"			
1	*0.9272*	*0.0012*	*0.9296*
10	*0.9289*	*0.0011*	*0.9311*
30	*0.9280*	*0.0009*	*0.9298*
50	*0.9274*	*0.0011*	*0.9296*
70	*0.9302*	*0.0011*	*0.9324*
90	*0.9286*	*0.0011*	*0.9308*
100	*0.9260*	*0.0011*	*0.9282*
Type 2 DSC only, intact fuel=4.4%, damaged=3.6%, 16 damaged, 56.2 mg B-10, B/Alum 0.199"			
1	*0.9295*	*0.0010*	*0.9315*
10	*0.9294*	*0.0011*	*0.9316*
30	*0.9311*	*0.0010*	*0.9331*
50	*0.9293*	*0.0011*	*0.9315*
70	*0.9291*	*0.0011*	*0.9313*
90	*0.9294*	*0.0011*	*0.9316*
100	*0.9297*	*0.0009*	*0.9315*

Table T.6-34
Criticality Analysis Results for GNF2 (4 failed) – Type 2 DSC

EMD (%)	k_{KENO}	*1 σ*	k_{eff}
Type 2 DSC only, intact fuel=3.7%, failed=3.7%, 4 failed, 20.1 mg B-10, B/Alum 0.102"			
1	*0.9300*	*0.0011*	*0.9322*
10	*0.9276*	*0.0010*	*0.9296*
30	*0.9309*	*0.0010*	*0.9329*
50	*0.9296*	*0.0010*	*0.9316*
70	*0.9300*	*0.0010*	*0.9320*
90	*0.9297*	*0.0010*	*0.9317*
100	*0.9282*	*0.0010*	*0.9302*
Type 2 DSC only, intact fuel=4.4%, failed=4.4%, 4 failed, 37.5 mg B-10, B/Alum 0.132"			
1	*0.9297*	*0.0012*	*0.9321*
10	*0.9286*	*0.0012*	*0.9310*
30	*0.9276*	*0.0012*	*0.9300*
50	*0.9277*	*0.0010*	*0.9297*
70	*0.9277*	*0.0012*	*0.9301*
90	*0.9271*	*0.0010*	*0.9291*
100	*0.9298*	*0.0012*	*0.9322*
Type 2 DSC only, intact fuel=5.0%, failed=5.0%, 4 failed, 56.2 mg B-10, B/Alum 0.199"			
1	*0.9306*	*0.0011*	*0.9328*
10	*0.9261*	*0.0012*	*0.9285*
30	*0.9297*	*0.0011*	*0.9319*
50	*0.9276*	*0.0011*	*0.9298*
70	*0.9287*	*0.0011*	*0.9309*
90	*0.9315*	*0.0012*	*0.9339*
100	*0.9325*	*0.0012*	*0.9349*

Table T.6-35
Criticality Analysis Results for GNF2 (4 failed + 12 damaged) – Type 2 DSC

EMD (%)	k_{KENO}	*1 σ*	k_{eff}
Enrichment of intact fuel=3.7%, failed=3.0%, damaged=3.0%, 4 failed+12 damaged, 20.1 mg B-10, B/Alum 0.102"			
1	*0.9345*	*0.0010*	*0.9365*
10	*0.9312*	*0.0011*	*0.9334*
30	*0.9331*	*0.0011*	*0.9353*
50	*0.9307*	*0.0011*	*0.9329*
70	*0.9325*	*0.0010*	*0.9345*
90	*0.9337*	*0.0010*	*0.9357*
100	*0.9322*	*0.0010*	*0.9342*
Enrichment of intact fuel=4.4%, failed=3.4%, damaged=3.4%, 4 failed + 12 damaged, 37.5 mg B-10, B/Alum 0.132"			
1	*0.9300*	*0.0011*	*0.9322*
10	*0.9287*	*0.0011*	*0.9309*
30	*0.9299*	*0.0011*	*0.9321*
50	*0.9313*	*0.0009*	*0.9331*
70	*0.9306*	*0.0010*	*0.9326*
90	*0.9294*	*0.0010*	*0.9314*
100	*0.9308*	*0.0010*	*0.9328*
Enrichment of intact fuel=5.0%, failed=3.5%, damaged=3.5%, 4 failed + 12 damaged, 56.2 mg B-10, B/Alum 0.199"			
1	*0.9295*	*0.0009*	*0.9313*
10	*0.9315*	*0.0011*	*0.9337*
30	*0.9291*	*0.0011*	*0.9313*
50	*0.9274*	*0.0010*	*0.9294*
70	*0.9276*	*0.0012*	*0.9300*
90	*0.9259*	*0.0011*	*0.9281*
100	*0.9282*	*0.0012*	*0.9306*

Table T.6-36
GNF2 Theoretical Density Sensitivity Results

Case ID	All 61 fuel TD 0.97			All 61 fuel TD 0.965		
EMD (%)	k_{KENO}	1 σ	k_{eff}	k_{KENO}	1 σ	k_{eff}
GNF2 Fuel in Type 1A Basket with 16 Damaged Fuels						
1	0.9298	0.0011	0.9320	0.9294	0.0010	0.9314
10	0.9293	0.0009	0.9312	0.9286	0.0011	0.9308
30	0.9286	0.0010	0.9306	0.9307	0.0010	0.9327
50	0.9287	0.0010	0.9307	0.9281	0.0010	0.9301
70	0.9297	0.0010	0.9317	0.9287	0.0010	0.9307
90	0.9281	0.0010	0.9301	0.9280	0.0012	0.9304
100	0.9311	0.0012	0.9335	0.9285	0.0011	0.9307
GNF2 Fuel in Type 1A Basket with 4 Failed and 12 Damaged Fuels						
1	0.9345	0.0010	0.9365	0.9319	0.0010	0.9339
10	0.9312	0.0011	0.9334	0.9342	0.0011	0.9364
30	0.9331	0.0011	0.9353	0.9337	0.0010	0.9357
50	0.9307	0.0011	0.9329	0.9299	0.0010	0.9318
70	0.9325	0.0010	0.9345	0.9309	0.0011	0.9331
90	0.9337	0.0010	0.9357	0.9317	0.0011	0.9339
100	0.9322	0.0010	0.9342	0.9310	0.0010	0.9329

T.9 Acceptance Tests and Maintenance Program

Background for this particular UFSAR chapter:

Beginning with CoC 1004 Amendment 10, which was incorporated into UFSAR Revision 11, Chapter T.9, "Acceptance Tests and Maintenance Program," contained information which was incorporated by reference into the Technical Specifications (TS) associated with a particular amendment. It is known that certain general licensees reconcile the CoC 1004 UFSAR revisions provided to them to their loaded systems, pursuant to 10 CFR 72.48 and 10 CFR 72.212. In doing so they sometimes find the changed UFSAR portions incorporated by reference into the TS to be impossible to reconcile because the 10 CFR 72.48 regulation does not allow proposed activities which involve changes to the TS.

In order to facilitate this reconciliation process by general licensees, the following statements are provided, addressing the licensing basis for certain amendments, as they relate to certain UFSAR chapters which contain TS incorporated by reference. Additionally, so that the actual information is contained in the current CoC 1004 UFSAR, to facilitate the reconciliation by general licensees, the UFSAR Revision 11, 12, 13, and 14 versions of Chapter T.9 are inserted and annotated in this part of the UFSAR. For clarity, this includes annotating the version of Chapter T.9 directly associated with the latest UFSAR revision in which a change to Chapter T.9 occurred.

- Systems loaded to CoC 1004 Amendment 10 have Technical Specifications incorporated by reference from UFSAR Revisions 11 and 12 in Chapter T.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to systems loaded to Amendment 10.
- Systems loaded to CoC 1004 Amendment 11 have Technical Specifications incorporated by reference from UFSAR Revision 13 Chapter T.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 11.
- Note that CoC 1004 Amendment 12 was submitted and docketed, associated with a U.S. Department of Energy project, but due to a lack of review funding the NRC returned it without a review.
- Systems loaded to CoC 1004 Amendment 13 have Technical Specifications incorporated by reference from UFSAR Revisions 14 and 15 Chapter T.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 13.
- Systems loaded to CoC 1004 Amendment 14 have Technical Specifications incorporated by reference by FCN 721004-1575, which will be incorporated into UFSAR Revisions 16 and 17 Chapter T.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 14.
- *Systems loaded to CoC 1004 Amendment 15 have Technical Specifications incorporated by reference from UFSAR Revision 18 Chapter T.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 15.*

T.9 Acceptance Tests and Maintenance Program

T.9.1 Acceptance Tests

The pre-operational testing requirements for the NUHOMS® system are given in Chapter 9.0, with the exceptions described in the following sections. The NUHOMS®-61BTH DSC has been enhanced to provide leaktight confinement and the basket includes an updated poison plate design. Additional acceptance testing of the NUHOMS®-61BTH DSC welds and of the poison plates are described.

T.9.1.1 Visual Inspection

Visual inspections are performed at the fabricator's facility to ensure that the DSC, the Transfer Cask and the HSM conform to the drawings and specifications. The visual inspections include weld, dimensional, surface finish, and cleanliness inspections. Visual inspections specified by codes applicable to a component are performed in accordance with the requirements and acceptance criteria of those codes.

All weld inspection is performed using qualified processes and qualified personnel according to the applicable code requirements, e.g., ASME or AWS. Non-destructive examination (NDE) requirements for welds are specified on the drawings provided in Chapter T.1; acceptance criteria are as specified by the governing code. NDE personnel are qualified in accordance with SNT-TC-1A [9.2].

The confinement welds on the DSC are inspected in accordance with ASME B&PV Code Subsection NB [9.1] including alternatives to ASME Code specified in Section T.3.1.2.3.

DSC non-confinement welds are inspected to the NDE acceptance criteria of ASME B&PV Code Subsection NG or NF, based on the applicable code for the components welded.

T.9.1.2 Structural

The DSC confinement boundary except the inner top cover/shield plug to the DSC shell weld is pressure tested at the fabricator's shop in accordance with ASME Article NB-6300. The test pressure is set between 14.5 to 16.0 psig for 61BTH DSC with Type 1 basket for future 10CFR71 application. This bounds the 1.1xDSC design pressure of 10 psig. The test pressure is set between 18.5 to 20.0 psig for 61BTH DSC with Type 2 basket for future 10CFR71 application. This bounds the 1.1xDSC design pressure of 15 psig.

The inner top cover/shield plug to the DSC shell weld is also pressure tested using a test pressure of between 14.5 to 16.0 psig for 61BTH DSC with Type 1 basket and between 18.5 to 20.0 psig for 61BTH DSC with Type 2 basket. This pressure test is performed at the field after the fuel assemblies are loaded in the DSC. This test is in accordance with the alternatives to the ASME Code specified in Section T.3.1.2.3.

HSM-H reinforcement and concrete are tested as described in Section 3.4.2.

T.9.1.3 Leak Tests

DSC confinement welds in the DSC shell and bottom are leak tested at the fabricator's shop to an acceptance criterion of $1x10^{-7}$ ref cm^3/s, i.e., "leaktight" as defined in ANSI N14.5 [9.4]. Personnel performing the leak test are qualified in accordance with SNT-TC-1A [9.2].

The weld between the DSC shell and inner top cover and the siphon/vent cover welds are also leak tested to an acceptance criteria of 1×10^{-7} ref cm^3/s in the field after the fuel assemblies are loaded in the canister.

T.9.1.4 Components

The Standardized NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. No other components of the NUHOMS® system require testing, except as discussed in this chapter.

T.9.1.5 Shielding Integrity

The Transfer Cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6" × 6" grid, the detector will encompass a 6" × 6" square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed.

The gamma and neutron shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

T.9.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section T.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section T.9.1.7.6.

T.9.1.7 Poison Acceptance

CAUTION

Sections T.9.1.7.1 through T.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated *a*luminum

(b) Boron carbide / *a*luminum metal matrix composite (MMC)

(c) *BORAL®*

The 61BTH DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content of these three types of materials is given in Table T.9-1.

References to metal matrix composites throughout this chapter are not intended to refer to *BORAL®*, which is described later in this section.

T.9.1.7.1 **Borated Aluminum**

See the Caution in Section T.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also occur)*. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section T.9.1.7.7. The specified acceptance testing assures that

at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

T.9.1.7.2 **Boron Carbide / Aluminum Metal Matrix Composites (MMC)**

See the Caution in Section T.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, *molten metal infiltration,* or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding *or produced by molten metal infiltration* shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B_4C particles in MMCs shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 61BTH DSC, MMCs shall pass the qualification testing specified in Section T.9.1.7.8, and shall subsequently be subject to the process controls specified in Section T.9.1.7.9.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section T.9.1.7.7. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

T.9.1.7.3 **BORAL®**

See the Caution in Section T.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns. The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

All changes on this page are AMD 13

The criticality calculations take credit for 75% of the minimum specified B10 areal density of BORAL®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken from the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

T.9.1.7.4 **Visual Inspections of Neutron Absorbers**

See the Caution in Section T.9.1.7.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. Blisters shall be treated as non-conforming. *For clad MMCs and f*or BORAL®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet. *Material that does not meet these criteria shall be reworked, repaired, or scrapped.*

T.9.1.7.5 Other Visual Inspections Criteria (non-Technical Specifications)

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

T.9.1.7.6 Thermal Conductivity Testing

*Acceptance t*esting shall conform to ASTM E1225[1], ASTM E1461[2], or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, and may be reduced if the first five tests meet the specified minimum thermal conductivity. *For cast products, the lot shall be defined by the heat or ingot. For other products, the lot shall be defined as material produced in a single production campaign using the same heat or lots of aluminum and boron carbide feed materials.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

[1] ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"

[2] ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The measured thermal conductivity values shall satisfy the minimum required conductivities as specified in Section T.4.3.

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section T.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

T.9.1.7.7 Specification for Acceptance Testing of Neutron Absorber *Content*

Acceptance testing for neutron absorber content shall be performed by either neutron transmission or by B-10 volume density measurement.

CAUTION

Portions of T.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

T.9.1.7.7.1 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

a) Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.1 inch diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard. Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than the statistically derived minimum thickness from T.9.1.7.7 a) or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

T.9.1.7.7.2 Specification for Acceptance Testing of Neutron Absorbers by B-10 Volume Density Measurement

a) B-10 volume density measurement acceptance testing procedures shall be subject to approval by the certificate holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, as long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for B-10 volume density measurements shall be such that there is at least one density measurement for each 2000 square inches of final product in each lot.

Areal density is determined by measuring the B-10 volume density in test samples and converting the measured values to areal density. The method of measurement of B-10 volume density shall be subject to approval by the certificate holder. The method of measurement of B-10 volume density shall be qualified against neutron transmission testing. Results of the two test methods shall be compared and a penalty shall be derived to account for the performance based results of neutron transmission testing.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B-10 areal densities are determined by volume density as described above. The lower tolerance limit of B-10 volume density is then determined, defined as the mean value of B-10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6]. Finally, the minimum specified value of B-10 areal density is divided by the lower tolerance limit of B-10 volume density to arrive at the minimum plate thickness that provides the specified B-10 areal density.

Any plate that is thinner than the statistically derived minimum thickness from T.9.1.7.7.2 a) or the minimum design thickness, whichever is greater, shall be treated as nonconforming, with the following exception. Local depressions are acceptable, as long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than

90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

T.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

CAUTION

Portions of Section T.9.1.7.8.4 and Section T.9.1.7.8.5 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

T.9.1.7.8.1 Applicability and Scope

Metal matrix composites (MMCs) acceptable for use in the 61BTH DSC are described in Section T.9.1.7.2.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section T.9.1.7.9 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the Certificate Holder.

T.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in Section T.9.1.7.8.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section T.9.1.7.8.5.

T.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for unclad MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below

842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for MMCs (clad or unclad) consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

All changes on this page are AMD 13

T.9.1.7.8.4 **Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity**

At least three samples, one each from approximately the two ends and middle of the qualification material run shall be subject to:

a) room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:

- **Minimum yield strength, 0.2% offset: 1.5 ksi**
- **Minimum ultimate strength: 5 ksi**
- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

b) Testing to verify more than 98% of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a means to be approved by the Certificate Holder. The maximum interconnected porosity is 0.5 volume %.

c) Delamination Testing of Clad MMC

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage. *An example of such a test would be: (1) immerse a specimen at least 6 x 6 inches in water under pressure ≥30 psig for at least 24 hours, (2) place the specimen in a vacuum furnace preheated to at least 300°F and evacuate the furnace. Acceptance criterion: no blistering or delamination of the cladding.*

3 Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998.

4 Boralyn testing submitted to the NRC under docket 71-1027, 1998.

5 ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.

6 ASTM E290, Standard Methods for Bend Testing of Materials for Ductility

T.9.1.7.8.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

Uniformity of the boron distribution shall be verified either by:

a) **Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or**

b) **Quantitative testing for the B10 areal density, B10 density, the boron carbide weight fraction, *or the boron weight fraction,* on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section T.9.1.7.7, or by chemical analysis for boron carbide *or boron* content in the composite.**

T.9.1.7.8.6 Qualification Report

Qualification report shall be prepared by, or subject to approval by the Certificate Holder.

T.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION

Sections T.9.1.7.9.1 and T.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

T.9.1.7.9.1 **Applicability and Scope**

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section T.9.1.7.8 is required, depending on the characteristics of the material that could be affected by the process change.

T.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, reduce corrosion resistance, reduce the mechanical strength or ductility of the MMC.

7 ASTM E94, Recommended Practice for Radiographic Testing

8 ASTM E142, Controlling Quality of Radiographic Testing

9 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

T.9.1.7.9.3 Identification and Control of Key Process Changes

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section T.9.1.7.9.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that are established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average *(d50)* particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a magnesium-alloyed aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, and

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

T.9.2 Maintenance Program

The NUHOMS®-61BTH system is a totally passive system and therefore will require little, if any, maintenance over the lifetime of the ISFSI. Typical NUHOMS®-61BTH system maintenance tasks will be performed in accordance with the UFSAR.

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T.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, *2004* Edition with *2006* Addenda.

9.2 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

9.3 *Deleted.*

9.4 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.5 "Aluminum Standards and Data, 2003," The Aluminum Association.

9.6 Natrella, "Experimental Statistics," Dover, 2005.

9.7 *Deleted.*

9.8 *Deleted.*

9.9 *Deleted.*

9.10 *Deleted.*

This version of Chapter T.9 is associated with CoC 1004 Amendment 15 and is from UFSAR Revision 18. Please see Page T.9 Introduction - 1 for a discussion as to why certain versions of Chapter T.9 are being maintained in the UFSAR.

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Table T.9-1
B10 Specification for the NUHOMS® 61BTH Poison Plates

Basket Type	*Specified Minimum B10 Areal Density for Borated Aluminum/MMC for 90% Credit (g/cm²)*	*Specified Minimum B10 Areal Density for BORAL® for 75% Credit (g/cm²)*
Type 1 DSC		
A	*0.021*	*0.025*
B	*0.032*	*0.038*
C	*0.040*	*0.048*
D	*0.048*	*0.058*
E	*0.055*	*0.066*
F	*0.062*	*0.075*
Type 2 DSC		
A	*0.022*	*0.027*
B	*0.032*	*0.038*
C	*0.042*	*0.050*
D	*0.048*	*0.058*
E	*0.055*	*0.066*
F	*0.062*	*0.075*

Table T.9-2
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This version of Chapter T.9 is associated with CoC 1004 Amendment 15 and is from UFSAR Revision 18. Please see Page T.9 Introduction - 1 for a discussion as to why certain versions of Chapter T.9 are being maintained in the UFSAR.

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Table T.9-3
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There is no change to the determination of the tornado wind and tornado missile loads acting on the Standardized HSM or HSM-H as detailed in Section T.2.2.1.

There is no change to the determination of the tornado wind and tornado missile loads acting on the HSM-H/HSM-HS as detailed in Appendix P, Section P.2.2.1.

T.11.2.3.2 Accident Analysis

An evaluation of the HSM and transfer cask with respect to tornado winds and tornado missiles is presented in Section 8.2.2. Changes to this analysis, as a result of the addition of the NUHOMS®-61BTH DSC, are presented in Section T.3.7.1. Evaluation of the Standardized HSM and TC with respect to tornado missile is also presented in Section 8.2.2. The tornado missile impact evaluation of the HSM-H is presented in the following sections.

The evaluation of the HSM-H for the effect of DBT wind pressure loads is addressed in Section P.3.7.1.1.

The missile impact analysis presented in Section P.11.2.3.2.1 is applicable here. Therefore, a loaded HSM-H rotates a maximum of 0.60° from vertical. The loaded HSM-H is stable against overturning as tip-over does not occur until the CG rotates past the edge point (point B, Figure T.11-1) to an angle of more than 24.65° [= $\tan^{-1}(52.0/118.77)$].

The tornado missile impact evaluation of the HSM-H/HSM-HS presented in Section U.11.2.3.2.1 is not changed.

T.11.2.3.3 Accident Dose Calculations

The increase in the dose rates at the localized impact location following the missile impact accident is expected to be bounded by the dose rates at the HSM-H vents, calculated to be 600 mrem/hour in Table T.5-1, since the structural analysis results demonstrate that there is no full penetration. This represents an increase in the *roof centerline dose rate* by a factor greater than 20 and is conservative.

For the purpose of this calculation, it is conservatively assumed that the affected area is twice the area of impact ~ 1.6 ft^2. The approximate surface areas at the HSM-H front is 140 ft^2, at the HSM-H roof is 200 ft^2 and that at the HSM-H side is 280 ft^2. *The impact area, therefore, represents approximately 0.6% to 1.2% of the surface area of the HSM-H, and the average dose rate on the surface of the impacted HSM will not increase appreciably. This increase does not significantly affect the ISFSI site dose rates and the results from Section T.10.2 for a 2x10 array of undamaged HSMs (specifically Table T.10-17) can be utilized to determine the exposure from a damaged HSM. This method is conservative because the missile impact will affect at most a single HSM, while a 2x10 array has approximately 20 front and 20 roof vents. The total dose rate is then the dose rate of the damaged HSM summed with the dose rate of the undamaged HSMs in the array, or twice the dose rate of the undamaged array using the conservative assumptions outlined above.*

*The dose received by a person located 100 meters away from the ISFSI for the assumed 8-hour duration would be less than 5 mrem (2*8 hours*Model 80 dose rate at 100m, 1.06E-01 mrem/hour) with a 2x10 array of HSMs. As an additional conservatism, Model 80 dose rates are used because these dose rates bound HSM-H dose rates. The dose to an offsite person located 500 meters away for the assumed 8-hour duration would be less than 0.01 mrem (2*8 hours*Model 80 dose rate at 500m, 5.25E-04 mrem/hour) with a 2x10 array of HSMs.*

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U.1.1 Introduction

The NUHOMS®-32PTH1 System is designed to store up to 32 (including reconstituted) B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14, and WE 14x14 class PWR fuel assemblies. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of *2.0* years. Each of the 32PTH1 DSC types is designed to store up to 32 Control Components (CCs) which include burnable poison rod assemblies (BPRAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), and neutron source assemblies (NSAs). The design characteristics, including physical and radiological parameters of the payload, are described in Chapter U.2.

Reconstituted assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or *an unlimited number of* lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods or Zr rods or Zr pellets or unirradiated stainless steel rods are acceptable for storage in 32PTH1 DSC as intact fuel assemblies. The maximum number of reconstituted fuel assemblies with irradiated stainless steel rods per DSC is four.

Provisions have been made for storage of up to 16 damaged fuel assemblies in lieu of an equal number of intact assemblies in the cells located at the center of the 32PTH1 basket. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. *The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means and the retrievability is ensured following the normal and off-normal conditions. The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is ensured following normal and off-normal conditions.* The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

Provisions have also been made for storage of up to *four failed fuel cans (*FFCs*)* in cells located at the corners of the interior 4x4 compartment cells of the 32PTH1 basket or *up to 16 FFCs* in a checkerboard pattern, as described in Chapter U.2.

The NUHOMS®-32PTH1 System consists of the following new or modified components:

- A 32PTH1 DSC, with three alternate configurations, described in detail in Section U.1.2, provides confinement, an inert environment, structural support, and criticality control for the 32 PWR fuel assemblies,
- A modified HSM-H module, described in Section U.1.2, is provided for environmental protection, shielding and heat rejection during storage, and
- OS200 or OS200FC TC for onsite transfer of the 32PTH1 DSCs.

The NUHOMS®-32PTH1 System requires the use of non-safety related auxiliary transfer equipment similar to those described in Section 1.3.2.2 (for OS200 TC) and Appendix P (for OS200FC TC) of the UFSAR. There is no change to any of the design features of the auxiliary transfer equipment except for the dimension changes necessary to accommodate the larger OS200 TC relative to the OS197 TC.

Approval of the NUHOMS®-32PTH1 System components described in Section U.1.2 is sought under the provisions of 10CFR 72, Subpart L for use under the general license provisions of 10CFR 72, Subpart K. The 32PTH1 system components are intended for storage on a reinforced concrete pad.

U.2.1 Spent Fuel To Be Stored

As described in Chapter U.1, there are three alternate design configurations for the NUHOMS®-32PTH1 DSC depending on the canister length: a short (185.75 in.) DSC designated as 32PTH1-S, a medium (193.00 in.) DSC designated as 32PTH1-M, and a long (198.5 in.) DSC designated as 32PTH1-L DSC. Each of the DSC configurations is designed to store intact (including reconstituted) and/or damaged and/or failed PWR fuel assemblies as specified in Table U.2-1 and Table U.2-3. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235. The maximum allowable assembly average burnup is limited to 62 GWd/MTU. The minimum required cooling time for fuel to be stored with *380, 475, and 492* kgU/FA is explicitly specified as a function of burnup and enrichment in *Technical Specifications Tables 1-3a through 1-3p.*

Each of the DSC types is designed to store control components (CCs) with thermal and radiological characteristics as listed in Table U.2-2. The CCs include burnable poison rod assemblies (BPRAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), neutron source assemblies (NSAs) and neutron sources. Furthermore, non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as guide tube or instrument tube tie rods or anchors, guide tube inserts, BPRA spacer plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.

The NUHOMS®-32PTH1 DSC is also authorized to store fuel assemblies containing blended low enriched uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiation and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding, and criticality evaluations for these rods are bounded by those of the regular fuel rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Reconstituted assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or *an unlimited number of* lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods, or Zr rods or Zr pellets, or unirradiated stainless steel rods are acceptable for storage in 32PTH1 DSC as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel replacement rods or 32 with UO_2 replacement rods.

The NUHOMS®-32PTH1 DSCs can also accommodate up to a maximum of 16 damaged fuel assemblies placed in the center cells of the DSC as shown in Figure U.2-1 through Figure U.2-3. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater than hairline cracks, or pinhole leaks, *including non-cladding damage*. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is being able to be handled by normal means and retrievability is assured following normal and off-normal conditions. *The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is ensured following normal and off-normal conditions.* The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

The NUHOMS®-32PTH1F DSC, an alternate version of the NUHOMS®-32PTH1 DSC, is designed to accommodate failed fuel in up to a maximum of four failed fuel cans (*FFCs*) placed in the corner cells of the interior 4x4 compartment cells of the basket, as shown in Figure U.2-5 or *up to 16 FFCs* in a checkerboard loading as shown in Figure U.2-3. Failed fuel is defined as fuel rods that have been removed from a fuel assembly and placed in a secondary container, such as a rod storage basket. Failed fuel may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means. Individual fuel rods that are not failed can be stored directly in the FFC without a secondary container such as an RSB. The maximum number of fuel rods that may be stored in a failed fuel can is 100, with a total uranium loading limited to 250 kg initial uranium. The total weight of the failed fuel can plus all its contents shall be less than 1715 lbs, 1625 lbs, and 1665 lbs, for the 32PTH1-L, 32PTH1-M, and 32PTH1-S DSCs, respectively.

The 32PTH1 DSC basket is designed with 3 alternate options: Type 1 basket with solid aluminum transition rails and Type 2 basket with steel transition rails including aluminum inserts, and Type 2-W, a variant of Type 2 with larger fuel compartments. Unless otherwise stated, all requirements for the Type 2 basket also apply to the Type 2-W variant, including requirements shown for baskets 2A through 2E. Type 1 basket is the preferred option for canisters with high decay heat loads, since the solid aluminum rails allow a more direct heat conduction path from the basket edge to the DSC shell.

The NUHOMS®-32PTH1 DSCs may store up to 32 PWR fuel assemblies arranged in any of the *six* alternate heat load zoning configurations (HLZC) as shown in Figure U.2-1 through Figure U.2-3 and Figure U.2-5 *through Figure U.2-7.* The maximum decay heat per fuel assembly and the maximum canister heat load allowed is as specified in Figure U.2-1 through Figure U.2-3 and Figure U.2-5 *through Figure U.2-7.* The *HLZCs 1, 5 and 6 with a maximum DSC heat load of 40.8 kW, 35.2 kW and 37.6 kW, respectively, are* applicable to Type 1 32PTH1 DSC only. HLZCs 2, 3, and 4 with a maximum DSC heat load of 31.2 kW, 24.0 kW and 31.2 kW, respectively, are applicable to both Type 1 and Type 2 32PTH1 DSCs. The maximum allowed heat load for the various 32PTH1 system configurations are presented in Table U.2-18.

In addition, the NUHOMS®-32PTH1 DSC basket is provided with three alternate neutron absorber plate materials (poison material) for criticality control: borated aluminum alloy, boron carbide/aluminum metal matrix composite (MMC) and Boral®. For criticality analysis, 90% of B10 content present in the borated aluminum and MMC poison plates is credited, while only 75% is credited for Boral®. The selection of the poison material does not have any impact on the thermal analysis, since it is based on the limiting thermal conductivity of Boral® as discussed in Section U.4.3.

Each of the two NUHOMS®-32PTH1 DSC basket types is analyzed for several alternate basket configurations for criticality control, depending on the boron loadings analyzed (designated as "A" basket for the lowest B-10 loading to "E" basket for the highest B-10 loading) and Basket-Type (Type 1 or Type 2).

The NUHOMS®-32PTH1 DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight with CCs that can be accommodated in the 32PTH1-L, 32PTH1-M, and 32PTH1-S is 1,715 lbs, 1,625 lbs, and 1,665 lbs, respectively.

The maximum fuel cladding temperature limit of 400°C (752°F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-32PTH1 DSC per Interim Staff Guidance (ISG) No. 11, Revision 3 [2.5]. In addition, ISG-11 does not permit repeated thermal cycling of the fuel cladding (limited to less than 10 cycles) with cladding temperature differences greater than 65°C (117°F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570°C (1058°F) is applicable to accidents or off-normal storage thermal transients [2.5].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, thermal and confinement. These evaluations are performed in Chapter U.5, U.6, U.4 and U.7 respectively. The fuel assembly classes considered are listed in Table U.2-3. It was determined that the B&W 15x15 *may be used as a representative fuel assembly for dose rate calculations.* For criticality safety, the B&W 15x15 assembly is the most reactive assembly type for a given enrichment. This assembly is used to determine the most reactive configuration in the DSC. Using this most reactive configuration, criticality analysis for all other fuel assembly classes is performed to determine the maximum enrichment allowed as a function of the soluble boron concentration and fixed poison plate loading. For thermal analysis, the WE 14x14 fuel assembly is limiting for the 32PTH1 DSCs, since it results in the lowest effective fuel thermal conductivity. The confinement analysis is based on B&W 15x15 fuel assembly, since it results in a smaller free volume inside the DSC cavity as compared to a 14x14 fuel assembly.

For calculating the maximum internal pressure in the NUHOMS®-32PTH1 DSC, it is assumed that 1% of the fuel rods are damaged for normal conditions, up to 10% of the fuel rods are damaged for off-normal conditions, and 100% of the fuel rods will be damaged following a design basis accident event [2.1]. A minimum of 100% of the fill gas and 30% of the fission gases within the ruptured fuel rods are assumed to be available for release into the DSC cavity, consistent with NUREG-1536 [2.1].

The maximum internal pressures used in the structural analysis for the NUHOMS®-32PTH1 DSC are 15, 20, and 140 psig for normal, off-normal and accident conditions, respectively, during storage and transfer operations for the 32PTH1 DSCs.

U.2.1.1 General Operating Functions

No change to Section 3.1.2.

Table U.2-1

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1aa.

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Table U.2-2

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1ee.

Table U.2-3

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1bb.

Table U.2-5

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-1dd.

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Table U.2-16
Summary of NUHOMS®-32PTH1 DSC and HSM-H Component Design Loadings(1)

Component	Design Load Type	SAR Section Reference	Design Parameters	Applicable Codes
32PTH1-DSC:	---	---	---	ASME Code, 1998 Edition with 2000 Addenda, Section III, Subsection NB and Appendix F (Shell) and Subsections NG, NF and Appendix F (Basket) with alternatives noted in Table U.3.1-1.
	Flood	U.2.2.2	Maximum water height: 50 ft and water velocity of 15'/sec.	10CFR72.122(b)
	Seismic	U.2.2.3	(a) Standard Seismic Criteria: 0.30g Horizontal and 0.25g *Vertical* (b) High Seismic Alternate: 1.0g Horizontal and 1.0g Vertical ground acceleration.	NRC Reg. Guides 1.60 [2.11] and 1.61 [2.12]
	Dead Load	U.3.6.1.2 U.3.6.1.3	Maximum enveloping weight of loaded 32PTH1 DSC: 110,000 lbs.	ANSI 57.9-1984
	Normal and Off-Normal Pressure	U.3.6.1.2 U.3.6.1.3	Enveloping internal pressure of ≤15 psig (Normal) and ≤ 20 psig (Off-Normal)	10CFR72.122(h)
	Test Pressure	U.3.6.1.2	Enveloping internal pressure of 23 psig applied w/o DSC outer top cover plate	10CFR72.122(h)
	Normal and Off-Normal Operating Temperature	U.3.6.1.2 U.3.6.1.3 U.3.6.2.2	Normal: Ambient air temperature 0°F to 106°*F* Off Normal: Ambient air temperature -40°F to 117°F	ANSI 57.9-1984
	Normal Handling Loads	U.3.6.1.2 U.3.6.1.3	1. Hydraulic ram load of 110,000 lb.(DSC HSM insertion) 80,000 lb (DSC HSM extraction) 2. Transfer (to/from ISFSI) Loads of: 2a. +/-1.0g axial 2b. +/-1.0g transverse 2c. +/-1.0g vertical 2d. +/-0.5g axial +/-0.5g transverse +/-0.5g vertical	ANSI 57.9-1984
	Off-Normal Handling Loads	U.3.6.1.2	Hydraulic ram load of: 110,000 lb(DSC HSM insertion) 80,000 lb(DSC HSM extraction)	ANSI-57.9-1984
	Accidental Cask Drop Loads	U.3.7.5	Equivalent static deceleration of 75g for horizontal side drops, and 25g oblique corner drop	10CFR72.122(b)
	Accident Internal Pressure	U.4	Enveloping internal pressure of ≤140 psig based on 100% fuel cladding rupture and fill gas release, 30% fission gas release, and ambient air temperature of 117°F	10CFR72.122(h)

Table U.2-17
B-10 Specification for the NUHOMS®-32PTH1 Poison Plates

NUHOMS®-32PTH1 DSC Basket Type	Minimum B-10 Areal Density, (grams/cm^2)	
	Borated Aluminum or MMC	Boral®
1A or 2A	0.007	0.009
1B or 2B	0.015	0.019
1C or 2C	0.020	0.025
1D or 2D	0.032	N/A
1E or 2E	0.050	N/A

Table U.2-18
Maximum Allowable Heat Load for the NUHOMS®-32PTH1 System

System Configuration	32PTH1 DSC Type	32PTH1 Basket Type(1),(2)	HSM Configuration	TC Configuration	Max. Heat Load (kW) per DSC
1	32PTH1-S, 32PTH1-M or 32PTH1-L	1A, 1B, or 1C or 1D or 1E	HSM-H/ HSM-HS	OS200FC	40.8 (HLZC 1, with intact or damaged fuel)
				OS200FC	31.2 (HLZC 2, with damaged fuel, HLZC 4 with intact and/or damaged, and/or failed fuel)
				OS200FC	*35.2 (HLZC 5 with intact fuel)*
				OS200FC	*37.6 (HLZC 6 with intact fuel)*
				OS200	31.2 (HLZC 2, with intact fuel, HLZC 4 with intact fuel)
				OS200	24 (HLZC 3 with intact or damaged fuel.)
2	32PTH1-S, 32PTH1-M or 32PTH1-L	2A, 2B, or 2C or 2D or 2E	HSM-H HSM-HS	OS200FC	31.2 (HLZC 2
				OS200	24.0 (HLZC 3)

Notes:

(1) Basket Type 1 (1A, 1B, 1C, 1D 1E) has aluminum transition rails in the DSC basket.

(2) Basket Type 2 (2A, 2B, 2C, 2D, 2E) has steel transition rails in the DSC basket.

The detailed information associated with this figure can be found in CoC 1004 Amendment *15* Technical Specifications Figure 1-28.

Figure U.2-3

The detailed information associated with this figure can be found in CoC 1004 Amendment *15* Technical Specifications Figure 1-28a.

Figure U.2-5

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-28b.

Figure U.2-6

The detailed information associated with this figure can be found in CoC 1004 Amendment 15 Technical Specifications Figure 1-28c.

Figure U.2-7

module-to-module separation.. These ties also restrain relative horizontal sliding (front-to-back), and vertical movement (in-phase tipping) between HSMs.

- As with the "standard" HSM-H, the "high seismic" HSM-Hs are not tied to the ISFSI pad and are allowed to slide freely during the seismic event; therefore, the ISFSI pad is designed such that an array made of high seismic HSM-Hs has 10 feet of space around all sides available for sliding and to facilitate retrievability of the 32PTH1 DSC, if necessary, following the postulated design basis earthquake.

U.3.1.1.4 General Description of the OS200 TC

The OS200 TC is based on the OS197/OS197H/OS197FC TCs described in the UFSAR Chapters 3, 4, 8 and Appendix P. Certain design features of the OS200 such as its increased diameter to accommodate the larger diameter 32PTH1 DSC and the neutron panel stiffener ring configuration are based on the OS187H TC described in [3.1].

The shell, or cask body cylinder assembly, is an open ended (at the top) cylindrical unit with an integral closed bottom end. This assembly consists of a concentric inner liner (SA-240, Type 304) and an outer shell (SA-240, Type 304) welded to massive closure flanges (SA-182, Type F304N) at the top and bottom ends. The annulus between the shells is filled with lead shielding. The top cover is bolted to the top flange using 24 1-1/2 in. diameter bolts.. A cover plate is provided to seal the bottom hydraulic ram access penetration of the cask (by 12-1/2 in. bolts with O-ring) during fuel loading.

Two lifting trunnions are provided for handling the transfer cask in the plant's fuel/reactor building using a lifting yoke and an overhead crane. Lower support trunnions are provided on the cask for pivoting the transfer cask from/to the vertical and horizontal positions and to support the cask on the skid/transfer trailer.

The gross weight of the loaded transfer cask is approximately 120 tons (240.0 kips) including a payload of 55.0 tons (110.0 kips). Table U.3.2-1 and Table U.3.2-2 summarize the 32PTH1 system component weights and OS200 TC component weights, respectively.

Similarly to the OS197FC TC, there is an option available for ambient air *to be* circulated at the bottom of the OS200FC TC through the ram access opening and distributed to the annular space between the DSC and the TC. The cooling air travels through the TC length and exists through the vent passages around the periphery of the TC top lid, in between the bolt holes. Drawings NUH-08-8001-SAR to NUH-08-8003-SAR describe the geometry and dimensions of the OS200 TC. These drawings are provided in Chapter U.1, Section U.1.5.

U.3.1.2 Design Criteria

The design criteria for the 32PTH1 DSC and OS200 TC and revisions to the HSM-H design criteria are provided in Chapter U.2, Section U.2.2.

Table U.3.1-1

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, Alternatives to the ASME Code for the NUHOMS-32PTH1 DSC Confinement Boundary.

The change on this page is a result of the issuance of Amendment 15.

Table U.3.1-2

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, Alternatives to the ASME Code for the NUHOMS-32PTH1 DSC Basket Assembly

Proprietary Information on Pages U.3.6-36 through U.3.6-36a
Withheld Pursuant to 10 CFR 2.390.

U.3.8 References

3.1. Safety Analysis Report, "NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel," NRC Docket No. 72-1030.

3.2. Updated Final Safety Analysis Report, CoC 72-1029 "Standardized Advanced NUHOMS® Horizontal Storage System for Irradiation Nuclear Fuel," ANUH-01.0150, Revision 2, August 2006, Docket No. 72-1029.

3.3. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Subsections NB, NF, NG, and Appendices 1998, with 2000 Addenda.

3.4. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section II, Part D, 1998 with 2000 Addenda.

3.5. Kaufman, J.G., ed., "Properties of Aluminum Alloys: Tensile, Creep, and Fatigue Data and High and Low Temperatures," The Aluminum Association (Washington, D.C.) and ASM International (Metals Park, Ohio), 1999.

3.6. Pacific Northwest Laboratory Annual Report – FY 1979, "Spent Fuel and Fuel Pool Component Integrity," May 1980.

3.7. G. Wranglen, "An Introduction to Corrosion and Protection of Metals," Chapman and Hall, 1985, pp. 109-112.

3.8. A.J. McEvily, Jr., ed., "Atlas of Stress Corrosion and Corrosion Fatigue Curves," ASM Int'l, 1995, p. 185.

3.9. TN Document No. 31-B9604.97-003, dated December 19, 1997; Addendum to TN West Document No. 31-B9604.0102, Rev. 2, "An Assessment of Chemical, Galvanic and Other Reactions in NUHOMS® Spent Fuel Storage and Transportation Casks."

3.10. Baratta, et al. "Evaluation of Dimensional Stability and Corrosion Resistance of Borated Aluminum," Final Report submitted to Eagle-Pitcher Industries, Inc. by the Nuclear Engineering Department, Pennsylvania State University.

3.11. "Hydrogen Generation Analysis Report for TN-68 Cask Materials," Test Report No. 61123-99N, Rev. 0, Oct 23, 1998, National Technical Systems.

3.12. W.C. Young, "Roark's Formulas for Stress and Strain," Sixth Edition, McGraw-Hill, New York, N.Y., 1989.

3.13. ANSYS Engineering Analysis System, Computer Code and Users Manual for ANSYS Release. 5.6 and 7.0, Swanson Analysis Systems, Inc., Houston, PA.

3.14. U.S. Nuclear Regulatory Commission (U.S. NRC), "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)," Regulatory Guide 3.48 (Task FP-029-4), (October 1981).

A total of *six (6)* heat load zoning configurations (HLZCs) are allowed for the 32PTH1 DSCs as shown in Figure U.4-1 through Figure U.4-3 and Figure U.2-5 *through Figure U.2-7*. The different DSC types, the maximum decay heat loads, and the applicable HLZCs are summarized in the following table. AMD 15

System Configuration	32PTH1 DSC Type	Basket Type	Max. Heat Load per DSC (kW)	HLZC Number	Transfer Cask	Storage Module
1	32PTH1-S or 32PTH1-M or 32PTH1-L	1A, 1B, 1C, 1D or 1E	40.8	1	OS200FC	HSM-H
			31.2	2, 4	OS200	
			35.2	*5*	*OS200FC*	
			37.6	*6*	*OS200FC*	
2(1)	32PTH1-S or 32PTH1-M or 32PTH1-L	2A, 2B, 2C, 2D or 2E	31.2	2, 4	OS200FC	
			24.0	3	OS200	

AMD 15

Note (1) Include Type 2-W DSCs as evaluated in Section U.4.A.

Borated aluminum, Boral® or MMC neutron absorbers can be used in 32PTH1 DSC.

The thermal evaluations presented herein include steady-state and transient analyses of the thermal response of the NUHOMS® 32PTH1 System components to a defined set of thermal operating conditions. These operating conditions envelop the thermal conditions expected during all normal, off-normal, and postulated accident operations during loading, transfer and storage as defined in Chapter U.2. The allowable temperatures for the 32PTH1 system components, are presented and comparisons are made with calculated temperatures as the basis for acceptance. 72.48

The thermal analysis methodology used for the 32PTH1 DSC, OS200 TC and HSM-H is the same as that used for the 24PTH system described in Appendix P, Chapter P4. A description of the analysis for HSM-H and OS200 TC containing 32PTH1 DSC for normal, off-normal, and accident conditions of storage and transfer is provided in Sections U.4.4 and U.4.5, respectively. Section U.4.6 describes the 32PTH1 DSC analysis for storage and transfer conditions for HLZCs #1, 2, and 3. Section U.4.9 describes the thermal analysis for HLZC #4. *Section U.4.11 describes the thermal analysis for HLZCs #5 and #6.* The DSC cavity internal pressures are also calculated in Section U.4.6 for all storage and transfer conditions. Section U.4.7 describes the evaluation performed for loading/unloading conditions. The thermal evaluation concludes that the NUHOMS® 32PTH1 System listed above meets all the design criteria. AMD 15

An evaluation of the effective thermal conductivity of the fuel assemblies to use in the 32PTH1 DSC thermal analysis is based on the methodology described in Appendix P, Section P.4.8. Section U.4.8 presents the evaluation of the fuel assembly and DSC basket effective thermal properties for a helium environment.

The methodology used to evaluate the thermal performance of the HSM-H was validated by thermal tests performed on a 1:1 scale of an HSM-H mockup structure for heat loads varying from 32 to 44 kW [4.31].

The thermal tests [4.31] showed that the HSM-H thermal analysis methodology described in Sections U.4.4.2 and U.4.4.3 conservatively predicts the DSC shell and HSM-H component temperatures. Bounding values are used in Chapter U.3 for the thermal stress evaluation for these components.

11. Water [4.18]

Temperature (°F)	Conductivity (Btu/hr-in-°F)	Specific Heat (Btu/lbm-°F)	Coef. of Exp., 1/°R	Density (lbm/ft^3)	Dynamic Viscosity (lbm/hr-ft)
44	2.803E-02	1.003	2.56E-05	62.428	3.440
62	2.879E-02	0.999	9.67E-05	62.366	2.613
80	2.951E-02	0.998	1.53E-04	62.241	2.068
98	3.024E-02	0.998	2.01E-04	61.994	1.681
116	3.082E-02	0.998	2.43E-04	61.749	1.396
134	3.130E-02	0.999	2.80E-04	61.445	1.183
152	3.178E-02	1.000	3.14E-04	61.144	1.016
170	3.217E-02	1.002	3.47E-04	60.787	0.883
188	3.245E-02	1.004	3.88E-04	60.375	0.784
206	3.269E-02	1.006	4.05E-04	59.969	0.699
224	3.289E-02	1.009	4.38E-04	59.512	0.629
242	3.303E-02	1.012	4.67E-04	59.006	0.573
260	3.313E-02	1.017	4.98E-04	58.508	0.525
296	3.313E-02	1.028	5.61E-04	57.379	0.448

12. Soil [4.19]

Conductivity (Btu/hr-in-°F)	Density (lbm/in^3)	Specific Heat (Btu/lbm-°F)
0.0144	0.0578	0.191

13. Concrete(1)

Temperature (°F)	Conductivity (Btu/hr-in-°F) [4.20]	Density (lbm/in^3)(1)	Specific Heat (Btu/lbm-°F) [4.30]
70	0.0958	0.084	0.22
1382	0.0479		

(1) Conservatively, the properties used herein for the HSM-H walls and roof do not include the effects of rebar. Additionally, the ISFSI pad is conservatively modeled as concrete and does not include the effects of rebar. Including the effects of rebar would increase the effective conductivity of the HSM-H concrete walls, roofs, and ISFSI pad.

OS200 TC Surface	Insolation (gcal/cm^2)	Applied heating averaged over 12 hours (Btu/hr-ft^2) with absorptivity
Cask Cylindrical Shell	400	72.15
Cask Vertical Ends	200	36.08

The sensitivity of the model to the thermal resolution was tested based on a preliminary thermal model that used model uses approximately 6,900 nodes, 4,560 solids, and 3,600 planar elements to define the cask body and DSC geometry. The peak cask temperature obtained for the loss of neutron shield accident was 3% lower than the results obtained with the current modeling based on the use of approximately 9,080 nodes, 6,225 solids, and 5,075 planar elements to define the cask body and DSC geometry. Based on the fact that a 30 to 40% increase in the modeling elements results in only a 3% change in the peak temperature it is concluded that the current modeling provides an accurate representation of the OS200 TC's thermal performance.

U.4.5.3 Description of Cases Evaluated for the OS200 TC

The thermal analyses of the OS200 TC are performed for the design basis ambient air temperatures defined in Section U.4.5.1. The evaluated cases include the vertical loading condition inside of the fuel handling facility, normal and off-normal horizontal transfer conditions without and with air circulation, and four accident scenarios. The first accident scenario evaluates the potential interruption of the air circulation system and establishes the time available to re-establish the air circulation, complete the transfer operation, or initiate some other recovery mode. The second accident scenario evaluated the potential loss of both the air circulation system and the water in the neutron shield. The evaluation establishes the transient heat up trend and the ultimate temperatures achieved under steady-state conditions. The third accident scenario involves a 15-minute hypothetical fire. The maximum duration of the fire event will be controlled under actual operations by administratively limiting the available fuel sources within the vicinity of the TC. The evaluation establishes the maximum temperatures reached as a result of the fire event, as well as the post-fire, steady-state conditions. The fourth final accident scenario involves an undamaged TC under an elevated ambient condition of 133 °F. The evaluation addresses the maximum steady-state temperatures that would be achieved should this accidental ambient condition occur.

Table U.4-4, Table U.4-5, and Table U.4-6 summarize the cases evaluated for OS200 TC with the 32PTH1 DSC with the HLZC #1, HLZC #2, and HLZC #3 heat load configurations, respectively. Since the OS200 TC is able to accommodate the HLZC #3 heat load configuration without air circulation, fewer design cases are needed to establish the thermal performance of the TC with this payload.

U.4.5.4 OS200 TC Thermal Model Results

The following sections present the predicted thermal results for the OS200 TC with the 32PTH1 DSC. The resultant DSC shell temperatures from these analyses are used as boundary conditions in the 32PTH1 DSC basket analysis presented in Section U.4.6, Section U.4.9, *and Section U.4.11.*

duration for the transfer operations (defined as the time elapsed after the initiation of draining of Cask/DSC annulus water until the completion of insertion of the DSC into the HSM-H) will vary depending on the DSC configuration and the heat load, and whether or not the air circulation option for the TC is utilized. The following table summarizes the permissible operational conditions:

Fuel Basket Type	DSC Heat Load Zoning Configurations	Transfer Time Limit (1) (2) (4)
Type 1 with intact fuel	HLZC 2 and 4 (≤31.2 kW)	No time limit
Type 1 with damaged fuel	HLZC 2 and 4 (≤31.2 kW)	38 hrs. (3)
Type 1	HLZC 1, *5 and 6* (≤40.8 kW)	13 hrs. (3)
Type 2 with intact fuel	HLZC 2 and 4 (24 kW to 31.2 kW)	14 hrs. (3)
Type 2 with damaged fuel	HLZC 2 and 4 (>24 kW to ≤31.2 kW)	10 hrs. (3)
Type 1 or Type 2	HLZC 3 (≤24 kW)	No time limit

Notes:

(1) Transfer time is defined as the time elapsed after the initiation of draining of cask/DSC annulus water until the completion of insertion of the DSC into the HSM-H.

(2) Initiate recovery operations such as air circulation if the operation time exceeds the limit.

(3) Initiate recovery operations such as air circulation if the operation time exceeds the limit. Two hours is considered sufficient time to initiate the air circulation option.

(4) The transfer operation time limit is reset only if the TC annulus is refilled with water.

U.4.6 NUHOMS® 32PTH1 DSC Thermal Analysis

The thermal analysis of the NUHOMS® 32PTH1 DSC is based on finite element models developed using the ANSYS computer code [4.22]. Two basket types are considered for 32PTH1 DSC. Type 1 basket and Type 2 basket are identical in all aspects, except for the transition rails. Type 1 basket transition rails are aluminum. The transition rails for Type 2 basket consist of stainless steel profiles with aluminum inserts. The transition rails for Type 1 and Type 2 baskets are shown in Figure U.4-48.

A total of *six* HLZCs are allowed for the 32PTH1 DSCs as shown in Figure U.4-1 through Figure U.4-3 and Appendix U.2, Figure U.2-5 *through Figure U.2-7*. Maximum of 16 damaged fuel assemblies can be stored in the 32PTH1 DSC.

This section discusses the thermal analysis of the 32PTH1 DSC with HLZCs #1 to #3. A fourth HLZC, authorizes storage of up to 16 damaged fuel assemblies, with up to four failed fuel assemblies stored in failed fuel cans (FFCs), while the remaining fuel compartments store intact fuel assemblies as shown in Appendix U.2, Figure U.2-5. The thermal analysis of HLZC # 4 is presented in Section U.4.9. As shown in Section U.4.9, the thermal analysis of HLZC # 4 is bounded by the thermal analysis of HLZC # 2 presented in this Section.

An additional evaluation that authorizes storage of up to 16 failed fuel assemblies based on HLZC # 3 is presented in Section U.4.10. As shown in Section U.4.10, the thermal analysis of HLZC # 3 with 16 failed fuel assemblies is bounded by the thermal analysis of HLZC # 3 presented in this section.

Section U.4.11 discusses the thermal analysis of the 32PTH1 Type 1 DSC with HLZCs #5 and #6 as shown in Appendix U.2, Figure U.2-6 and Figure U.2-7. As shown in Section U.4.11, the thermal analyses of HLZCs #5 and #6 are bounded by the thermal analysis of HLZC #1 presented in this section.

The DSC model with Type 1 basket is used for the analysis for total heat load up to 40.8 kW per canister with HLZC #1. The DSC model with Type 1 basket includes aluminum R45 and R90 rails. No time limit is required for transfer operation when the DSC with Type 1 basket contains a maximum heat load of 31.2 kW or less.

The DSC model with Type 2 basket is used for the analysis with total heat load of 31.2 kW per canister. This model includes R45 and R90 stainless steel rails with aluminum inserts. No time limit is required for transfer operation when the DSC with Type 2 basket contains a maximum heat load of 24.0 kW or less.

U.4.6.1 NUHOMS® 32PTH1 DSC Thermal Analysis Model

A three-dimensional (3D) finite element model of the 32PTH1 DSC is developed using the ANSYS computer code [4.22] to determine the maximum fuel cladding and DSC component temperatures. The 3D DSC model represents a longitudinally full-length, one-half (180°) cross section of the 32PTH1 DSC. This model includes the DSC shell, shield plugs, basket, and fuel assemblies.

The three-dimensional models representing the DSC with Type 1 and Type 2 baskets are shown in Figure U.4-45 through Figure U.4-48. The fuel assemblies are modeled as homogenized regions within the fuel compartments. The effective thermal properties for the intact and damaged fuels are calculated in Section U.4.8. The ANSYS models comprise the shell assembly (including the shell, and top and bottom end assemblies) and the basket assembly (including fuel compartment tubes, aluminum and neutron absorber plates, and the R45 and the R90 transition rails). All these DSC components are modeled using ANSYS SOLID70 elements. Radiation between the rails and the DSC shell is modeled using radiation LINK31 elements. Axial radiation is also considered between the top and bottom surfaces of the fuel assemblies to the shield plugs.

The DSC with Type 2 basket consists of steel R45 and R90 rails with aluminum inserts. The geometry of this model is identical to the one developed in [4.28] for use in the 32PTH HD SAR analysis. The DSC with Type 1 basket is similar to the DSC with Type 2 basket, except for the rails. The DSC with Type 1 basket consists of aluminum R45 and R90 rails. The other components of the model remain essentially unchanged.

Sufficient mesh refinement has been built into the model (see Section U.4.6.2). The total numbers of nodes and elements in the ANSYS model are approximately 600,000 for both

U.4.11 Thermal Evaluations of 32PTH1 Type 1 DSC Loaded with HLZCs #5 and #6

This section evaluates the thermal performance of the 32PTH1 Type 1 DSC based on HLZCs #5 and #6 during storage and transfer conditions with intact FAs. HLZCs #5 and #6 have maximum heat loads of 35.2 kW and 37.6 kW, as shown in Appendix U.2, Figure U.2-6 and Figure U.2-7, respectively. Because HLZCs #5 and #6 have higher maximum allowable heat loads than that for the 32PTH1 Type 2 DSC of 31.2 kW, they are only applicable to the 32PTH1 Type 1 DSC.

Since no other changes are considered to the 32PTH1 Type 1 DSC except for the HLZC, the thermal evaluation of 32PTH1 Type 1 DSC with HLZCs #5 and #6 is based on a sensitivity study of the bounding storage and transfer conditions described in Section U.4.11.1 and U.4.11.2, respectively. In addition, as shown in Figure U.2-6 and Figure U.2-7, HLZC #5 is bounded by HLZC #6 because HLZC #5 has both lower individual and total heat loads than those of HLZC #6. Therefore, the sensitivity study for the bounding storage and transfer conditions of the 32PTH1 Type 1 DSC is evaluated based on HLZC #6.

U.4.11.1 Storage of 32PTH1 Type 1 DSC in HSM-H

U.4.11.1.1 Bounding Storage Condition

According to Tables U.4-15, the normal hot storage with 106 °F ambient temperature is the bounding normal storage load case. [

]

U.4.11.1.2 Material

Material properties for fuel assemblies, stainless steel SA240 Type 304, aluminum 6061, helium, and concrete are listed in Section U.4.2. Material properties for stainless steel SA240 Type 316 and air are listed in Table 8-6 and Table 8-28 in Chapter 8 of the NUHOMS® EOS SAR [4.44], respectively. Material properties for carbon steel A36 are bounded by the values listed in Item 5 in Section U.4.2.

[ ]

U.4.11.1.3 Methodology

[

] *In addition to modeling the convection within the HSM-H cavity, the model also includes the thermal conduction within the basket and the HSM-H; radiation heat*

transfer among the DSC shell, heat shields, and HSM-H; and heat dissipation from the HSM-H and the vent outlet via convection and radiation to the ambient.

Section U.4.11.1.4 presents the computer-aided design (CAD) model and the meshes for the HSM-H with the 32PTH1 Type 1 DSC basket within an external air domain in ANSYS ICEM CFD [4.46]. The meshes are imported into ANSYS FLUENT [4.45] to develop a CFD model for thermal evaluation.

Section U.4.11.1.5 describes the detailed setup of the CFD model in ANSYS FLUENT [4.45].

U.4.11.1.4 *Computer-Aided Design and Meshing*

[ ]

Proprietary Information on Pages U.4-53u through U.4-53v
Withheld Pursuant to 10 CFR 2.390.

To ensure the high quality of the mesh, the guidelines on grids and grid design from NUREG-2152 [4.49] are considered in choosing the mesh parameters and techniques.

- *Maintain the expansion ratio between two consecutive cells below 1.3 in the regions where high gradients of temperature and velocity are expected or material changes. In solid regions, the expansion ratio is allowed to be 2 or higher.*
- *Avoid highly skewed elements with angles less than 45 degrees or larger than 135 degrees, especially in critical regions. In this mesh, around 99% of the elements have an angle between 45 and 135 degrees, and around 70% of the total elements have an angle between 81 and 99 degrees.*
- *Keep the aspect ratio of most elements less than 20 except for those in the near wall regions. In this mesh, 95% elements maintain the aspect ratio below 20, and 56% elements have the aspect ratio smaller than 5.*
- *Use finer and high-quality mesh in critical regions with high temperature and velocity gradients or with significant changes in geometry, such as regions near the DSC outer surfaces.*
- *Ensure sufficient resolution in the near-wall regions adjacent to the wall to capture the large variations in the flow. The low-Reynolds k-ε turbulence model in ANSYS FLUENT [4.45] enables the viscosity-affected region to be resolved with a mesh all the way to the wall. In the mesh model, a minimum of 15 elements is applied in the radial direction across the narrow region with 1.0" thickness around the outer surface of the DSC shell, where the first layer element is specified as 0.05" and grows with stretching factor smaller than 1.1. Furthermore, CFD results show that the average y^+ at the outer surface of 32PTH1 Type 1 DSC shell is 1.23. The dimensionless wall distance y^+ is defined as:*

$$y^+ = \frac{\rho y U_\tau}{\mu},$$

- *where ρ is the fluid density, μ is the fluid viscosity, y is the element size, $U_\tau = \sqrt{\tau_w / \rho}$ is the friction velocity, and τ_w is the wall shear stress.*

Proprietary Information on Pages U.4-53x through U.4-53gg
Withheld Pursuant to 10 CFR 2.390.

U.4.11.1.6 Results

U.4.11.1.6.1 Maximum Component Temperatures

The following table compares the maximum fuel cladding and DSC component temperatures for the 32PTH1 Type 1 DSC in the HSM-H with HLZC #6 to the design basis values presented in Table U.4-2, Table U.4-15, and Table U.4-16 for normal hot storage condition with 106 °F ambient. The design basis values presented for the 32PTH1 Type 1 DSC are based on the bounding temperatures determined for HLZC #1 with a maximum heat load of 40.8 kW for the normal hot storage condition with 106 °F ambient.

Maximum Component Temperatures for 32PTH1 Type 1 DSC in HSM-H for Normal Storage with 106 °F Ambient Temperature

Components	Design Basis, Tables U.4-2, U.4-15 and U.4-16	HLZC #6	ΔT ($T_{HLZC\#6} - T_{Design\ Basis}$)
Fuel Cladding	733		
Fuel Compartment	701		
Neutron Absorber	701		
R45 & R90 Rails	511		
DSC Shell	484		
Concrete	290		
Top Heat Shield	264		
Side Heat Shield	261		
DSC Support Rail	347		

As shown in the above table, the maximum temperatures of all components for the 32PTH1 Type 1 DSC and the HSM-H with HLZC #6 under the normal storage condition are bounded by design basis values listed in Table U.4-2, Table U.4-15 and Table U.4-16. Therefore, the design basis values in Table U.4-2, Table U.4-20 and Table U.4-21 for off-normal storage condition and Table U.4-3, Table U.4-24 and Table U.4-25 for accident blocked vent condition also remain bounding for HLZC #6.

The following figures show the temperature profiles for the fuel cladding, DSC shell and HSM-H and the velocity contours on the symmetry middle plane of the HSM-H model.

Proprietary Information on Pages U.4-53ii through U.4-53jj
Withheld Pursuant to 10 CFR 2.390.

U.4.11.1.6.2 GCI Calculation

[ ]

U.4.11.1.6.3 Maximum Internal Pressures

As shown in Section U.4.11.1.6.1, the maximum temperatures of all components for the 32PTH1 Type 1 DSC and the HSM-H with HLZC #6 are bounded by design basis values for HLZC #1. Therefore, the average helium temperatures determined for HLZC #6 are is also bounded by the design basis values listed in Table U.4-35 for determining the maximum internal pressures. The maximum internal pressures in Table U.4-19, Table U.4-23, and Table U.4-27 remain bounding for HLZC #6 under normal, off-normal, and accident storage conditions, respectively.

U.4.11.2 Transfer of 32PTH1 Type 1 DSC in OS200FC Transfer Cask

A review of Table U.4-15, Table U.4-16, Table U.4-20 and Table U.4-21 shows that the bounding transfer operation for the 32PTH1 Type 1 DSC in OS200FC TC is the vertical loading with HLZC #1 (40.8 kW heat load) and 140 °F ambient temperature, since it has the smallest margin to the fuel cladding temperature limit.

The thermal evaluations presented in Section U.4.5 for transfer operations are performed by considering the maximum heat load either as a heat flux on the radial inner surface of the DSC or as a volumetric heat generation rate applied over a homogenized basket. Because of this approach, the thermal evaluations presented in Section U.4.5 for transfer operations are not dependent on the HLZC, but are dependent only on the maximum heat load. Since the maximum heat loads considered for HLZCs #5 and #6 remain bounded by those previously evaluated for HLZC #1, the DSC shell temperature profiles from HLZC #1 remain applicable for HLZCs #5 and #6. Therefore, the DSC shell temperature profiles from the thermal evaluation results presented for the 32PTH1 DSC at 40.8 kW in Section U.4.5 are used as boundary conditions to evaluate the thermal performance of the 32PTH1 Type 1 DSC with HLZCs #5 and #6.

As discussed in Section U.4.11, HLZC #6 is the bounding HLZC selected for both HLZCs #5 and #6. To determine the most appropriate temperatures and time limit for a transfer operation with HLZCs #5 and 6, the 32PTH1 Type 1 DSC model with HLZC #1 (transfer limit of 15.75 hours) was re-evaluated using HLZC #6. No other changes are considered in this evaluation. The following table compares the maximum fuel cladding and DSC component temperatures of the 32PTH1 Type 1 DSC with HLZC #6 to the design basis values with HLZC #1 presented in Table U.4-20 and Table T.4-21 for the vertical transfer condition (DSC in TC, 140 °F ambient) with a time limit of 15.75 hours.

Maximum Component Temperatures for 32PTH1 Type 1 DSC for Vertical Transfer, 140 °F Ambient, 15.75 hours

Components	Design Basis, Tables U.4-20 and U.4-21	HLZC #6	ΔT ($T_{HLZC\#6} - T_{Design\ Basis}$)
Fuel Cladding	730		
Fuel Compartment	699		
Neutron Absorber	699		
R45 & R90 Rails	508		

The results of this evaluation show that the maximum temperatures of all components for the 32PTH1 Type 1 DSC with HLZC #6 under the vertical transfer condition are bounded by the design basis values with HLZC #1listed in Table U.4-20 and Table U.4-21. Therefore, the time limits for transfer operations determined for the 32PTH1 Type 1 DSC with HLZC #1 in Section U.4.5.5 are applicable to the 32PTH1 Type 1 DSC with HLZCs #5 and #6.

In addition, the average helium temperature determined for HLZC #6 is bounded by that of HLZC #1. The maximum internal pressures in Table U.4-19, Table U.4-23, and Table U.4-27 remain bounding for HLZC #6 under normal, off-normal, and accident transfer conditions, respectively.

Based on this discussion, no further evaluations are required for the 32PTH1 Type 1 DSC with HLZCs #5 and #6, and all design criteria described in Section U.4.1 are satisfied.

The following figures show typical temperature plots for the 32PTH1 Type 1 DSC with HLZC #6 for off-normal vertical transfer condition with 140 °F ambient, and without solar insolation at 15.75 hours.

Proprietary Information on This Page
Withheld Pursuant to 10 CFR 2.390

U.4.12 References

4.1. Report, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages," Office of Civilian Radioactive Waste Management, DOE/RW-0472, Revision 2, September 1998.

4.2. ASME Boiler and Pressure Vessel Code, Section II, Part D, Properties, 1998, including 2000 addenda.

4.3. Roshenow, W. M., J. P. Hartnett, and Y. I. Cho, "Handbook of Heat Transfer," 3rd Edition, 1998.

4.4. Bolz, R. E., G. L. Tuve, "CRC Handbook of Tables for Applied Engineering Science," 2nd Edition, 1973. Transfer, McGraw Hill, 1989.

4.5. NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.

4.6. Consolidated Safety Analysis Report for IF-300 Shipping Cask, CoC 9001.

4.7. Chun, Ramsey; Witte, Monika; Schwartz, Martin, "Dynamic Impact Effects on Spent Fuel Assemblies," Lawrence Livermore National Laboratory, Report UCID-21246, October, 1987.

4.8. NUREG/CR-0497, "MATPRO-Version 11: "A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor, Fuel Rod Behavior," EG&G, Inc. February, 1979.

4.9. Glasstone, S., Seasonske, A., "Nuclear Reactor Engineering," Third Edition, 1981.

4.10. Young, W. C., "Roark's Formulas for Stress and Strain," Sixth Edition, McGraw Hill.

4.11. Kreith, "The CRC Handbook of Thermal Engineering," 2000.

4.12. Standard Specification for Boral® Composite Sheet, AAR Advanced Structures, Livonia, Michigan.

4.13. Rohsenow, Hartnett, "Handbook of Heat Transfer Fundamentals," 2nd Edition, 1985.

4.14. Roth, A., "Vacuum Technology," 2nd Edition, 1982.

4.15. U.S. Nuclear Regulatory Commission Interim Staff Guidance No. 11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel," November 17, 2003.

4.16. SAND90 2806, Sanders, T. L., et al., "A Method for Determining the Spent Fuel Contribution to Transport Cask Containment Requirements," TTC1019, UC 820, November 1992.

4.17. Frank Kreith, "Principles of Heat Transfer," Third Edition, Harper and Row Publishers.

All changes on this page are Amd 15.

4.33. Azzazy Technology Inc., "Emissivity Measurements of 304 Stainless Steel," prepared for SCE, Report No ATI-2000-09-601, September 6, 2000, Transnuclear File No. NUH32PT-0100-01.

4.34. Section 4 of Amendment No. 1 to the Certificate of Compliance No. 1029 for the Standardized Advanced NUHOMS® Storage System, NRC Docket No. 72-1029, May 31, 2005.

4.35. ASHRAE Handbook Fundamentals 4th Edition, American Society of Heating, Refrigerating, and Air-Conditioning Engineers, New York, NY, 1983.

4.36. Guyer, "Handbook of Applied Thermal Design", McGraw-Hill, Inc., 1989.

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

4.38. Gubareff, G. G., Janssen, J. E. and Torborg, R. H. "Thermal Radiation Properties Survey", 2nd Edition, Honeywell Research Center, 1960.

4.39. McAdams, William H., "Heat Transmission", McGraw-Hill Book Company, New York, NY, 1954.

4.40. Desai, C. P. and Vafai, K., An Investigation and Comparative Analysis of Two- and Three-dimensional Turbulent Natural Convection in a Horizontal Annulus, International Journal of Heat and Mass Transfer, Vol. 37, No. 16, pp. 2475-2504, 1994.

4.41. K. Minato, et. al., "Thermal Conductivities of Irradiated UO_2 and (U, Gd)O_2 Pellets", Journal of Nuclear Materials, 300 (2002) 57-64.

4.42. C. Ronchi, et. al., "Effect of Burn-up on the Thermal Conductivity of Uranium Dioxide up to 100,000 MWd/t", Journal of Nuclear Materials, 327 (2004) 58-76.

4.43. J.M. Cuta, U.P. Jenquin, and M.A. McKinnon, "Evaluation of Effect of Fuel Assembly Loading Patterns on Thermal and Shielding Performance of a Spent Fuel Storage/Transportation Cask," PNNL-13583, Pacific Northwest National Laboratory, November 2001.

4.44. TN Document, "NUHOMS® EOS System Safety Analysis Report," Docket Number 72-1042, Rev. 7, July 2016

4.45. ANSYS FLUENT, Version 14.0, ANSYS, Inc.

4.46. ANSYS ICEM CFD, Version 14.0, ANSYS, Inc.

4.47. TN Document, "Updated Final Safety Analysis Report for the Standardized Advanced NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," ANUH-01.0150, Rev. 7, August 2016.

4.48. *ANSYS Design Modeler, Version 14.0, ANSYS, Inc.*

4.49. *U.S. NRC, Office of Nuclear Material Safety and Safeguards, "Computational Fluid Dynamics Best Practice Guidelines for Dry Cask Applications-Final Report," NUREG-2152, Rev. 0, March 2013.*

4.50. *I. E. Idelchik, Handbook of Hydraulic Resistance, 3rd Edition, Begell House, Inc., 1996.*

4.51. *S. Suffield, J. Cuta, J. Fort, B. Collins, H. Adkins and E. Siciliano, "Thermal Modeling of NUHOMS HSM-15 and HSM-1 Storage Modules at Calvert Cliffs Nuclear Power Station ISFSI," PNNL-21788, 2012.*

4.52. *American Society of Mechanical Engineers, "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer," ASME V&V 20-2009, November 30th, 2009.*

U.5 Shielding Evaluation

The radiation shielding evaluation for the Standardized NUHOMS® System (during loading, transfer and storage) for the other NUHOMS® canisters is discussed in other sections and appendices of the UFSAR. The following radiation shielding evaluation specifically addresses the shielding evaluation of the NUHOMS® 32PTH1 system with design-basis PWR fuel and control components (CCs) loaded in a NUHOMS® 32PTH1 DSC.

The radiation shielding evaluation described below is for the NUHOMS® 32PTH1 DSC transferred in a NUHOMS® OS200 Transfer Cask and stored in an HSM-H module. There are three alternate configurations depending on the canister length applicable to the 32PTH1 system: 1) 32PTH1-S, 2) 32PTH1-M and 3) 32PTH1-L. Each DSC has a different length and can have alternative rail configurations. The 32PTH1 DSC basket is designed with 3 alternate options: Type 1 basket with solid aluminum transition rails, Type 2 basket with steel transition rails including aluminum inserts, and Type 2-W, a variant of Type 2 with larger fuel compartments. The 32PTH1 DSCs are authorized to be transferred in the OS200 TC and stored in the HSM-H module. Note that HSM and HSM-H are used interchangeably throughout this chapter. For each 32PTH1 DSC there are *six* Heat Load Zoning Configurations (HLZCs) described in Chapter U.2, Figure U.2-1 through Figure U.2-3 *and Figure U.2-5 through Figure U.2-7*. The OS200 TC is similar to the OS197FC TC described in Appendix P with modifications to accommodate the 32PTH1 DSCs.

Of the three 32PTH1 DSC configurations the 32PTH1-L DSC has the least amount of axial shielding and is used for the OS200 TC and HSM shielding evaluation. Additionally, the DSC rail configuration is modeled to calculate conservative dose rates around the OS200 TC and HSM.

Each DSC configuration is designed to store up to 32 intact (and up to 16 damaged, with remaining intact) PWR fuel assemblies. The 32PTH1 DSCs are also designed to store up to 32 intact standard PWR fuel assemblies with or without Control Components (CC); such as burnable poison rod assemblies (BPRAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Thimble Plug Assemblies (TPAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Sources, and Neutron Source Assemblies (NSAs).

The design-basis PWR fuel source terms are derived from the bounding fuel, B&W 15x15 Mark B assembly design as described in Section U.5.2.

The NUHOMS® 32PTH1 DSCs are designed to store PWR fuel assemblies with CCs with the characteristic sources for CCs described in Table U.5-10. The 32PTH1 DSCs have a maximum decay heat of 1.5 kW per assembly and a maximum heat load of 40.8 kW per canister. Fuel in the 32PTH1 DSCs may be stored in *six* alternate heat zoning configurations as shown in Chapter U.2. Note that while the B&W, CE, and Westinghouse fuel designs are specifically listed, storing reload fuel designed by other manufacturers is also allowed provided an analysis is performed to demonstrate that the limiting

features listed in Chapter U.2, Table U.2-3 bound the specific manufacturer's replacement fuel. The limiting parameters are the design basis radiological and decay heat source terms.

The design-basis fuel source terms for this evaluation are defined as the source terms from fuel with the burnup/initial enrichment/cooling time combination that *results in* the maximum dose rate on the surface of the HSM and/or TC. This approach is consistent with the method used to generate the fuel qualification tables for the Standardized NUHOMS® 24P and -52B DSC designs as described in Section 7.2.3, 32PT DSC design as described in Appendix M or NUHOMS® 24PTH DSC design described in Appendix P. The design basis fuel source term in conjunction with the design basis CC source term (Table U.5-10) is used to calculate dose rates for the NUHOMS® 32PTH1 system.

The enveloping heat load zoning configuration (HLZC) utilized in the shielding evaluation is shown in Figure U.5-3 for all cases except for the 32PTH1 Type 2-W. This HLZC produces the highest dose rates on the surfaces of the HSM-H and OS200 TC as compared to HLZCs 1 *through 6* because the highest source fuel assemblies are on the outer periphery of the basket region where self-shielding due to adjacent assemblies is limited. To bound the shielding analysis for all HLZCs, fuel assemblies with a decay heat of 1.5 kW at the outer 28 locations are used along with 1.0 kW fuel assemblies in the central 4 compartments. This results in a shielding analysis corresponding to a total of 46 kW decay heat per DSC which is very conservative because the total decay heat in 32PTH1 DSC is limited to 40.8 kW. These bounding gamma and neutron source terms are then used in the radiation shielding models to conservatively calculate dose rates on and around the NUHOMS® 32PTH1 system.

The original 490 kgU fuel qualification tables (FQTs) developed for the 32PTH1 DSC using SCALE4.4/SAS2H have been replaced with FQTs for 380 kgU, 475 kgU, and 492 kgU developed by SCALE6.0/ORIGEN-ARP. These FQTs are documented in Section M.5.2.6.

In this chapter, the original design basis source terms developed by SCALE4.4/SAS2H, and the associated dose rate analysis, are retained as the analysis of record. However, SCALE6.0/ORIGEN-ARP design basis source terms are also developed consistent with the unified FQTs (Technical Specifications Tables 1-3a through 1-3p). These source terms are documented in Section U.5.2.6. The CC source terms from Table U.5-10 are added to the fuel source terms. The MCNP5 32PTH1 DSC/HSM-H and OS200 input files are rerun in MCNP5 using the SCALE6.0/ORIGEN-ARP design basis source terms to determine the impact on the dose rates. To minimize rework, the original analysis is maintained as the analysis of record; however, scaling factors are developed that are used to scale up the dose rates of the analysis of record. This updated shielding analysis that determines the effect of uranium loading (380 kgU per assembly) on the dose rates is summarized in Section U.5.4.12.

In the updated analysis, the decay heat in the central 4 compartments is modeled at 0.8 kW/FA and the decay heat in zone 2 is modeled at 1.3 kW/FA, for a total decay heat of 42.8 kW. This decay heat bounds the maximum decay heat of 40.8 kW and sufficiently bounds HLZC 1 through 6. This arrangement is also summarized on Figure U.5-3.

The bounding burnup, minimum initial enrichment and cooling time combinations for the fuel assemblies used in the shielding analyses of the 32PTH1 DSC in the HSM-H and the OS200 TC are as follows for the 32PTH1-M, 32PTH1-L, and 32PTH1-S DSC configurations:

- 32PTH1 DSC in HSM-H and OS200 TC inner 4 compartments (radial zone 1 in Figure U.5-3):
  62 GWd/MTU, 3.4 wt. % U-235, 20.5-year cooled fuel
- 32PTH1 DSC in HSM-H and OS200 TC middle 12 compartments (radial zone 2 in Figure U.5-3):
  62 GWd/MTU, 3.4 wt. % U-235, 8.5-year cooled fuel
- 32PTH1 DSC in HSM-H outer 16 compartments (radial zone 3 in Figure U.5-3):
  32 GWd/MTU, 2.6 wt. % U-235, 3.0-year cooled fuel
- 32PTH1 DSC in OS200 TC outer 16 compartments (radial zone 3 in Figure U.5-3):
  62 GWd/MTU, 3.4 wt. % U-235, 8.5-year cooled fuel

The 32PTH1 Type 2-W DSC is analyzed in the OS200 TC with a uniform 1 kW loading in all 32 compartments (Figure U.5-23): 62 GWd/MTU, 3.4 wt. % U-235, 20.5-year cooled fuel.

The method of selecting the bounding source terms is explained in detail in Section U.5.2.

The design basis CC source term that envelops all CCs allowed in the 32PTH1 DSCs is taken from Appendix M for BPRAs. This is the same CC source term used in the 24PTH system as

described in Appendix P. The source term energy distribution is shown in Table U.5-10. Any CC to be stored in a 32PTH1 DSC must be bounded by this source term.

Reconstituted and/or damaged fuel assemblies are also acceptable for storage in the 32PTH1 DSC. The maximum number of reconstituted fuel assemblies that can be loaded per DSC is 32. Fuel assemblies may contain up to 10 rods that are reconstituted with stainless steel that is irradiated. There is no limit on the number of rods reconstituted with unirradiated stainless steel or Zircalloy or low enriched UO_2 or other non-fuel material. There is no effect on the source terms/shielding due to the position of the reconstituted rods in the fuel rod array. Reconstituted fuel *with irradiated stainless steel rods* has a rather small effect on the dose rate such that for cooling times less than 10 years, 1 year of cooling time is added. *Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.* Damaged fuel, under normal conditions, has essentially no impact on the dose rate as the neutron and gamma source terms would not be impacted and gross axial source redistribution is not likely. Therefore, shielding analysis results with intact fuel are also applicable to the damaged fuel under normal conditions. For accident conditions, damaged fuel is assumed to redistribute and is analyzed separately.

The NUHOMS®-32PTH1F DSC, an alternate version of the NUHOMS®-32PTH1 DSC, is designed to accommodate failed fuel in up to a maximum of four failed fuel cans (*FFCs*) placed in the corner cells of the interior 4x4 compartment cells of the basket, as shown in Figure U.2-5, *or up to 16 FFCs, as shown in Figure U.2-3.* Failed fuel is defined as fuel debris or fuel rods that have been removed from a fuel assembly and placed in a secondary container, such as a rod storage basket. Failed fuel may contain breached rods, grossly breached rods, and other defective rods. The maximum number of fuel rods that may be stored in a rod storage basket is 100, with a total uranium loading less than 250 KgU. The total weight of the FFC plus all its contents shall be less than 1715 lbs, 1625 lbs, and 1665 lbs, for the 32PTH1-L, 32PTH1-M, and 32PTH1-S DSCs, respectively. The shielding analysis results with NUHOMS®-32PTH1 DSC are applicable to the NUHOMS®-32PTH1F DSC.

An upgraded version of the NUHOMS® HSM-H design, designated as NUHOMS® HSM-HS, is also provided to allow the use of the NUHOMS® System in locations where higher seismic levels exist. From a shielding standpoint, the HSM-HS module is identical to the HSM-H module. Therefore, all calculations performed with the HSM-H are applicable to the HSM-HS.

U.5.1 Discussion and Results

All 32PTH1 DSC, OS200 transfer cask and HSM-H dose calculations are performed using MCNP5 code [5.2] and a composite (hypothetical) DSC shielding configuration. The axial geometry of the 32PTH1-M DSC is used to accommodate the design basis B&W 15x15 Mark B fuel assembly. The shielding at the ends of the 32PTH1-M DSC is critically reduced to simulate the 32PTH1-L DSC axial shielding configuration. Steel rails are used for calculation of long term storage, transfer, welding and accident dose rates. The presence of solid aluminum rails that fill the space between the peripheral fuel compartments and the DSC shell results in a more effectively shielded configuration when the DSC is dry. Decontamination dose rates are calculated with solid aluminum rails which produce conservative results.

The 32PTH1 Type 2-W DSC is analyzed, with hollow steel rails, in normal transfer operation (dry DSC) and decontamination operations (wet DSC).

Table U.5-1 summarizes the maximum and average dose rates for the NUHOMS® 32PTH1 Design Basis (also referred to as "bounding") DSC loaded into the NUHOMS® HSM-H.

Table U.5-2 provides a summary of the dose rates on and around the OS200 TC for transfer of the 32PTH1 DSC under normal, off-normal and accident conditions.

Table U.5-3 provides a summary of the dose rates on and around the OS200 TC for decontamination and welding operations for the 32PTH1 DSC.

Table U.5-*21* summarizes the maximum dose rates on and around the OS200 TC for normal transfer operations for the 32PTH1 Type 2 -W DSC loaded at 32 kW decay heat. The 32PTH1 Type 2-W dose rates are bounded by those, in normal condition, shown in Table U.5-2 for the 32PTH1 DSC design basis loaded with enveloping heat load zoning configuration of 46 kW decay heat, Figure U.5-3.

Table U.5-*22* summarizes the maximum dose rates on and around the OS200 TC for decontamination operations for the 32PTH1 Type 2 - W DSC loaded at 32 kW decay heat. The 32PTH1 Type 2-W dose rates are bounded by those, in decontamination operations shown in Table U.5-3 for the 32PTH1 DSC design basis loaded with enveloping heat load zoning configuration of 46 kW decay heat, Figure U.5-3.

Based on the dose rates under normal condition and decontamination operations shown in Table U.5-*21* and Table U.5-*22*, it is expected that the 32PTH1 Type 2-W DSC dose rates under off-normal, accident conditions and in welding operations are also bounded by those obtained for the 32PTH1 DSC design basis in Table U.5-2 and Table U.5-3.

Therefore, the dose rates under normal, off-normal and accident conditions shown in Table U.5-2, and the dose rates for the decontamination and welding operations detailed in Table U.5-3, for the 32PTH1 DSC, are bounding and applicable to the 32PTH1 Type 2-W DSC. And by extension, the maximum and average HSM-H dose rates shown in Table U.5-1 for the 32PTH1 DSC design basis also apply to the 32PTH1 Type 2-W DSC.

The dose rates reported in Tables U.5-1 through U.5-3, Table U.5-21, and Table U.5-22 are scaled by footnotes to account for dose rate increases due to the unified FQTs and corresponding source terms. The unified FQTs are documented in Section M.5.2.6, and the corresponding source terms are documented in Section U.5.2.6. The scaling factors are developed in Section U.5.4.12.

A discussion of the method used to determine the design-basis fuel source terms is included in Section U.5.2. The design basis CC source term which is from Appendix M is shown in Table U.5-10. The shielding material densities are given in Section U.5.3. The method used to determine the dose rates due to design-basis fuel assemblies with CCs in the various NUHOMS® 32PTH1 DSC design configurations is provided in Section U.5.4. Radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The shielding evaluation is performed with the MCNP5 [5.2] code with the ENDF/B-VI cross section library. Sample input files used for calculating neutron and gamma source terms and dose rates are included in Section U.5.5.

The NUHOMS®-32PTH1 DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. **[**

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U.5.2 Source Specification

The original design basis source terms were generated using the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1]. The development of these source terms are provided in this section. The original SAS2H FQTs have been replaced with FQTs developed using SCALE6.0/ORIGEN-ARP [5.20]. The development of these FQTs is documented in Section M.5.2.6. Because the FQTs have changed, SCALE6.0/ORIGEN-ARP design basis source terms are developed consistent with the unified FQTs (Technical Specifications Tables 1-3a through 1-3p) to determine the impact on the dose rates. To minimize rework, the corresponding source terms developed using SCALE6.0/ORIGEN-ARP are documented in Section U.5.2.6. The thermal and radiological source terms for the CCs which are taken from Appendix M, are shown in Table U.5-10.

The B&W 15x15 assembly is the bounding fuel assembly design for shielding purposes because it has the highest CO-59 content of the hardware regions as compared to the 14x14, other 15x15, 16x16, and 17x17 fuel assemblies which are also authorized contents of the NUHOMS® 32PTH1 DSC. The neutron flux during reactor operation is peaked in the active fuel or in-core region of the fuel assembly and drops off rapidly outside the active fuel region. Much of the fuel assembly hardware is outside of the active fuel region of the fuel assembly. To account for this reduction in neutron flux, the fuel assembly is divided into four exposure "regions." The four axial regions used in the source term calculation are: the bottom (nozzle) region, the fuel (active fuel) region, the (gas) plenum region, and the top (nozzle) region. The B&W 15x15 fuel assembly masses for each irradiation region are listed in Table U.5-4. The light elements that make up the various materials for the various fuel assembly materials are taken from Reference [5.4] and are listed in Table U.5-5. The design-basis heavy metal weight is 0.490 MTU *in the analysis of record.* These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. To account for the reduction in neutron flux outside the active fuel regions neutron flux (fluence) scaling factors are applied to light element composition for each region. The neutron flux scaling factors which are from Reference [5.4] are given in Table U.5-6.

Evaluations of the existing data with SAS2H and the 44-group ENDF/B-V library used in the analysis are documented in References [5.11] and [5.12]. These comparisons all show generally good agreement between the calculations and measurements, and show no trend as a function of burnup in the data that would suggest that the isotopic predictions, and therefore neutron and gamma source terms, would not be in good agreement. A similar conclusion is also reached by the results documented in JAERI report [5.13]. In fact, for the case with 46,460 MWd/MTU burnup, the isotopic predictions are all within 2% of those measured. Therefore, the uncertainty in the gamma source term, and associated dose rates, is estimated to be within ±5 %.

The above discussion does not include high-burnup data up to 62 GWd/MTU. However, as documented in Reference [5.14] and confirmed in the SAS2H analysis, the total neutron source with increasing burnup is more and more dominated by spontaneous fission neutrons. Reviewing the output from the SAS2H runs, the neutron source term is due almost entirely to the spontaneous fission of Cm-244 (~94% of all neutrons both spontaneous fission and (α,n)). After reviewing the measured Cm-244 content compared to the Cm-244 content predicted by SAS2H and the 44-group ENDF/B-V library documented in References [5.11] and [5.12] for burnups up to 46,460 MWd/MTU, it is readily apparent that the calculated values are within ±11 % of the measured

values, with most of the predicted values within ±5% of the measured. Finally, there is no observed trend as a function of burnup in the data that would indicate that the predicted Cm-244 content is significantly different at higher burnups. Therefore, as the Cm-244 isotope accounts for more than 94% of the total neutron source term, the uncertainty in the neutron source and associated neutron dose rates is expected to be less than ±11 %.

As documented in Reference [5.14] and as observed in preparing the fuel qualification tables, the gamma radiation source strength increases nearly linearly with burnup relative to the direct gamma component and the neutron radiation source strength increases with burnup to the fourth power. Therefore, as burnups go beyond 45 GWd/MTU, the contribution from neutron (and associated n,γ) components to the total dose rates measured on the surfaces of the DSC, TC and HSM-H increase in relative importance to that of the gamma component. However, this increase in the importance of the neutron source term has a relatively minor effect on the area dose rates on and around the HSM as these are dominated by the gamma component as shown in Table U.5-1. The surface dose rates on the HSM are dominated by the gamma component because the HSM is constructed of thick reinforced concrete, which is an excellent neutron shield. Therefore, even a postulated substantial increase in the neutron source term would have a relatively minor effect on the site dose rate evaluation presented in Chapter U.11 of the amendment application.

For the TC, the neutron source term has a relatively minor effect on the area dose rates during most of the cask handling operations, since the DSC cavity and the annulus between the TC and DSC is filled with water and most of the work is done around the top of the cask. The neutron component is of more importance on and around the TC during transfer operations but, in general, only represents a small portion of the total dose rate on the top of the TC. While the neutron dose rate on the bottom of the TC is slightly higher, relatively little occupational dose is received from this area. The dose rates for the design basis fuel on the surfaces of HSM and TC are shown in Table U.5-1 through Table U.5-3. These tables show that gamma dose rates are substantially higher than neutron dose rates.

The occupational exposure calculations demonstrate that most of the dose received by workers during cask loading and transfer operations is due to the gamma radiation on and around the cask. The only surface of the TC that is dominated by neutrons is at the bottom of the cask. A small fraction of the total occupational exposure is due to the doses around the bottom of the cask because very little work is performed on or around the bottom of the cask with fuel in the TC.

As discussed above, any impact of uncertainties in source terms is expected to be negligible for the 32PTH1 system. Therefore, isotopic depletion calculations with SAS2H for fuel burned above 45 GWd/MTU are appropriate.

The above discussion on the applicability of SCALE 4.4/SAS2H to compute gamma and neutron high-burnup source terms is also largely applicable to SCALE 6.0/ORIGEN-ARP because both code systems utilize ORIGEN-S for the depletion calculation. For PWR fuel assemblies, SAS2H and ORIGEN-ARP generate comparable source terms for equivalent program inputs. The TRITON T-DEPL module of SCALE 6.0 is used to generate ORIGEN-ARP libraries applicable to B&W 15x15 fuel assemblies. These libraries are then used by ORIGEN-ARP to compute gamma and neutron source terms (see Section U.5.2.6).

Oak Ridge National Laboratory has benchmarked TRITON based on measured data from six different PWRs. This benchmarking is documented in NUREG/CR-6968 [5.17], NUREG/CR-7012 [5.18], and NUREG/CR-7013 [5.19] and includes measurement samples up to a burnup of 78.3 GWd/MTU. A summary of experimental samples utilized in the benchmark analysis is provided in Table U.5-23. The benchmark references show that TRITON computed results agree well with experiments, thus verifying the use of SCALE 6.0/ORIGEN-ARP to compute gamma and neutron source terms for high-burnup fuel (burnup ≤ 62 GWd/MTU). Because SAS2H and ORIGEN-ARP compute similar source terms for PWR fuel, the overall uncertainty in the gamma and neutron source terms developed above for SAS2H (±5% for gammas and ±11% for neutrons) is also applicable to the SCALE 6.0/ORIGEN-ARP generated source terms.

As discussed in Chapter U.5, reconstituted and/or damaged fuel is also acceptable for the DSC payload. AMD 15 Reconstituted fuel may contain up to 10 irradiated solid stainless steel rods that replace fuel rods. Reconstituted fuel has a rather small effect on the dose rate such that for cooling times less than 10 years, 1 year of cooling time is added if reconstituted irradiated stainless steel rods are present. If the cooling time is greater than 10 years, no additional cooling time is needed. AMD 15 & 72.48 *Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.* Additional discussion on the method used to analyze reconstituted fuel is provided in Section U.5.2.5. Under normal conditions, damaged fuel has essentially no impact on the dose rate as the source term would not be impacted and gross axial source redistribution is not likely. Damaged fuel under accident conditions is addressed by assuming the fuel turns to rubble. This assumption is only applicable to the transfer casks shielding analysis.

AMD 15 Parameters that influence the source term calculations are fuel assembly power (expressed in MW/Fuel Assembly (MW/FA)) and the total time between cycles. Other depletion parameters like cycle length and number of cycles are derived from the target burnup, MTU loading and specific power. The time between cycles utilized is 30 days and is adequately bounding. AMD 15

The design-basis source terms are defined as the burnup/initial enrichment/cooling time combination that result in the maximum dose rate on the surface of the HSM (HSM-H) or TC AMD 15 (OS200). Note that for a given HLZC, the design basis HSM source will not necessarily be the same as the corresponding design basis TC source. The 1-D discrete ordinates code ANISN [5.5] and the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.3] is used to determine the HSM and TC dose rate by radial zone for each entry in the fuel qualification tables and thereby determine the design basis source. As ANISN is a 1-D code, a single dose location must be selected for both the HSM and TC for analysis purposes. For the HSM, the middle of the roof centerline is selected as the dose location, and for the middle of the TC the cask side is selected as the dose location. This approach, described in detail in Section U.5.2.4, is consistent with the method used to determine the fuel qualification tables for the Standardized NUHOMS® canister designs described in Section 7.2.3 and Appendices M.5 and P.5. The radiological source terms generated in the SAS2H/ORIGEN-S runs are used in the ANISN evaluations to calculate the surface dose rates.

HLZC 1 (Figure U.2-1 in Chapter U.2) produced the bounding total surface dose rate for both the HSM-H and OS200 TC containing the 32PTH1 DSC. The enveloping HLZC selected for the shielding analysis (shown in Figure U.5-3) of the 32PTH1 DSC bounds the actual heat load configuration shown in Figure U.2-1 because 1.5 kW fuel is assumed in all 28 peripheral locations *in the analysis of record.* AMD 15

$$p(E) = C \exp(-E/a)\sinh(bE)^{1/2}$$

with input parameters a=0.906 MeV and b=3.848 MeV^{-1}, as given in the MCNP manual [5.2].

U.5.2.3 Axial Peaking

Axial burnup peaking factors for PWR fuel are taken from References [5.6] and [5.16]. These peaking factors are assumed to match the gamma axial source distribution because the gamma source is proportional to burnup. The neutron source is approximately proportional to the fourth power of the burnup. Therefore, the axial neutron source distribution may be determined as the fourth power of the axial burnup profile.

Axial peaking changes with increasing burnup. As the design basis source occurs at different burnups for the various decay heat and shielding configurations, different axial peaking factors are selected for the various TC and HSM calculations. The axial peaking factors used are provided in Table U.5-11. The OS200 TC calculations use peaking factors for a burnup >46 GWd/MTU because the design basis source for 1.5 kW fuel in a TC occurs at a burnup of 62 GWd/MTU.

The HSM-H calculation uses peaking factors from Reference [5.16] because the design basis source for 1.5 kW fuel in an HSM-H occurs at an average burnup of 47 ((62 + 32) x 0.5) GWd/MTU.

The neutron and gamma peaking factors are shown as a function of the active fuel region height in Table U.5-11. These factors are directly applied to each MCNP interval in the fuel region.

The average values of the axial peaking distributions are also provided in Table U.5-11. For the gamma distribution, the average value is 1.0. However, for the neutron distribution, the average value of the distribution is greater than 1.0. The average value of the axial neutron distribution may be interpreted as the ratio of the true total neutron source in an assembly to the neutron source calculated by SAS2H/ORIGEN-S for an average assembly burnup. Therefore, to properly correct the magnitude of the neutron source, the neutron source per assembly as reported in Table U.5-7, Table U.5-8 and Table U.5-9 is multiplied by the average value of the neutron source distribution as reported in Table U.5-11.

U.5.2.4 ANISN Evaluation for Bounding Source Terms

To determine which combination of burnup, wt. % initial enrichment and cooling time results in the bounding dose rates on the surface of the HSM-H and OS200 TC, the total source term, which includes the contribution from the fuel as well as the hardware in the entire assembly (including end fittings) is used to calculate its total ANISN dose rate on the HSM-H roof and OS200 TC radial model using the ANISN code.

An ANISN TC model is developed for the OS200 TC. An ANISN HSM model is also developed, for the HSM-H. The CC contribution is fixed and is included in the design basis shielding evaluation as such and therefore is not included in this ANISN evaluation.

ANISN [5.5] determines the fluence of particles throughout one-dimensional geometric systems by solving the Boltzmann transport equation using the method of discrete ordinates. Particles can be generated by either particle interaction with the transport medium or extraneous sources incident upon the system. Anisotropic cross-sections can be expressed in a Legendre expansion of arbitrary order.

The ANISN code implements the discrete ordinates method as its primary mode of operation. Balance equations are solved for the flow of particles moving in a set of discrete directions in each cell of a space mesh and in each group of a multigroup energy structure. Iterations are performed until all implicitness in the coupling of cells, directions, groups, and source regeneration is resolved.

ANISN coupled with the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.3] and the ANSI/ANS-6.1.1-1977 flux-to-dose conversion factors [5.8] is chosen to generate the ANISN dose rates used to determine the relative strength of the various source terms from fuel assemblies to determine the design basis source terms for the HSM-H and OS200 TC. These design basis source terms are used with MCNP models of the *32PTH1* system to calculate the bounding system dose rates. ANISN provides an efficient method to select the design basis source terms.

The surface dose rates are calculated using ANISN models to perform the evaluation for the fuel assembly parameters in the fuel qualification table. The ANISN model used to calculate the relative dose rates on the HSM-H surface is similar to a cut through the center of the MCNP HSM-H roof model used for the shielding evaluation. The ANISN model used to generate the relative dose rates on the TC is similar to a cut through the center of the MCNP OS200 TC side model used for the shielding evaluation. Figure U.5-1 and Figure U.5-2 provide sketches for the ANISN models of the HSM-H roof and OS200 TC centerline, respectively. An example ANISN input file is included in Section U.5.5.4.

With the exception of the fuel region, the material densities used in the ANISN models are the same as those used in the MCNP models as provided in Table U.5-12. The ANISN and MCNP number densities in the fuel region differ because in the MCNP models, the basket is modeled explicitly, while in the ANISN models the basket is homogenized with the fuel. The ANISN number densities for the fuel/basket region are provided in Table U.5-13.

To simplify the number of ANISN calculations required, a "response function" is developed using ANISN. A separate response function is developed for both the OS200 TC and HSM-H. To generate a gamma response function, a separate ANISN model is executed with a single gamma per assembly in each of the 18 CASK-81 gamma energy groups. As ANISN requires the source in particles per second per unit volume, the volume of the homogenized source region is 6.87E+06 cm^3 (r=72.1 cm and h=420.7 cm, including the top and bottom nozzle regions), resulting in a gamma source of 32/6.87E+06=4.657E-06 $\gamma/s\text{-}cm^3$. Once the dose rate resulting from a single gamma per assembly is known for each energy group, the dose rate for an arbitrary

gamma source can be determined simply by multiplying the source strength in each group by the dose rate contribution for that group and summing the results.

The neutron response function is generated in a similar fashion to the gamma response function, although only one ANISN neutron file is required because the neutron spectrum is adequately represented by the Cm-244 spectrum provided in Table U.5-14. Therefore, the ANISN model is executed with one neutron per assembly. As ANISN requires the source in particles per second per unit volume, the volume of the homogenized source region is 5.90E+06 cm^3 (r=72.1 cm and h=361.42 cm, height for the fuel region only). The resulting neutron source for ANISN is provided in Table U.5-14. The dose rate from secondary capture gammas is calculated in addition to the neutron dose rate. This method allows for the calculation of the neutron and capture gamma dose rate on the surface of the OS200 TC or HSM-H knowing only the magnitude of the neutron source.

The response functions for the OS200 TC and HSM-H are provided in Table U.5-15 and Table U.5-16, respectively. Response functions for a uniform fuel *loading* configuration are also shown in Table U.5-16. These response functions are used to compute the dose rate for each entry in the *original* fuel qualification tables. *T*he burnup/enrichment/cooling time combination that results in the highest dose rate is selected as the design basis source. 72.48 AMD 15

U.5.2.5 Reconstituted Fuel

As explained in Section U.5.2, reconstituted fuel assemblies may contain up to 10 irradiated stainless steel rods that replace damaged fuel rods. Because steel rods replace fuel rods, the decay heat of a reconstituted assembly is typically less than the decay heat of an equivalent standard assembly. Conversely, because steel contains Co-59 which activates to form Co-60, for low cooling times a reconstituted assembly typically generates higher dose rates than an equivalent standard assembly. As the half-life of Co-60 is 5.27 years, after 10 years the Co-60 activity has reduced by almost a factor of four and a reconstituted assembly no longer generates higher dose rates than an equivalent standard assembly. To bound this effect, the fuel qualification tables require that for fuel assembly with irradiated reconstituted steel rods with cooling times less than 10 years, additional one year of cooling time is required. For cooling times of 10 years or greater, no additional cooling time is required to bound the reconstituted fuel with steel rods.

To quantify this statement, additional SAS2H runs are generated for reconstituted assemblies. For each burnup and enrichment corresponding to a transition point in a fuel qualification table (i.e., the point where the cooling time experiences a change of 0.5 years), reconstituted assembly SAS2H models are developed.

The SAS2H input files for a reconstituted assembly are very similar to the input files for a standard assembly except for the following changes: (1) The number of fuel rods is reduced from 208 to 198, (2) the POWER input variable is adjusted to maintain the correct burnup for the reduced fuel loading, and (3) the light elements change to reflect that 10 fuel rods have been replaced with steel rods. The constituent masses of the reconstituted fuel assembly required for the SAS2H input is provided in Table U.5-4.

Note that a reconstituted rod cannot be irradiated for more than two cycles because the first cycle will always contain fresh, undamaged fuel. To accurately model this behavior, two SAS2H models are generated for each transition point. The first SAS2H model is for only one cycle of irradiation of 10 reconstituted rods, while the second SAS2H model is for three cycles of irradiation of 10 reconstituted rods. By subtracting the single cycle source term of the reconstituted rods from the total source term (fuel and reconstituted rods) for three cycles, the source term for three cycle irradiation of fuel and two cycle irradiation of reconstituted rods is generated.

This source term is inserted into the HSM-H and OS200 TC response functions to determine the dose rates for comparison to the design basis source dose rates. If the reconstituted fuel dose rate for either the HSM-H or OS200 TC exceeded the dose rate with design basis fuel, an additional 0.5 year of cooling time is added to the reconstituted fuel source term. When the reconstituted fuel is examined in this fashion, no more than one additional year of cooling time is required for reconstituted fuel to be bounded by the design basis source if the decay time listed in the fuel qualification table is less than 10 years. After a cooling time greater than 10 years the effects of reconstituted fuel become insignificant. *Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.* AMD 15 & 72.48

A sensitivity study performed using source terms consistent with the unified FQTs (Technical Specifications Tables 1-3a through 1-3p) indicates that the additional cooling time requirements for reconstituted fuel remain valid for the unified FQTs.

U.5.2.6 SCALE6.0/ORIGEN-ARP Source Terms

Because the original FQTs have been replaced with unified FQTs (Technical Specifications Tables 1-3a through 1-3p), the design basis source terms developed in Section U.5.2 are obsolete because they are based upon burnup, enrichment, and cooling time combinations that are no longer applicable. Therefore, SCALE6.0/ORIGEN-ARP design basis source terms are developed based on the unified FQTs. The FQTs are documented in Section M.5.2.6.

The methodology used to develop the SCALE6.0/ORIGEN-ARP design basis source terms is the same as described in Section U.5.2. The ANISN transfer cask and HSM response functions developed in Section U.5.2.4 are used to evaluate the source terms for each FQT burnup, enrichment, and cooling time (BECT) combination. The BECT combination that results in the maximum dose rate is selected as the design basis source. AMD 15

A heat load zone configuration is developed that bounds HLZC 1 through 6 (the HLZCs are defined in Figures U.2-1 through U.2-3 and Figures U.2-5 through U.2-7). The decay heat in the central 4 compartments is modeled at 0.8 kW/FA and the decay heat in zone 2 is modeled at 1.3 kW/FA, for a total decay heat of 42.8 kW. The modeled HLZC is referred to as "Modified HLZC 1" because it is the same as HLZC 1 except the inner zone is conservatively increased from 0.6 kW/FA to 0.8 kW/FA. The DSC decay heat bounds the maximum DSC decay heat of 40.8 kW and sufficiently bounds HLZC 1 through 6. This arrangement is also summarized on Figure U.5-3 in the lower table. Based on the ANISN response function analysis, the SCALE6.0/ORIGEN-ARP design basis source terms for Modified HLZC 1 are:

HSM

- *Table U.5-24: 0.8 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 9.6 years cooled*
- *Table U.5-25: 1.3 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 4.6 years cooled*
- *Table U.5-26: 1.5 kW/FA, 19 GWd/MTU, 0.8 wt.% U-235, 2.0 years cooled*

Transfer cask

- *Table U.5-24: 0.8 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 9.6 years cooled*
- *Table U.5-25: 1.3 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 4.6 years cooled*
- *Table U.5-27: 1.5 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 3.9 years cooled*

Note that the 0.8 kW/FA and 1.3 kW/FA sources are the same for both the HSM and transfer cask, although the 1.5 kW/FA source terms are different.

Because the FQTs in Section M.5.2.6 are developed for uranium loadings of 380 kgU, 475 kgU, and 492 kgU for fixed heat loads, it is observed that 380 kgU source terms bound 475 or 492 kgU source terms because self-shielding of the sources by the uranium in the fuel matrix is reduced. Therefore, SCALE 6.0/ORIGEN-ARP design basis source terms are developed only for 380 kgU.

The MCNP5 neutron models are run with the NONU card to suppress subcritical neutron multiplication. Subcritical neutron multiplication is addressed by multiplying the neutron source computed by ORIGEN-ARP by $1/(1 - k_{eff})$. Values of k_{eff} appropriate for the burnups of the sources are provided in the source term tables (Table U.5-24 through U.5-27).

The gamma source in the active fuel region is modeled with the axial burnup profile appropriate for the burnup of the source and is obtained from Table 20 of ORNL/TM-12973 [5.16]. These profiles are summarized in Appendix M.5, Table M.5-53. The neutron profile is derived as the 4^{th} power of the gamma profile and is also summarized in Table M.5-53. The burnup peaking factor accounts for the increase in the neutron source magnitude due to the axial burnup profile. The burnup peaking factors used in the neutron calculations are provided in the source term tables (Table U.5-24 through U.5-27).

The CC source terms provided in Table U.5-10 are applicable and may be added to the fuel-only source terms provided in Table U.5-24 through U.5-27.

U.5.4 Shielding Evaluation

Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from 32 *0.490 MTU* fuel assemblies are calculated with the MCNP Code [5.2] at various locations on and around the NUHOMS® 32PTH1 DSCs within the HSM and OS200 TC.

The following shielding evaluation discussion specifically addresses the NUHOMS® 32PTH1 DSC in an OS200 TC and the NUHOMS® 32PTH1 DSC in HSM-H using the *0.490 MTU* design-basis source terms described in the above sections. *Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from 32 0.380 MTU fuel assemblies with CCs are also calculated with the MCNP5 Code [5.2] at various locations on and around the NUHOMS® 32PTH1 DSCs within the HSM and TC.*

The shielding evaluation that determines the effect of loading 0.380 MTU per assembly on the dose rates is described in Section U.5.4.12.

U.5.4.1 Computer Program

MCNP [5.2] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Pointwise (continuous energy) cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation are accounted for in the cross section set. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. Important standard features that make MCNP very versatile and easy to use include a powerful general source; an extensive collection of cross-section data; and an extensive collection of variance reduction techniques that can be employed to track particles through very complex deep penetration problems. MCNP was employed to take advantage of its mesh tallies capabilities in calculating dose rates distributed over the surface of the HSM. It also allows more point detectors to be used in a single run that substantially reduces the number of input/out decks needed to perform ISFSI site dose rate calculations described in Chapter U.10.

U.5.4.2 Spatial Source Distribution

The source components are:

- The neutron sources due to the active fuel region,
- The gamma source due to the active fuel region,
- The gamma source due to the plenum,
- The gamma source due to the top region,
- The gamma source due to the bottom region,
- The gamma source due to the CC in the active fuel region,
- The gamma source due to the CC in the plenum region, and
- The gamma source due to the CC in the top region.

U.5.4.9 OS200 TC Models During Fuel Loading Operations

MCNP models are developed for the cask decontamination and welding operations during fuel loading using the 32PTH1 DSC.

Cask Decontamination: The 32PTH1 DSC and the OS200 TC are assumed to be completely filled with water up to the bottom surface of the DSC shield plug and including the region between 32PTH1-DSC and cask, which is referred to as the "cask/32PTH1-DSC annulus." The 32PTH1-DSC top shield plug and inner top cover plate are assumed to be in place and the temporary shielding has not yet been installed. These models are utilized axially to determine the dose rates prior to the installation of the welding system. Results for this case are provided in Table U.5-3.

Welding and 32PTH1-DSC Draining: Before the start of welding operation, approximately 60% of the water in the DSC cavity is removed due to hydrogen generation. A dry DSC cavity is assumed in all welding models to be conservative. Temporary shielding consisting of three inches of NS-3 and one inch of steel is assumed to cover the 32PTH1-DSC inner top cover plate. In addition, the DSC outer top cover plate is not present. The cask/32PTH1-DSC annulus is assumed to remain completely filled with water. Results for this case are provided in Table U.5-3.

U.5.4.10 Impact on Dose Rates due to Reduced Density Concrete and Gaps between HSMs

The purpose of this section is to report the impact on surface maximum and average dose rates for a 2x1 side-by-side array of HSM-Hs containing 32PTH1 DSCs due to lowering the concrete density from 145 pounds per cubic foot (pcf) to 140 pcf and adding a 1.5-inch gap between adjacent HSMs. Table U.5-*18* and Table U.5-*19* provide ratios of the maximum and surface-average dose rate results, respectively, with the 1.5-inch gaps and lower density concrete to the maximum and surface-average dose rates with no gaps and a concrete density of 145 pcf.

The results in Table U.5-*18* show that the lower density concrete and 1.5-inch gaps cause an overall increase of approximately 40% in the maximum dose rates on the HSM roof, an increase of 10% in the maximum dose rate at the HSM-H front bird screen and an increase of 14% in the maximum dose rate at the HSM-H door centerline. A larger ratio of 4.64 is observed between the maximum dose rates for the HSM end (side) shield wall surface. This increase in dose rate is localized and occurs outside the gap between the HSM and rear shield wall at approximately 650 cm from the front and is due to radiation streaming through the gap.

The results in Table U.5-*19* show that the lower density concrete and 1.5-inch gaps cause approximately a factor of 2 increase on the surface-average dose rate for the HSM-H roof and for the HSM-H end (side) shield wall surface, a 23% increase on the surface-average dose rate for the HSM-H front and a ratio of 4.5 increase on the HSM back shield wall.

The ratios shown in Table U.5-*18* and Table U.5-*19* can be used as scaling factors to increase the maximum and surface-average dose rates of an HSM-H array to account for low density concrete and 1.5-inch gaps during HSM fabrication and installation.

The shielding analysis results with the NUHOMS®-32PTH1 DSC are applicable to the NUHOMS®-32PTH1 Type 2-W DSC.

U.5.4.11 <u>*Not Used*</u>

U.5.4.12 <u>*Shielding Analysis with a Loading of 0.380 MTU per Fuel Assembly*</u>

As discussed in Section U.5.4, additional shielding analysis is performed with reduced FA Uranium loading of 0.380 MTU/FA. The objective of this analysis is to determine the impact that reducing the FA Uranium loading has on system dose rates, site dose, and occupational exposure. The results of this analysis are employed to scale the dose rate, site dose, and occupational exposure results for the 32PTH1 System. For this purpose, the MCNP5 models employed for the 0.490 MTU/FA analyses are rerun with updated source terms as described in Section U.5.2.6, and with updated material specifications to reflect the reduction in MTU/FA. The material specification update for the 0.380 MTU/FA analysis uses the same active fuel stack height as that of the 0.490 MTU/FA models, and the material density is reduced accordingly to achieve the 0.380 MTU/FA total mass. This approach is very conservative in that actual 0.380 MTU/FA assemblies use full pellet density UO^2 pellets which are stacked to a lower active fuel stack height. By reducing the density instead of reducing the stack height, self-shielding is significantly reduced below that of actual 0.380 MTU/FA assemblies, resulting in conservatively high dose rate estimates.

MCNP5 calculations are performed for the 32PTH1 DSC inside the HSM-H in the normal storage configuration. MCNP5 calculations are also performed for the 32PTH1 DSC inside the OS200 TC in the decontamination and welding configurations, and in the normal and accident transfer configurations. The resulting dose rates, site dose, and occupational exposure are compared to the 0.490 MTU/FA dose rates, site dose, and occupational exposure to determine scaling factors for each of these configurations.

<u>*Storage System Dose Rates*</u>

For the 32PTH1/HSM-H storage configuration, a comparison is shown in Table U.5-28 of the average 0.490 MTU/FA dose rates reported in Table U.5-1 and the average 0.380 MTU/FA dose rates. These results are grouped into two categories:

1. *HSM roof and front dose rates, which are labeled as "R+F,"*

2. *HSM back and side shield wall dose rates, which are labeled as "SSW."*

A scaling factor is calculated for each of the 4 average dose rates shown in the table labeled as "Derived Scaling Factors" and the maximum value for each category (R+F and SSW) is chosen and labeled as "Maximum Derived Scaling Factors". The maximum derived R+F scaling factor is 1.18, or an 18% increase, and the maximum derived SSW scaling factor is 1.36, or a 36% increase.

A comparison is shown in Table U.5-29 of the maximum 0.490 MTU/FA dose rates reported in Table U.5-1 and the maximum 0.380 MTU/FA dose rates. The maximum dose rates are categorized identically as the average dose rates, with the Roof Centerline, the Roof Bird Screen, the Door Exterior Surface, and the Front Bird Screen in the R+F category, and the End (Side) Shield Wall Surface in the SSW category. The Maximum Derived Scaling Factors from Table U.5-28 are applied to the 0.490 MTU/FA maximum dose rate values of each category in Table U.5-29 to verify the validity of using the scaling factor derived from average dose rates to predict maximum dose rates. In Table U.5-29, the "Scaled 0.380 MTU/FA Results" column is the maximum 0.380 MTU/FA dose rate data produced in this manner. Comparing the scaled 0.380 MTU/FA results with the calculated 0.380 MTU/FA results demonstrates that the R+F scaling factor always generates a bounding dose rate, whereas the SSW scaling factor under predicts the dose rate by ~6%. The SSW scaling factor result is judged to be sufficiently close to the actual value so as to be acceptable for use.

Storage System Site Dose

As stated above, the 32PTH1/HSM front average dose rate increases by ≈18% when loading 0.380 MTU/FA FA's (the R+F scaling factor is 1.18). Since the site dose value is dominated by the HSM front average dose rate, the site dose value will also increase by ≈18% when loading 0.380 MTU/FA FA's.

Storage System Dose Rates and Site Dose Scaling Factors

Based on these results, three scaling factors are reported below. They are implemented to scale the existing 0.490 MTU/FA storage dose rate and site dose values to determine bounding values for FA loadings containing from 0.380 MTU/FA up to but not including 0.490 MTU/FA:

1. *The dose rates for the HSM front and roof are to be scaled by 1.18.*

2. *The dose rates for the HSM side and rear are to be scaled by 1.36.*

3. *The site dose is to be scaled by 1.18.*

These scaling factors are included as footnotes in the dose rate results summarized in Table U.5-1. These scaling factors are also employed to scale the generic site dose (2X10 back-to-back and front-to-front arrays) results calculated for the 32PTH1 system in Appendix U.10, and to scale the dose rate consequences of accidents for the 32PTH1 system in Appendix U.11.

Transfer System Dose Rates[1]

For the 32PTH1/OS200 TC transfer system, maximum dose rates for the three transfer configurations at both 0.490 MTU/FA and 0.380 MTU/FA are compared:

- *Table U.5-30 shows a comparison of the maximum 0.490 MTU/FA dose rates originally reported in Table U.5-2 and the maximum 0.380 MTU/FA dose rates for the normal transfer configuration.*

[1] *For the discussion in this section, the fully loaded 32PTH1 DSC placed into the OS200 TC is referred to as the "Transfer System."*

- *Table U.5-31 shows a comparison of the maximum 0.490 MTU/FA dose rates originally reported in Table U.5-3 and the maximum 0.380 MTU/FA dose rates for the decontamination configuration.*
- *Table U.5-32 shows a comparison of the maximum 0.490 MTU/FA dose rates originally reported in Table U.5-3 and the maximum 0.380 MTU/FA dose rates for the welding configuration.*

The results in each of these tables are grouped into three categories:

1. *OS200 TC side dose rates, which are labeled as "Side,"*
2. *OS200 TC top dose rates, which are labeled as "Top,"*
3. *OS200 TC bottom dose rates, which are labeled as "Bottom."*

A scaling factor is calculated for each dose rate group (top, side, and bottom) for each configuration (normal transfer, decontamination, and welding). These scaling factors are labeled as "Derived Scaling Factors." The maximum value for each category in each configuration is chosen and labeled as "Maximum Derived Scaling Factors." The maximum derived scaling factors are as follows:

- *For the normal transfer configuration,*
  - *The side scaling factor is 1.14, or a 14% increase,*
  - *The top scaling factor is 1.60, or a 60% increase,*
  - *The bottom scaling factor is 1.51, or a 51% increase.*
- *For the decontamination configuration,*
  - *The side scaling factor is 1.10, or a 10% increase,*
  - *The top scaling factor is 1.59, or a 59% increase,*
  - *The bottom scaling factor is 1.91, or a 91% increase.*
- *For the welding configuration,*
  - *The side scaling factor is 1.24, or a 24% increase,*
  - *The top scaling factor is 1.63, or a 63% increase,*
  - *The bottom scaling factor is 1.54, or a 54% increase.*

The highest bounding scaling factor for each cask position from the three configurations listed above is conservatively chosen to scale all dose rates from that cask position in all configurations. The bounding TC side scaling factor is the welding configuration factor of 1.24, the bounding TC top scaling factor is the welding configuration factor of 1.63, and the bounding TC bottom scaling factor is the decontamination configuration factor of 1.91.

Transfer System Dose Rate Scaling Factors

Based on these results, three scaling factors are reported below. They are implemented to scale the existing 0.490 MTU/FA normal transfer, welding and decontamination dose rates to determine bounding values for FA loadings containing from 0.380 MTU/FA up to but not including 0.490 MTU/FA:

- *The dose rates for the TC side are to be scaled by 1.24,*
- *The dose rates for the TC top are to be scaled by 1.63,*
- *The dose rates for the TC bottom are to be scaled by 1.91.*

These scaling factors are included as footnotes in the dose rate results summarized in Tables U.5-2, Table U.5-3, Table U.5-21, and Table U.5-22.

The 0.380 MTU/FA OS200 TC accident transfer configuration dose rates are bound by the 0.490 MTU/FA analysis results, and do not require updating.

Transfer System Occupational Exposure

The 0.380 MTU/FA normal transfer, decontamination, and welding dose rates from Tables U.5-30, U.5-31, and U.5-32 respectively are substituted into the area dose rate fields of the chapter U.10 occupational exposure summary table, Table U.10-1, and the total occupational exposure from loading 0.380 MTU/FA fuel is calculated. A comparison is shown in Table U.5-33 of the 0.490 MTU/FA area dose rates, total exposure by operational step, and cumulative operational exposure originally reported in Table U.10-1 and the 0.380 MTU/FA area dose rates, total exposure by operational step, and cumulative operational exposure. At the bottom of Table U.5-33, the 0.490 MTU/FA cumulative operational exposure is 1934 person-millirem and the 0.380 MTU/FA cumulative operational exposure is 2425 person-millirem. Loading 0.380 MTU/FA fuel results in an increase in occupational exposure of 25% (which is a scaling factor of 1.25) when compared to loading 0.490 MTU/FA fuel.

Transfer System Occupational Exposure Scaling Factors

Based on these results, the scaling factor of 1.25 is implemented to scale the existing 0.490 MTU/FA transfer system pool to pad loading operation occupational exposure to determine the bounding value for FA loadings containing from 0.380 MTU/FA up to but not including 0.490 MTU/FA. This scaling factor is employed to scale the occupational exposure results calculated for the 32PTH1 system in Appendix U.10.

U.5.6 References

5.1 Oak Ridge National Laboratory, RSIC Computer Code Collection, "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluations for Workstations and Personal Computers," NUREG/CR-0200, Revision 6, ORNL/NUREG/CSD-2/V2/R6.

5.2 "Monte Carlo N-Particle Transport Code System," CCC-701, Oak Ridge National Laboratory, RSICC Computer Code Collection, August 2001.

5.3 CASK-81 - 22 Neutron, 18 Gamma-Ray Group, P3, Cross Sections for Shipping Cask Analysis," DLC-23, Oak Ridge National Laboratory, RSIC Data Library Collection, August 1987.

5.4 Ludwig, S.B., and J.P. Renier, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018 Oak Ridge National Laboratory, December 1989.

5.5 "ANISN-ORNL - One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering", CCC-254, Oak Ridge National Laboratory, RSIC Computer Code Collection, April 1991.

5.6 "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," NUREG/CR-6801, Oak Ridge National Laboratory.

5.7 Jenal, J. P., P. J. Erickson, W. A. Rhoades, D. B. Simpson, and M. L. Williams, "The Generation of a Computer Library for Discrete Ordinates Quadrature Sets," ORNL/TM-6023, Oak Ridge National Laboratory, October 1977.

5.8 "American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors," ANSI/ANS-6.1.1-1977, American Nuclear Society, LaGrange Park, Illinois, March 1977.

5.9 K. Ueki, N. Nariyama, A. Ohashi, A. Yamaji. "Measurement of Dose-Equivalent Rates around a Cask and Monte Carlo Analysis with Actual Configuration of Fuel Basket". Journal of Nuclear Science and Technology, (Supplement 1, p.324-328), Atomic Energy Society of Japan, March 2000, ISSN 0022-3131.

5.10 *Not Used.*

5.11 MD DeHart and OW Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, September 1996.

5.12 OW Hermann, SM Bowman, MC Brady, CV Parks, "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, March 1995.

5.13 Japan Atomic Energy Research Institute, "Technical Development on Burn-up Credit for Spent LWR Fuels," JAERI-Tech 2000-071, September 21, 2000.

5.14 U.S. Nuclear Regulatory Commission, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High Burnup LWR Fuel," NUREG/CR-6700, Published January 2001, ORNL/TM-2000/284.

5.15 "Characteristics of Potential Repository Waste," DOE/RW-0184-R21, Volume 1, Oak Ridge National Laboratory, Tennessee, July 1992.

5.16 M. D. DeHart, "Sensitivity and Parametric Evaluations of Significant Aspects of Burn-up Credit for PWR Spent Fuel Packages", ORNL/TM-12973, May 1996.

5.17 U.S. Nuclear Regulatory Commission, "Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors," NUREG/CR-6968, Published February 2010, ORNL_TM-2008-71.

5.18 U.S. Nuclear Regulatory Commission, "Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel," NUREG/CR-7012, Published January 2011, ORNL-TM-2010-41.

5.19 U.S. Nuclear Regulatory Commission, "Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation -- Vandellós II Reactor," NUREG/CR-7013, Published January 2011, ORNL-TM-2009-321.

5.20 ORNL/TM-2005/39, Version 6, SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Oak Ridge National Laboratory, January 2009.

Table U.5-1
Summary of Bounding Maximum and Average Dose Rates with NUHOMS®-32PTH1 Bounding DSC in HSM-H (2)

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1,3) (mrem/hr)	Total MCNP 1σ Error
HSM Roof (centerline)(4)	14.13	0.04	0.74	0.01	14.86	0.04
HSM Roof Bird screen(4)	114.71	0.01	6.47	0.01	121.17	0.01
HSM End (Side) Shield Wall Surface(5)	1.49	0.05	0.06	0.01	1.54	0.05
HSM Door Exterior Surface (centerline)(4)	0.61	0.07	0.08	0.06	0.70	0.06
HSM Front Bird screen(4)	471.28	0.04	5.89	0.02	477.17	0.04

Dose Rate Location	Gamma Average (mrem/hr)	Gamma MCNP 1σ Error	Average Neutron (mrem/hr)	Neutron MCNP 1σ Error	Average Total(3) (mrem/hr)	Total MCNP 1σ Error
HSM Roof(4)	11.27	0.01	0.62	<0.01	11.89	0.01
HSM End (Side) Shield Wall Surface(5)	0.36	0.01	0.02	<0.01	0.38	0.01
HSM Front(4)	14.87	0.06	0.31	0.02	15.19	0.06
HSM Back Shield Wall(5)	0.07	0.01	0.004	0.01	0.07	0.01

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the maximum of total dose rate is not always the sum of the gamma plus neutron dose rate maximums.

(2) Dose is calculated using bounding 32PTH1 DSC, from the shielding performance stand point. This DSC contains the design basis assembly source loaded in accordance with bounding HLZC depicted in Figure U.5-3. Also, it is assumed that design basis CC sources are present in each fuel assembly. Dose rates can be higher by 6% to account for the use of grout during HSM fabrication and installation.

(3) Use the ratios shown in *Table U.5-18* and *Table U.5-19* to increase the maximum and surface-average dose rates respectively to account for reduced density concrete and gaps of up to 1.5" as described in Section U.5.4.10.

(4) These dose rates increase by 18% when loading 0.380 MTU FAs.

(5) These dose rates increase by 36% when loading 0.380 MTU FAs.

Table U.5-2
Summary of NUHOMS® 32PTH1 DSC, OS200 TC Maximum Dose Rates During Transfer Operations

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total[(1)] (mrem/hr)	Total MCNP 1σ Error
Cask Side Surface (Radial) [(3)]	4.07E+02	0.0048	2.02E+02	0.0085	6.09E+02	0.0043
Cask Top Axial Surface[(4)]	2.32E+02	0.0602	3.86E+01	0.0452	2.51E+02	0.0556
Cask Bottom Axial Surface[(5)]	2.15E+03[(2)]	0.0181	1.40E+03[(2)]	0.0136	3.55E+03[(2)]	0.0122
50 cm from Cask Side (Radial)[(3)]	2.39E+02	0.0047	1.23E+02	0.0082	3.62E+02	0.0042
50 cm from Cask Top Axial Surface[(4)]	4.71E+01	0.0793	2.07E+01	0.0453	5.95E+01	0.0653
50 cm from Cask Bottom Axial Surface[(5)]	9.25E+02	0.0190	3.43E+02	0.0194	1.27E+03	0.0148
1m from Cask Side (Radial) [(3)]	1.61E+02	0.0046	8.39E+01	0.0082	2.45E+02	0.0041
1m from Cask Top Axial Surface[(4)]	2.95E+01	0.0488	1.44E+01	0.0692	3.79E+01	0.0393
1m from Cask Bottom Axial Surface[(5)]	4.65E+02	0.0200	1.40E+02	0.0299	6.05E+02	0.0169
Cask 1 m (Radial) Accident Condition	1.74E+02	0.0280	3.58E+03	0.003	3.76E+03	0.0031
Cask 100 m (Radial) Accident Condition	9.38E-02	0.0175	1.00E+00	0.0029	1.10E+00	0.0030
Cask 500 m (Radial) Accident Condition	5.25E-04	0.0188	3.40E-03	0.0043	3.92E-03	0.0045

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 113 mrem/hr gamma, 71 mrem/hr neutron for a total average dose rate of 184 mrem/hr.

(3) The Side dose rates increase by 24% when loading 0.380 MTU FAs.

(4) The Top dose rates increase by 63% when loading 0.380 MTU FAs.

(5) The Bottom dose rates increase by 91% when loading 0.380 MTU FAs.

Table U.5-3
Summary of NUHOMS® 32PTH1 DSC, OS200 TC Maximum Dose Rates During Decontamination and Welding Operations

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1) (mrem/hr)	Total MCNP 1σ Error
Decontamination						
Cask Side Surface (Radial)(4)	3.46E+02	0.0035	3.73E+02	0.0040	7.19E+02	0.0027
Top Axial Surface(5)	8.32E+02	0.0226	9.34E+00	0.0308	8.33E+02	0.0226
Cask Bottom Axial Surface(6)	1.70E+03(2)	0.0153	6.83E+01(2)	0.0424	1.77E+03(2)	0.0148
50 cm from Cask Side (Radial) (4)	2.02E+02	0.0035	2.29E+02	0.0037	4.31E+02	0.0025
50 cm from Top Axial Surface(5)	6.15E+02	0.0298	5.64E+00	0.0443	6.16E+02	0.0298
50 cm from Cask Bottom Axial Surface(6)	7.31E+02	0.0160	1.82E+01	0.0534	7.49E+02	0.0157
1m from Cask Side (Radial)(4)	1.35E+02	0.0034	1.58E+02	0.0037	2.92E+02	0.0025
1m from Top Axial Surface(5)	4.27E+02	0.0345	3.61E+00	0.0537	4.28E+02	0.0344
1m from Cask Bottom Axial Surface(6)	3.68E+02	0.0174	8.06E+00	0.0808	3.76E+02	0.0171
Welding						
Cask Side Surface (Radial)(4)	3.09E+02	0.0053	1.47E+02	0.0076	4.56E+02	0.0043
Top Axial Surface(5)	7.20E+02	0.1107	4.21E+01	0.0618	7.62E+02	0.1047
Cask Bottom Axial Surface(6)	2.17E+03(3)	0.0164	1.22E+03(3)	0.0110	3.39E+03(3)	0.0112
50 cm from Cask Side (Radial) (4)	1.85E+02	0.0051	8.99E+01	0.0072	2.75E+02	0.0042
50 cm from Top Axial Surface(5)	4.25E+02	0.0214	1.74E+01	0.0592	4.42E+02	0.0207
50 cm from Cask Bottom Axial Surface(6)	9.27E+02	0.0171	3.00E+02	0.0157	1.23E+03	0.0135
1 cm from Cask Side (Radial) (4)	1.27E+02	0.0050	6.23E+01	0.0077	1.89E+02	0.0042
1 cm from Top Axial Surface(5)	2.94E+02	0.0231	1.17E+01	0.0788	3.06E+02	0.0224
1 cm from Cask Bottom Axial Surface(6)	4.66E+02	0.0182	1.21E+02	0.0244	5.87E+02	0.0153

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 86 mrem/hr gamma, 11 mrem/hr neutron for a total average dose rate of 97 mrem/hr.

(3) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 108 mrem/hr gamma, 57 mrem/hr neutron for a total average dose rate of 165 mrem/hr. Note that this bottom axial dose rate has no impact on the occupational exposure because no operations are performed near bottom axial location.

(4) The Side dose rates increase by 24% when loading 0.380 MTU FAs.

(5) The Top dose rates increase by 63% when loading 0.380 MTU FAs.

(6) The Bottom dose rates increase by 91% when loading 0.380 MTU FAs.

Table U.5-18
Ratios of 2x1 HSM-H Array Maximum Dose Rates
(2x1 Array Dose Rates with 1.5-inch Gaps and 140 pcf Concrete / No Gaps Dose Rates and 145 pcf Concrete)

Location	*Maximum Gamma Ratio*	*Maximum Neutron Ratio*	*Maximum Total Ratio*
HSM Roof (centerline)	*1.45*	*1.35*	*1.45*
HSM Roof Bird Screen	*1.42*	*1.27*	*1.41*
HSM End (Side) Shield Wall Surface	*4.79*	*1.86*	*4.64*
HSM Door Exterior Surface (centerline)	*1.12*	*1.28*	*1.14*
HSM Front Bird Screen	*1.10*	*1.12*	*1.10*

Table U.5-19
Ratios of 2x1 HSM-H Array Surface Average Dose Rates
(2x1 Array Dose Rates with 1.5-inch Gaps and 140 pcf Concrete / No Gaps Dose Rates and 145 pcf Concrete)

Location	*Maximum Gamma Ratio*	*Maximum Neutron Ratio*	*Maximum Total Ratio*
HSM Roof	*1.99*	*1.51*	*1.96*
HSM End (Side) Shield Wall Surface	*2.39*	*1.91*	*2.36*
HSM Front	*1.23*	*1.21*	*1.23*
HSM Back Shield Wall	*4.68*	*2.42*	*4.52*

Table U.5-20
Deleted

Table U.5-21
Summary of NUHOMS® 32PTH1 Type 2-W DSC OS200 TC Maximum Dose Rates During Normal Transfer Operations

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total (mrem/hr)	Total MCNP 1σ Error
Cask Side Surface (Radial)(2)	2.27E+02	0.0025	1.38E+02	0.0061	3.66E+02	0.0028
Cask Top Axial Surface(3)	8.12E+01	0.0381	2.90E+01	0.0287	9.69E+01	0.0322
Cask Bottom Axial Surface(4)	7.54E+02	0.0140	1.02E+03	0.0095	1.77E+03	0.0081
1.5 ft from Cask Side (Radial)(2)	1.29E+02	0.0024	8.38E+01	0.0057	2.13E+02	0.0027
1.5 ft from Cask Top Axial Surface(3)	1.73E+01	0.0477	1.49E+01	0.0326	2.63E+01	0.0366
1.5 ft from Cask Bottom Axial Surface(4)	3.06E+02	0.0153	2.51E+02	0.0140	5.57E+02	0.0105
3 ft from Cask Side (Radial)(2)	8.45E+01	0.0024	5.81E+01	0.0058	1.43E+02	0.0027
3 ft from Cask Top Axial Surface(3)	1.10E+01	0.0316	9.76E+00	0.0448	1.76E+01	0.0231
3 ft from Cask Bottom Axial Surface(4)	1.55E+02	0.0177	1.03E+02	0.0209	2.58E+02	0.0135

Notes:

1. *Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.*
2. *The Side dose rates increase by 24% when loading 0.380 MTU FAs.*
3. *The Top dose rates increase by 63% when loading 0.380 MTU FAs.*
4. *The Bottom dose rates increase by 91% when loading 0.380 MTU FAs.*

Table U.5-22
Summary of NUHOMS® 32PTH1 Type 2-W DSC OS200 TC Maximum Dose Rates During Normal Decontamination Operations

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total (mrem/hr)	Total MCNP 1σ Error
Cask Side Surface (Radial)(2)	8.99E+01	3.3E-03	1.34E+02	3.0E-03	2.24E+02	2.2E-03
Top Axial Surface(3)	3.62E+02	2.42E-02	3.37E+00	2.49E-02	3.62E+02	2.42E-02
Cask Bottom Axial Surface(4)	5.74E+02	1.49E-02	3.74E+01	2.70E-02	6.11E+02	1.41E-02
1.5 ft from Cask Side (Radial)(2)	5.26E+01	3.26E-03	8.21E+01	2.70E-03	1.35E+02	2.08E-03
1.5 ft from Top Axial Surface(3)	2.48E+02	2.45E-02	1.93E+00	3.29E-02	2.48E+02	2.45E-02
1.5 ft from Cask Bottom Axial Surface(4)	2.31E+02	1.64E-02	1.03E+01	3.44E-02	2.41E+02	1.58E-02
3 ft from Cask Side (Radial)(2)	3.51E+01	3.26E-03	5.68E+01	2.70E-03	9.18E+01	2.08E-03
3 ft from Top Axial Surface(3)	1.69E+02	2.80E-02	1.19E+00	4.08E-02	1.69E+02	2.80E-02
3 ft from Cask Bottom Axial Surface(4)	1.18E+02	1.91E-02	4.87E+00	4.57E-02	1.22E+02	1.84E-02

Notes:

1. *Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.*
2. *The Side dose rates increase by 24% when loading 0.380 MTU FAs.*
3. *The Top dose rates increase by 63% when loading 0.380 MTU FAs.*
4. *The Bottom dose rates increase by 91% when loading 0.380 MTU FAs.*

Table U.5-23
Summary of Experimental Samples as a Function of Burnup Range

		Low Burnup (B < 45 GWd/MTU)		High Burnup (B > 45 GWd/MTU)	
Power Plant	**Reference**	**No of Samples**	**Range (GWd/MTU)**	**No of Samples**	**Range (GWd/MTU)**
Takahama-3	NUREG/CR-6968 Reference [5.17]	14	8.55 – 42.16	2	47.03 – 47.25
Three Mile Island - 1		10	22.80 – 44.8	9	50.10 – 55.70
Calvert Cliffs		3	27.35 – 44.34	-	-
Vandellós II	NUREG/CR-7013 Reference [5.19]	1	42.50	5	54.85 – 78.30
Gösgen	NUREG/CR-7012 Reference [5.18]	1	31.10	5	46.00 –70.30
GKN II		-	-	1	54.10
Total		29	-	22	-

Table U.5-24
Modified HLZC#1 0.8 kW Design Basis HSM and TC Source Term

Bounding Radiological Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 9.6 years of cooling						
E_{min}, MeV	***to***	***E_{max}, MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	*to*	*5.00e-02*	*1.975E+11*	*6.392E+14*	*3.835E+11*	*1.112E+11*
5.00e-02	*to*	*1.00e-01*	*3.130E+10*	*1.624E+14*	*6.856E+10*	*2.156E+10*
1.00e-01	*to*	*2.00e-01*	*1.163E+10*	*1.238E+14*	*1.919E+10*	*5.218E+09*
2.00e-01	*to*	*3.00e-01*	*6.657E+08*	*3.467E+13*	*1.015E+09*	*2.596E+08*
3.00e-01	*to*	*4.00e-01*	*1.455E+09*	*2.105E+13*	*1.706E+09*	*3.392E+08*
4.00e-01	*to*	*6.00e-01*	*2.180E+10*	*1.627E+14*	*1.424E+10*	*2.325E+07*
6.00e-01	*to*	*8.00e-01*	*1.293E+10*	*1.460E+15*	*1.533E+10*	*1.328E+09*
8.00e-01	*to*	*1.00e+00*	*4.280E+09*	*8.674E+13*	*8.930E+09*	*2.903E+09*
1.00e+00	*to*	*1.33e+00*	*9.044E+12*	*1.419E+14*	*1.994E+13*	*6.282E+12*
1.33e+00	*to*	*1.66e+00*	*2.554E+12*	*3.474E+13*	*5.631E+12*	*1.774E+12*
1.66e+00	*to*	*2.00e+00*	*9.587E+01*	*9.910E+10*	*6.248E+01*	*1.182E-02*
2.00e+00	*to*	*2.50e+00*	*6.111E+07*	*5.213E+10*	*1.347E+08*	*4.245E+07*
2.50e+00	*to*	*3.00e+00*	*5.221E+04*	*4.249E+09*	*1.151E+05*	*3.627E+04*
3.00e+00	*to*	*4.00e+00*	*1.009E-05*	*4.816E+08*	*5.163E-05*	*8.309E-06*
4.00e+00	*to*	*5.00e+00*	*0.0*	*3.459E+07*	*0.0*	*0.0*
5.00e+00	*to*	*6.50e+00*	*0.0*	*1.388E+07*	*0.0*	*0.0*
6.50e+00	*to*	*8.00e+00*	*0.0*	*2.723E+06*	*0.0*	*0.0*
8.00e+00	*to*	*1.00e+01*	*0.0*	*5.782E+05*	*0.0*	*0.0*
*Total Gamma, g/(sec*FA)*			*1.188E+13*	*2.867E+15*	*2.608E+13*	*8.199E+12*
*(1)Total Neutrons, n/(sec*FA)*			*9.996E+8*			

(1) This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where the dry k_{eff}= 0.25189 and bpf=1.152.

Table U.5-25
Modified HLZC#1 1.3 kW Design Basis HSM and TC Source Term

Bounding Radiological Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 4.6 years of cooling						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle (g/s)	In-core (g/s)	Plenum (g/s)	Top Nozzle (g/s)
0.00e+00	to	5.00e-02	4.723E+11	1.250E+15	7.888E+11	2.121E+11
5.00e-02	to	1.00e-01	6.083E+10	3.513E+14	1.324E+11	4.159E+10
1.00e-01	to	2.00e-01	2.914E+10	2.984E+14	4.132E+10	1.009E+10
2.00e-01	to	3.00e-01	1.757E+09	8.428E+13	2.260E+09	4.995E+08
3.00e-01	to	4.00e-01	4.391E+09	5.889E+13	4.319E+09	6.530E+08
4.00e-01	to	6.00e-01	7.735E+10	8.964E+14	5.044E+10	4.328E+07
6.00e-01	to	8.00e-01	4.190E+10	2.331E+15	3.421E+10	1.335E+09
8.00e-01	to	1.00e+00	1.382E+11	3.795E+14	3.231E+10	7.909E+10
1.00e+00	to	1.33e+00	1.744E+13	2.845E+14	3.844E+13	1.211E+13
1.33e+00	to	1.66e+00	4.924E+12	8.354E+13	1.086E+13	3.420E+12
1.66e+00	to	2.00e+00	2.479E+03	1.565E+12	5.201E+03	1.592E+03
2.00e+00	to	2.50e+00	1.178E+08	2.450E+12	2.597E+08	8.184E+07
2.50e+00	to	3.00e+00	1.007E+05	1.225E+11	2.219E+05	6.992E+04
3.00e+00	to	4.00e+00	1.121E-05	1.150E+10	5.738E-05	9.234E-06
4.00e+00	to	5.00e+00	0.0	4.195E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	1.684E+07	0.0	0.0
6.50e+00	to	8.00e+00	0.0	3.303E+06	0.0	0.0
8.00e+00	to	1.00e+01	0.0	7.013E+05	0.0	0.0
Total Gamma, g/(sec*FA)			2.319E+13	6.022E+15	5.038E+13	1.588E+13
(1)Total Neutrons, n/(sec*FA)			1.214E+9			

(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for a subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where the dry k_{eff}= 0.25189 and bpf=1.152.

Table U.5-26
Modified HLZC#1 1.5 kW Design Basis HSM Source Term

Bounding Radiological Source at 380 kgU/FA: 19 GWD/MTU, 0.8 wt. %, after 2.0 years of cooling						
E_{min}, MeV	***to***	***E_{max}, MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	to	5.00e-02	6.587E+11	2.781E+15	8.406E+11	1.823E+11
5.00e-02	to	1.00e-01	5.321E+10	9.152E+14	1.151E+11	3.600E+10
1.00e-01	to	2.00e-01	2.874E+10	8.695E+14	3.811E+10	8.770E+09
2.00e-01	to	3.00e-01	1.962E+09	2.352E+14	2.258E+09	4.384E+08
3.00e-01	to	4.00e-01	9.786E+09	1.850E+14	7.645E+09	5.686E+08
4.00e-01	to	6.00e-01	8.573E+10	1.097E+15	5.771E+10	8.519E+08
6.00e-01	to	8.00e-01	6.524E+10	1.558E+15	4.580E+10	6.451E+08
8.00e-01	to	1.00e+00	8.305E+11	2.954E+14	1.523E+11	4.729E+11
1.00e+00	to	1.33e+00	1.505E+13	2.656E+14	3.330E+13	1.046E+13
1.33e+00	to	1.66e+00	4.250E+12	8.283E+13	9.403E+12	2.955E+12
1.66e+00	to	2.00e+00	1.899E+07	6.288E+12	4.061E+07	1.254E+07
2.00e+00	to	2.50e+00	1.018E+08	1.676E+13	2.250E+08	7.071E+07
2.50e+00	to	3.00e+00	8.696E+04	4.659E+11	1.923E+05	6.042E+04
3.00e+00	to	4.00e+00	4.193E-06	4.269E+10	2.146E-05	3.454E-06
4.00e+00	to	5.00e+00	0.0	3.420E+06	0.0	0.0
5.00e+00	to	6.50e+00	0.0	1.373E+06	0.0	0.0
6.50e+00	to	8.00e+00	0.0	2.692E+05	0.0	0.0
8.00e+00	to	1.00e+01	0.0	5.716E+04	0.0	0.0
Total Gamma, g/(sec*FA)			2.104E+13	8.309E+15	4.396E+13	1.412E+13
(1)Total Neutrons, n/(sec*FA)			9.818E+7			
(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by $bpf/(1-k_{eff})$ to account for a subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where the dry k_{eff}= 0.34041 and bpf=1.403.						

Table U.5-27
Modified HLZC#1 1.5 kW Design Basis TC Source Term

Bounding Radiological Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 3.9 years of cooling						
E_{min}, MeV	***to***	***E_{max}, MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	to	5.00e-02	5.519E+11	1.577E+15	8.838E+11	2.314E+11
5.00e-02	to	1.00e-01	6.655E+10	4.610E+14	1.446E+11	4.543E+10
1.00e-01	to	2.00e-01	3.320E+10	3.998E+14	4.602E+10	1.102E+10
2.00e-01	to	3.00e-01	2.018E+09	1.132E+14	2.532E+09	5.455E+08
3.00e-01	to	4.00e-01	5.180E+09	8.171E+13	4.967E+09	7.131E+08
4.00e-01	to	6.00e-01	9.170E+10	1.164E+15	5.978E+10	4.821E+07
6.00e-01	to	8.00e-01	4.940E+10	2.595E+15	3.909E+10	1.337E+09
8.00e-01	to	1.00e+00	2.363E+11	4.727E+14	4.906E+10	1.349E+11
1.00e+00	to	1.33e+00	1.904E+13	3.200E+14	4.198E+13	1.323E+13
1.33e+00	to	1.66e+00	5.377E+12	9.694E+13	1.186E+13	3.735E+12
1.66e+00	to	2.00e+00	2.627E+04	2.466E+12	5.633E+04	1.741E+04
2.00e+00	to	2.50e+00	1.287E+08	4.224E+12	2.837E+08	8.938E+07
2.50e+00	to	3.00e+00	1.099E+05	1.936E+11	2.424E+05	7.637E+04
3.00e+00	to	4.00e+00	1.137E-05	1.809E+10	5.821E-05	9.368E-06
4.00e+00	to	5.00e+00	0.0	4.308E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	1.729E+07	0.0	0.0
6.50e+00	to	8.00e+00	0.0	3.392E+06	0.0	0.0
8.00e+00	to	1.00e+01	0.0	7.203E+05	0.0	0.0
Total Gamma, g/(sec*FA)			2.546E+13	7.288E+15	5.507E+13	1.739E+13
(1)Total Neutrons, n/(sec*FA)			1.247E+9			
(1)*This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for a subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where the dry k_{eff}=0.25189 and bpf=1.152.*						

Table U.5-28
Comparison of Bounding Average Dose Rates with NUHOMS®-32PTH1 Bounding DSC in HSM-H with 0.490 MTU/FA and 0.380 MTU/FA to Derive Dose Rate Scaling Factors for Storage

Dose Rate Location	Average Gamma (mrem/hour)		Average Neutron (mrem/hour)		Average Total (mrem/hour)				-
	0.490 MTU	0.380 MTU	0.490 MTU	0.380 MTU	0.490 MTU	0.380 MTU	DR Change	Derived Scaling Factors	Maximum Derived Scaling Factors
HSM Roof	11.27	13.47	0.62	0.51	11.89	13.98	+2.09	1.18	Max R+F = 1.18
HSM Front	14.87	12.60	0.31	0.23	15.19	12.83	-2.36	0.84	
HSM End (Side) Shield Wall Surface	0.36	0.50	0.02	0.02	0.38	0.52	+0.14	1.36	Max SSW= 1.36
HSM Back Shield Wall	0.07	0.07	0.00	0.00	0.07	0.07	0.00	1.06	

Table U.5-29
Verification of Derived Dose Rate Scaling Factors for Predicting Maximum Dose Rates for NUHOMS®-32PTH1 Bounding DSC in HSM-H with 0.380 MTU/FA for Storage

Dose Rate Location	Maximum Gamma (mrem/hour)		Maximum Neutron (mrem/hour)		Maximum Total (mrem/hour)				Scaling Factor Applied to 0.490 MTU/FA Max Total
	0.490 MTU	0.380 MTU	0.490 MTU	0.380 MTU	0.490 MTU	0.380 MTU	DR Change	Scaled 0.380 MTU/FA Results	
HSM Roof (centerline)	14.13	16.49	0.74	0.59	14.86	17.08	+2.22	17.47	R+F 1.18 scaling factor is applied
HSM Roof Birdscreen	114.71	123.93	6.47	0.04	121.17	123.97	+2.8	142.45	
HSM Door Exterior Surface (centerline)	0.61	0.18	0.08	0.20	0.70	0.38	-0.32	0.82	
HSM Front Birdscreen	471.28	476.86	5.89	4.78	477.17	481.64	+4.47	560.96	
HSM End (Side) Shield Wall Surface	1.49	2.15	0.06	0.04	1.54	2.19	+0.65	2.09	SSW 1.36 scaling factor is applied

Table U.5-30
Comparison of Bounding Maximum Dose Rates with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Dose Rate Scaling Factors for the Transfer Configuration

Transfer Configuration	*Maximum Gamma (mrem/hour)*		*Maximum Neutron (mrem/hour)*		*Maximum Total (mrem/hour)*			*Maximum Derived Scaling Factors*
Dose Rate Location	*0.490 MTU/FA*	*0.380 MTU/FA*	*0.490 MTU/FA*	*0.380 MTU/FA*	*0.490 MTU/FA*	*0.380 MTU/FA*	*Derived Scaling Factor*	
Cask Side Surface (Radial)	*407.0*	*501.2*	*202.0*	*136.3*	*609.0*	*637.6*	*1.05*	*Max Side = 1.14*
1.5 ft from Cask Side (Radial)	*239.0*	*314.1*	*123.0*	*82.9*	*362.0*	*397.0*	*1.10*	
3 ft from Cask Side (Radial)	*161.0*	*222.1*	*83.9*	*57.4*	*245.0*	*279.5*	*1.14*	
Cask Top Axial Surface	*232.0*	*363.8*	*38.6*	*29.9*	*251.0*	*380.7*	*1.52*	*Max Top = 1.60*
1.5 ft from Cask Top Axial Surface	*47.1*	*85.0*	*20.7*	*16.2*	*59.5*	*95.2*	*1.60*	
3 ft from Cask Top Axial Surface	*29.5*	*45.3*	*14.4*	*10.3*	*37.9*	*52.6*	*1.39*	
Cask Bottom Axial Surface	*2150.0*	*4129.6*	*1400.0*	*806.3*	*3550.0*	*4935.9*	*1.39*	*Max Bottom = 1.51*
1.5 ft from Cask Bottom Axial Surface	*925.0*	*1732.0*	*343.0*	*184.2*	*1270.0*	*1916.2*	*1.51*	
3 ft from Cask Bottom Axial Surface	*465.0*	*661.2*	*140.0*	*125.9*	*605.0*	*710.9*	*1.18*	

Table U.5-31
Comparison of Bounding Maximum Dose Rates with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Dose Rate Scaling Factors for the Decontamination Configuration

Decontamination Configuration	Maximum Gamma (mrem/hour)		Maximum Neutron (mrem/hour)		Maximum Total (mrem/hour)			Maximum Derived Scaling Factors
Dose Rate Location	0.490 MTU/FA	0.380 MTU/FA	0.490 MTU/FA	0.380 MTU/FA	0.490 MTU/FA	0.380 MTU/FA	Derived Scaling Factor	
Cask Side Surface (Radial)	346.0	435.1	373.0	314.6	719.0	749.7	1.04	Max Side = 1.10
1.5 ft from Cask Side (Radial)	202.0	267.3	229.0	194.7	431.0	462.0	1.07	
3 ft from Cask Side (Radial)	135.0	185.6	158.0	135.4	292.0	321.0	1.10	
Top Axial Surface	832.0	1320.7	9.3	9.9	833.0	1321.5	1.59	Max Top = 1.59
1.5 ft from Top Axial Surface	615.0	978.1	5.6	5.6	616.0	978.7	1.59	
3 ft from Top Axial Surface	427.0	675.5	3.6	3.6	428.0	676.5	1.58	
Cask Bottom Axial Surface	1700.0	3254.8	68.3	133.6	1770.0	3388.5	1.91	Max Bottom = 1.91
1.5 ft from Cask Bottom Axial Surface	731.0	1371.3	18.2	42.4	749.0	1413.7	1.89	
3 ft from Cask Bottom Axial Surface	368.0	662.1	8.1	11.9	376.0	662.1	1.76	

Table U.5-32
Comparison of Bounding Maximum Dose Rates with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Dose Rate Scaling Factors for the Welding Configuration

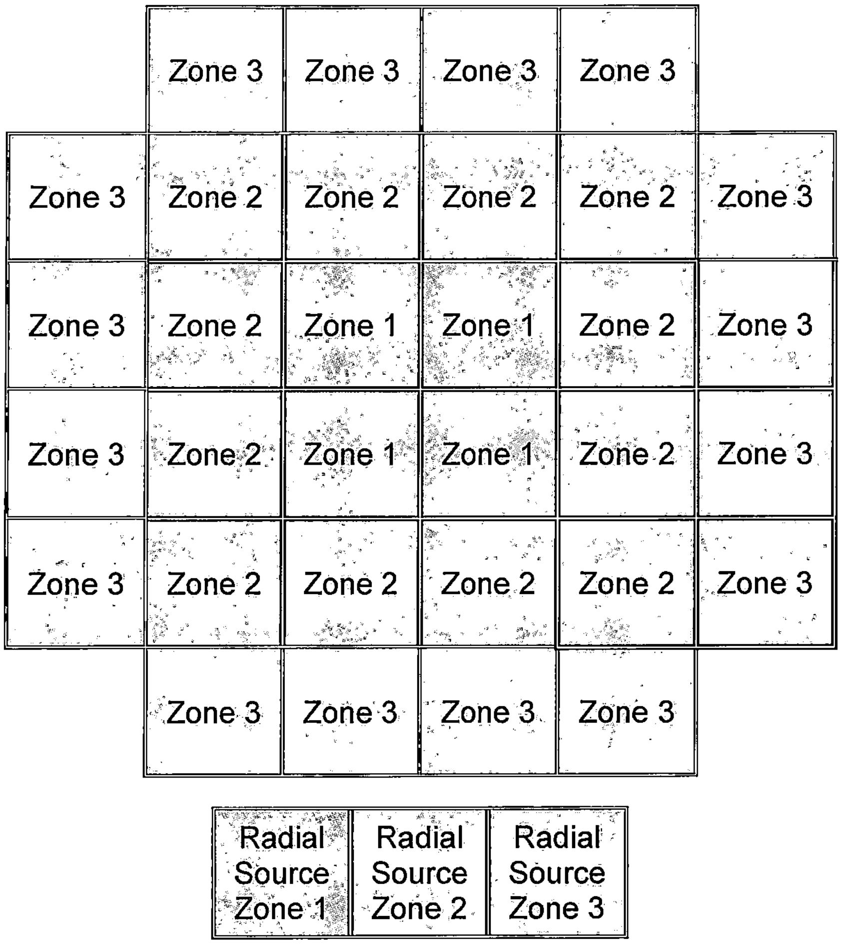
Welding Configuration	*Maximum Gamma (mrem/hour)*		*Maximum Neutron (mrem/hour)*		*Maximum Total (mrem/hour)*			*Maximum Derived Scaling Factors*
Dose Rate Location	***0.490 MTU/FA***	***0.380 MTU/FA***	***0.490 MTU/FA***	***0.380 MTU/FA***	***0.490 MTU/FA***	***0.380 MTU/FA***	***Derived Scaling Factor***	
Cask Side Surface (Radial)	*309.0*	*424.0*	*147.0*	*97.2*	*456.0*	*521.2*	*1.14*	*Max Side = 1.24*
1.5 ft from Cask Side (Radial)	*185.0*	*270.3*	*89.9*	*59.4*	*275.0*	*329.7*	*1.20*	
3 ft from Cask Side (Radial)	*127.0*	*193.2*	*62.3*	*41.3*	*189.0*	*234.6*	*1.24*	
Top Axial Surface	*720.0*	*1202.1*	*42.1*	*36.8*	*762.0*	*1238.9*	*1.63*	*Max Top = 1.63*
1.5 ft from Top Axial Surface	*425.0*	*653.3*	*17.4*	*17.4*	*442.0*	*670.7*	*1.52*	
3 ft from Top Axial Surface	*294.0*	*459.9*	*11.7*	*11.8*	*306.0*	*470.6*	*1.54*	
Cask Bottom Axial Surface	*2170.0*	*4133.4*	*1220.0*	*695.1*	*3390.0*	*4828.5*	*1.42*	*Max Bottom = 1.54*
1.5 ft from Cask Bottom Axial Surface	*927.0*	*1731.4*	*300.0*	*167.8*	*1230.0*	*1899.2*	*1.54*	
3 ft from Cask Bottom Axial Surface	*466.0*	*668.9*	*121.0*	*93.1*	*587.0*	*706.1*	*1.20*	

Table U.5-33
Comparison of Bounding Maximum Dose Rates and Cumulative Doses with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Occupational Exposure Scaling Factor for Pool to Pad Loading Operations

Location	Task Description	Area Dose Rate w/0.490 MTU/FA (mrem/hr)	Total Exposure w/0.490 MTU/FA (person-mrem)	Area Dose Rate w/0.380 MTU/FA (mrem/hr)	Total Exposure w/0.380 MTU/FA (person-mrem)	Area Dose Rate and Total Exposure Change
Auxiliary Building and Fuel Pool	Place the DSC into the Transfer Cask	2	8	2	8	1.00
	Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal	2	6	2	6	1.00
	Fill the DSC Cavity with Water (borated for PWRs)	2	12	2	12	1.00
	Place the Cask Containing the DSC in the Fuel Pool	2	5	2	5	1.00
	Verify and Load the Candidate Fuel Assemblies into the DSC	2	30	2	30	1.00
	Place the Top Shield Plug on the DSC	2	4	2	4	1.00
	Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area	2	5	2	5	1.00
		199	7	229	8	1.15
		146	98	169	112	1.15
Cask Decontamination Area	Decontaminate the Outer Surface of the Cask	146	256	169	295	1.15
		2	2	2	2	1.00
	Decontaminate the Top Region of the Cask and DSC	195	97	293	147	1.51
		64	32	90	45	1.41
	Drain Water from the DSC	199	17	229	19	1.15
	Remove Cask/DSC Annulus Seal and Set-Up Welding Machine	337	56	515	86	1.53
		87	65	123	92	1.41
		72	36	97	49	1.35
	Weld the Inner Top Cover to the DSC Shell and Perform NDE (PT)	2	24	2	24	1.00
		169	56	251	84	1.49
	Drain the Cask/DSC Annulus and the DSC Cavity	87	22	123	31	1.41
		169	3	251	4	1.49
		2	1	2	1	1.00
	Vacuum Dry and Backfill the DSC with Helium	72	36	97	49	1.35
		2	120	2	120	1.00
	Helium Leak Test the Shield Plug Weld	2	4	2	4	1.00
	Seal Weld the Prefabricated Plugs to the Vent and Siphon Port and Perform NDE (PT)	87	44	123	61	1.41
	Fit-Up the DSC Top Cover Plate	169	42	251	63	1.49
		87	44	123	61	1.41
	Weld the Outer Top Cover Plate to DSC Shell and Perform NDE (PT)	72	72	97	97	1.35
		169	28	251	42	1.49
		2	56	2	56	1.00
		169	56	251	84	1.49
	Install The Cask Lid	93	124	115	153	1.23

Table U.5-33
Comparison of Bounding Maximum Dose Rates and Cumulative Doses with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Occupational Exposure Scaling Factor for Pool to Pad Loading Operations
(Continued)

Location	Task Description	Area Dose Rate w/0.490 MTU/FA (mrem/hr)	Total Exposure w/0.490 MTU/FA (person-mrem)	Area Dose Rate w/0.380 MTU/FA (mrem/hr)	Total Exposure w/0.380 MTU/FA (person-mrem)	Area Dose Rate and Total Exposure Change
Reactor/ Fuel Building Bay	Ready the Cask Support Skid and Transport Trailer for the Service	2	8	2	8	1.00
	Place the Cask onto the Skid and Trailer	136	68	158	79	1.16
	Secure the Cask to the Skid	136	34	158	40	1.16
ISFSI Site	Ready The Cask Support Skid and Transport Trailer for the Service	negligible	0	negligible	0	-
	Transport the Cask to ISFSI	negligible	0	negligible	0	-
	Position the Cask in Close Proximity with the HSM	negligible	0	negligible	0	-
	Remove the Cask Lid	42	56	47	63	1.12
	Align and Dock the Cask with the HSM	108	54	130	65	1.20
	Position and Align Ram with Cask	174	174	222	222	1.28
	Remove Ram Access Cover Plate	562	47	757	63	1.35
	Transfer the DSC from the Cask to the HSM	negligible	0	negligible	0	-
	Lift the Ram Back onto the Trailer and Un-Dock the Cask from the	56	9	65	11	1.17
	Install HSM Access Door	15	15	15	15	1.00
	Totals	-	1934	-	2425	1.25



Design Basis Analysis			
Heat Zone	**Zone 1**	**Zone 2**	**Zone 3**
Number of Fuel Assemblies	4	12	16
Maximum Decay Heat (kW/FA)	1.0	1.5	1.5
Maximum Decay Heat per Zone (kW)	4.0	18.0	24.0
Maximum Decay Heat per DSC (kW)	46.0		

Scaling Factor Analysis			
Heat Zone	***Zone 1***	***Zone 2***	***Zone 3***
Number of Fuel Assemblies	*4*	*12*	*16*
Maximum Decay Heat (kW/FA)	*0.8*	*1.3*	*1.5*
Maximum Decay Heat per Zone (kW)	*3.2*	*15.6*	*24.0*
Maximum Decay Heat per DSC (kW)	*42.8*		

Figure U.5-3
32PTH1 DSC Bounding HLZC Used for Shielding Analysis for All Configurations except for 32PTH1 Type 2-W

U.9 Acceptance Tests and Maintenance Program

Background for this particular UFSAR chapter:

Beginning with CoC 1004 Amendment 10, which was incorporated into UFSAR Revision 11, Chapter U.9, "Acceptance Tests and Maintenance Program," contained information which was incorporated by reference into the Technical Specifications (TS) associated with a particular amendment. It is known that certain general licensees reconcile the CoC 1004 UFSAR revisions provided to them to their loaded systems, pursuant to 10 CFR 72.48 and 10 CFR 72.212. In doing so they sometimes find the changed UFSAR portions incorporated by reference into the TS to be impossible to reconcile because the 10 CFR 72.48 regulation does not allow proposed activities which involve changes to the TS.

In order to facilitate this reconciliation process by general licensees, the following statements are provided, addressing the licensing basis for certain amendments, as they relate to certain UFSAR chapters which contain TS incorporated by reference. Additionally, so that the actual information is contained in the current CoC 1004 UFSAR, to facilitate the reconciliation by general licensees, the UFSAR Revision 11, 12, 13, and 14 versions of Chapter U.9 are inserted and annotated in this part of the UFSAR. For clarity, this includes annotating the version of Chapter U.9 directly associated with the latest UFSAR revision in which a change to Chapter U.9 occurred.

- Systems loaded to CoC 1004 Amendment 10 have Technical Specifications incorporated by reference from UFSAR Revisions 11 and 12 in Chapter U.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to systems loaded to Amendment 10.
- Systems loaded to CoC 1004 Amendment 11 have Technical Specifications incorporated by reference from UFSAR Revision 13 Chapter U.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 11.
- Note that CoC 1004 Amendment 12 was submitted and docketed, associated with a U.S. Department of Energy project, but due to a lack of review funding the NRC returned it without a review.
- Systems loaded to CoC 1004 Amendment 13 have Technical Specifications incorporated by reference from UFSAR Revisions 14 and 15 Chapter U.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 13.
- Systems loaded to CoC 1004 Amendment 14 have Technical Specifications incorporated by reference by FCN 721004-1575, which will be incorporated into UFSAR Revisions 16 and 17 Chapter U.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 14.
- *Systems loaded to CoC 1004 Amendment 15 have Technical Specifications incorporated by reference from UFSAR Revision 18 Chapter U.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 15.*

U.9 Acceptance Tests and Maintenance Program

U.9.1 Acceptance Tests

The pre-operational testing requirements for the NUHOMS® system are given in Chapter 9.0, with the exceptions described in the following sections. The NUHOMS®-32PTH1 DSC has been enhanced to provide leaktight confinement and the basket includes an updated poison plate design. Additional acceptance testing of the NUHOMS®-32PTH1 DSC welds and of the poison plates are described.

U.9.1.1 Visual Inspection

Visual inspections are performed at the fabricator's facility to ensure that the DSC, the Transfer Cask and the HSM conform to the drawings and specifications. The visual inspections include weld, dimensional, surface finish, and cleanliness inspections. Visual inspections specified by codes applicable to a component are performed in accordance with the requirements and acceptance criteria of those codes.

All weld inspection is performed using qualified processes and qualified personnel according to the applicable code requirements, e.g., ASME or AWS. Non-destructive examination (NDE) requirements for welds are specified on the drawings provided in Chapter U.1; acceptance criteria are as specified by the governing code. NDE personnel are qualified in accordance with SNT-TC-1A [9.2].

The confinement welds on the DSC are inspected in accordance with ASME B&PV Code Subsection NB [9.1] including alternatives to ASME Code specified in Section U.3.1.2.3.

DSC non-confinement welds are inspected to the NDE acceptance criteria of ASME B&PV Code Subsection NG or NF, based on the applicable code for the components welded.

U.9.1.2 Structural

The DSC confinement boundary except the inner top cover/shield plug to the DSC shell weld is pressure tested at the fabricator's shop in accordance with ASME Article NB-6300. The test pressure is set between 21.5 to 23.0 psig for 32PTH1 DSC for future 10CFR71 application. This bounds the 1.1xDSC design pressure of 15 psig.

The inner top cover/shield plug to the DSC shell weld is also pressure tested using a test pressure between 21.5 to 23.0 psig for 32PTH1 DSC at the field after the fuel assemblies are loaded in the DSC. This test is in accordance with the alternatives to the ASME Code specified in Section U.3.1.2.3.

HSM-H reinforcement and concrete are tested as described in Section 3.4.2.

U.9.1.3 Leak Tests

DSC confinement welds in the DSC shell and bottom are leak tested at the fabricator's shop to an acceptance criterion of $1x10^{-7}$ ref cm^3/s, i.e., "leaktight" as defined in ANSI N14.5 [9.4].

Personnel performing the leak test are qualified in accordance with SNT-TC-1A [9.2].

The weld between the DSC shell and inner top cover/shield plug and the siphon/vent cover welds are also leak tested to an acceptance criteria of 1 x 10^{-7} ref cm^3/s in the field after the fuel assemblies are loaded in the canister.

U.9.1.4 Components

The NUHOMS® System does not include any components such as valves, rupture discs, pumps, or blowers. No other components of the NUHOMS® System require testing, except as discussed in this chapter.

U.9.1.5 Shielding Integrity

The Transfer Cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6" × 6" grid, the detector will encompass a 6" × 6" square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed.

The gamma and neutron shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

U.9.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section U.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section U.9.1.7.6.

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U.9.1.7 Poison Acceptance

CAUTION

Sections U.9.1.7.1 through U.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated *a*luminum

(b) Boron carbide / *a*luminum metal matrix composite (MMC)

(c) *BORAL®*

The 32PTH1 DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content for these materials is given in Table U.9-1.

References to metal matrix composites throughout this chapter are not intended to refer to *BORAL®*, which is described later in this section.

U.9.1.7.1 **Borated Aluminum**

See the Caution in Section U.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also occur)*. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section U.9.1.7.7. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

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U.9.1.7.2 Boron Carbide / Aluminum Metal Matrix Composites (MMC)

See the Caution in Section U.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, *molten metal infiltration,* or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding *or produced by molten metal infiltration* shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B_4C particles in MMCs shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 32PTH1 DSC, MMCs shall pass the qualification testing specified in Section U.9.1.7.8, and shall subsequently be subject to the process controls specified in Section U.9.1.7.9.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section U.9.1.7.7. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

U.9.1.7.3 BORAL®

See the Caution in Section U.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns. The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of BORAL®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken from the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

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U.9.1.7.4 **Visual Inspections of Neutron Absorbers**

See the Caution in Section U.9.1.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. Blisters shall be treated as non-conforming. *For clad MMCs and for* BORAL®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet. *Material that does not meet these criteria shall be reworked, repaired, or scrapped.*

U.9.1.7.5 Other Visual Inspections Criteria (non-Technical Specifications)

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4, "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

U.9.1.7.6 Thermal Conductivity Testing

*Acceptance t*esting shall conform to ASTM E1225[1], ASTM E1461[2], or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, and may be reduced if the first five tests meet the specified minimum thermal conductivity. *For cast products, the lot shall be defined by the heat or ingot. For other products, the lot shall be defined as material produced in a single production campaign using the same heat or lots of aluminum and boron carbide feed materials.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The measured thermal conductivity values shall satisfy the minimum required conductivities as

[1] ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique."
[2] ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method."

specified in Section U.4.3.

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section U.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

U.9.1.7.7 **Specification for Acceptance Testing of Neutron Absorber *Content***

Acceptance testing for neutron absorber content shall be performed by either neutron transmission or by B-10 volume density measurement.

CAUTION

Portions of Section U.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

U.9.1.7.7.1 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

a) Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.1 inch diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are

composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard.

Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than the statistically derived minimum thickness from U.9.1.7.7 a) or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

U.9.1.7.7.2 *Specification for Acceptance Testing of Neutron Absorbers by B-10 Volume Density Measurement*

a) B-10 volume density measurement acceptance testing procedures shall be subject to approval by the certificate holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, as long as it

results in accumulating material that is uniform for sampling purposes.

The sampling rate for B-10 volume density measurements shall be such that there is at least one density measurement for each 2000 square inches of final product in each lot.

Areal density is determined by measuring the B-10 volume density in test samples and converting the measured values to areal density. The method of measurement of B-10 volume density shall be subject to approval by the certificate holder. The method of measurement of B-10 volume density shall be qualified against neutron transmission testing. Results of the two test methods shall be compared and a penalty shall be derived to account for the performance based results of neutron transmission testing.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B-10 areal densities are determined by volume density as described above. The lower tolerance limit of B-10 volume density is then determined, defined as the mean value of B-10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6]. Finally, the minimum specified value of B-10 areal density is divided by the lower tolerance limit of B-10 volume density to arrive at the minimum plate thickness that provides the specified B-10 areal density.

Any plate that is thinner than the statistically derived minimum thickness from U.9.1.7.7.2 a) or the minimum design thickness, whichever is greater, shall be treated as nonconforming, with the following exception. Local depressions are acceptable, as long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

U.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

CAUTION

Portions of Section U.9.1.7.8.4, and Section U.9.1.7.8.5, are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

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U.9.1.7.8.1 Applicability and Scope

Metal matrix composites (MMCs) acceptable for use in the 32PTH1 DSC are described in Section U.9.1.7.2.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section U.9.1.7.9 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the Certificate Holder.

U.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in Section U.9.1.7.8.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section U.9.1.7.8.5.

U.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for unclad MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for MMCs (clad or unclad) consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

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3 Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998.

4 Boralyn testing submitted to the NRC under docket 71-1027, 1998.

U.9.1.7.8.4 **Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity**

At least three samples, one each from approximately the two ends and middle of the qualification material run shall be subject to:

a) room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:

- **Minimum yield strength, 0.2% offset: 1.5 ksi**
- **Minimum ultimate strength: 5 ksi**
- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

b) Testing to verify more than 98% of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a means to be approved by the Certificate Holder. The maximum interconnected porosity is 0.5 volume %.

c) Delamination Testing of Clad MMC

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage. *An example of such a test would be: (1) immerse a specimen at least 6 x 6 inches in water under pressure ≥30 psig for at least 24 hours, (2) place the specimen in a vacuum furnace preheated to at least 300 °F and evacuate the furnace. Acceptance criterion: no blistering or delamination of the cladding.*

U.9.1.7.8.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

Uniformity of the boron distribution shall be verified either by:

a) Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run,

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5 ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.
6 ASTM E290, Standard Methods for Bend Testing of Materials for Ductility.
7 ASTM E94, Recommended Practice for Radiographic Testing.
8 ASTM E142, Controlling Quality of Radiographic Testing.
9 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing.

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verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section U.9.1.7.7, or by chemical analysis for boron carbide content in the composite.

U.9.1.7.8.6 Qualification Report

Qualification *reports* shall be *prepared by, or* subject to approval by the Certificate Holder.

U.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION

Sections U.9.1.7.9.1 and U.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

U.9.1.7.9.1 Applicability and Scope

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section U.9.1.7.8 is required, depending on the characteristics of the material that could be affected by the process change.

U.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, reduce corrosion resistance, reduce the mechanical strength or ductility of the MMC.

U.9.1.7.9.3 Identification and Control of Key Process Changes

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section U.9.1.7.9.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that are established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average *(d50)* particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a magnesium-alloyed aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, and

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

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U.9.2 Maintenance Program

The NUHOMS®-32PTH1 system is a totally passive system and therefore will require little, if any, maintenance over the lifetime of the ISFSI. Typical NUHOMS®-32PTH1 system maintenance tasks will be performed in accordance with the UFSAR.

U.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition *through* 2000 Addenda.

9.2 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

9.3 Deleted.

9.4 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.5 "Aluminum Standards and Data, 2003" The Aluminum Association.

9.6 Natrella, "Experimental Statistics," Dover, 2005.

9.7 Deleted.

9.8 Deleted.

9.9 Deleted.

9.10 Deleted.

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Table U.9-1
B10 Specification for the NUHOMS® 32PTH1 Poison Plates

Poison Type	32PTH1 Basket Type	Minimum Poison Loading (B10 mg/cm²)	% Credit Used in Criticality Analysis
Borated Aluminum /MMC	1A or 2A	7	90
	1B or 2B	15	
	1C or 2C	20	
	1D or 2D	32	
	1E or 2E	50	
BORAL®	1A or 2A	9	75
	1B or 2B	19	
	1C or 2C	25	
	1D or 2D	N/A	
	1E or 2E	N/A	

U.10.1 Occupational Exposure

The occupational exposure results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section U.5.4.12. Loading 0.380 MTU fuel results in an increase in occupational exposure of 25%.

The expected occupational dose for placing a canister of spent fuel into dry storage is based on the operational steps outlined in Table 7.4-1 of the UFSAR. The total exposure for the occupational dose due to placing a single NUHOMS® 32PTH1 DSC loaded with design basis fuel assemblies into storage is conservatively estimated to be 2 person-rem as summarized in Table U.10-1. This is a very conservative estimate because the dose rates on and around 32PTH1 DSCs used in these calculations are based on very conservative assumptions for the design-basis source terms and analyses models. The calculated exposures are due mainly to the expected gamma dose rate during preparation for welding.

The NUHOMS® 32PTH1 system loading operations, the number of workers required for each operation, and the amount of time required for each operation are presented in Table U.10-1. This information is used as the basis for estimating the total occupational exposure associated with one fuel load. The dose rates applicable for each operation are based on the results presented in Section U.5.4 for loading operations. Engineering judgment and operational experience are used to estimate dose rates that were not explicitly evaluated. This evaluation assumes that a transfer trailer/skid with an integral ram is used for the DSC transfer operations. Licensees may elect to use different equipment and/or different procedures. Each Licensee must evaluate any such changes in accordance with its ALARA program.

Unique steps are sometimes necessary at the individual site to load the canister, complete closure operations and place the canister in the HSM-H. Specifically, the licensee may choose to modify the sequence of operations in order to achieve reduced dose rates for a larger number of steps, with the end result of reduced total exposure. The only requirement is that the licensee practice ALARA with respect to the total exposure received for a loading campaign. These estimated durations, manloading and dose rates are not limits.

The amount of time required to complete some operations as identified in Table U.10-1 may be greater than the actual amount of time spent in a radiation field. The process of vacuum drying the DSC includes setting up the vacuum drying system (VDS), verifying that the VDS is operating correctly, evacuating the DSC cavity, monitoring the DSC pressure, and disconnecting the VDS from the DSC. Of these tasks, only setup and removal of the VDS require a worker to spend time near the DSC. The most time consuming task, evacuating the DSC, does not require anyone to be present near DSC at all. The total exposure calculated for each task is therefore not necessarily equal to the number of workers multiplied by the total time required, multiplied by a dose rate. The exposure estimation for each task correctly accounts for cases such as vacuum drying and assumes that good ALARA practices are followed.

Localized regions of elevated dose rates should be anticipated and minimized with good ALARA practices. Such regions exist due primarily to radiation streaming, including for example, streaming through the cask/DSC annulus, the ventilation paths in OS200 lid and the DSC vent/siphon ports.

The results of the evaluations of the NUHOMS® 32PTH1 are presented in Table U.10-1.

The potential for streaming due to gaps associated with the alternate top closure have been evaluated using standard hand calculation methods. In this configuration streaming paths around

Table U.10-1
Occupational Exposure Summary, 32PTH1 System

Location	Task Description	# of workers	Duration (hr)	Area Dose Rate (mrem/hr)	Total Exposure (person-mrem)
Auxiliary Building and Fuel Pool	Place the DSC into the Transfer Cask	2	2	2	8
	Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal	3	1	2	6
	Fill the DSC Cavity with Water	1	6	2	12
	Place the Cask Containing the DSC in the Fuel Pool	5	0.5	2	5
	Verify and Load the Candidate Fuel Assemblies into the DSC	3	5	2	30
	Place the Top Shield Plug on the DSC	2	1	2	4
	Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area	5	0.5	2	5
		1	0.033	199	7
		1	0.667	146	98
Cask Decontamination Area	Decontaminate the Outer Surface of the Cask	1	1.75	146	256
		1	1	2	2
	Decontaminate the Top Region of the Cask and DSC	1	0.5	195	97
		1	0.5	64	32
	Drain Water from the DSC	1	0.083	199	17
	Remove Cask/DSC Annulus Seal and Set-Up Welding Machine	1	0.167	337	56
		1	0.75	87	65
		1	0.5	72	36
	Weld the Inner Top Cover to the DSC Shell and Perform NDE (PT)	2	6	2	24
		1	0.33	169	56
	Drain the Cask/DSC Annulus and the DSC Cavity	1	0.25	87	22
		1	0.017	169	3
		1	0.5	2	1
	Vacuum Dry and Backfill the DSC with Helium	1	0.5	72	36
		2	30	2	120
	Helium Leak Test the Shield Plug Weld	2	1	2	4
	Seal Weld the Prefabricated Plugs to the Vent and Siphon Ports and Perform NDE (PT)	1	0.5	87	44
	Fit-Up the DSC Outer Top Cover Plate	1	0.25	169	42
		1	0.5	87	44
	Weld the Outer Top Cover Plate to DSC Shell and Perform NDE (PT)	1	1	72	72
		1	0.167	169	28
		2	14	2	56
		1	0.333	169	56
	Install the Cask Lid	2	0.667	93	124
Reactor/ Fuel Building Bay	Ready the Cask Support Skid and Transfer Trailer for Service	2	2	2	8
	Place the Cask onto the Skid and Trailer	2	0.25	136	68
	Secure the Cask to the Skid	1	0.25	136	34
ISFSI Site	Ready the Cask Support Skid and Transfer Trailer for Service	2	2	negligible	0
	Transfer the Cask to ISFSI	6	1	negligible	0
	Position the Cask in Close Proximity with the HSM	3	1	negligible	0
	Remove the Cask Lid	2	0.67	42	56
	Align and Dock the Cask with the HSM	2	0.25	108	54
	Position and Align Ram with Cask	2	0.5	174	174
	Remove Ram Access Cover Plate	1	0.083	562	47
	Transfer the DSC from the Cask to the HSM	3	0.5	negligible	0
	Lift the Ram Back onto the Trailer and Un-Dock the Cask from the HSM	2	0.083	56	9
	Install HSM Access Door	2	0.5	15	15
Totals		N/A	87	N/A	1934

Total estimated dose is 2 person-rem per 32PTH1 canister load.
Total estimated dose increases by 25% when loading 0.380 MTU/FA

U.11.2.3.1 Cause of Accident

No change to the description presented in UFSAR Section 8.2.2.1. No change to the determination of the tornado wind and tornado missile loads acting on the HSM-H / HSM-HS as detailed in Appendix P, Section P.2.2.1.

U.11.2.3.2 Accident Analysis

An evaluation that investigates the effect of the addition of the NUHOMS® 32PTH1 DSC, is presented in Chapter U.3, Section U.3.7.1. The evaluation of the HSM-H for the effect of DBT wind pressure loads is addressed in Section U.3.7.1.1. The tornado missile impact evaluation of the HSM-H / HSM-HS is presented in the following sections.

U.11.2.3.2.1 HSM-H/HSM-HS Missile Impact Analysis

No change to the missile impact evaluation presented in Appendix P, Section P.11.2.3.2.1.

To accommodate the longest 32PTH1 DSC inside the HSM-H / HSM-HS cavity, the concrete thickness of the shield door is reduced from 22.5 inches to 18.5 inches. The missile evaluations of the shielded composite door, described in Section P.11.2.3.2.1, do not take credit for the 22.5 inch concrete thickness that is structurally composite with the 7.875 inch total steel plate thickness. Therefore, the shielded door evaluations for missile loadings as presented in Section P.11.2.3.2.1 remain applicable for the shielded door with reduced concrete thickness.

In addition, as shown in the drawings for the HSM-H / HSM-HS in Section U.1.5, an optional door has been added to the HSM-H / HSM-HS design. The optional door has an additional 6.875 inches concrete thickness but the steel thickness is reduced to 3 inches.

As noted above, the evaluation of the shielded door includes an additional missile (8" diameter armor piercing artillery shell with a mass of 280 lbs and impact velocity of 508 fps). The controlling missile evaluations require a minimum steel thickness of 2.5 inches. Therefore, a door with 3 inch steel thickness is qualified for missile loads.

U.11.2.3.3 Accident Dose Calculations

The increase in the dose rates at the localized impact location following the missile impact accident is expected to be bounded by the dose rates at the HSM-H vents, calculated to be *600* mrem/hour in Table U.5-1 *(477.17 mrem/hr*1.18 ~ 600 mrem/hr)*, since the structural analysis results demonstrate that there is no full penetration. This represents an increase in the *roof centerline* dose rate by a factor greater than 20 and is conservative.

For the purpose of this calculation, it is conservatively assumed that the affected area is twice the area of impact ~ 1.6 ft^2. The surface area at the HSM-H front is 140 ft^2, at the HSM-H roof is 200 ft^2 and that at the HSM-H side is 280 ft^2. *The impact area, therefore, represents approximately 0.6% to 1.2% of the surface area of the HSM-H, and the average dose rate on the surface of the impacted HSM will not increase appreciably. This increase does not significantly affect the ISFSI site dose rates and the results from Section U.10.2 for a 2x10 array of undamaged HSMs (specifically Table U.10-7) can be utilized to determine the exposure from a damaged HSM. This method is conservative because the missile impact will affect at most a single HSM, while a 2x10 array has approximately 20 front and 20 roof vents.*

The total dose rate is then the dose rate of the damaged HSM summed with the dose rate of the undamaged HSMs in the array, or twice the dose rate of the undamaged array using the conservative assumptions outlined above.

*The dose received by a person located 100 meters away from the ISFSI for the assumed 8-hour duration would be less than 5 mrem (2*8 hours*dose rate at 100m, 8.75E-02 mrem/hour*1.18 scaling factor) with a 2x10 array of HSMs. The dose to an offsite person located 500 meters away for the assumed 8-hour duration would be less than 0.01 mrem (2*8 hours*dose rate at 500m, 1.83E-04 mrem/hour*1.18 scaling factor) with a 2x10 array of HSMs.*

U.11.2.3.4 Corrective Actions

After excessive high winds or a tornado, the HSM-H/HSM-HS and OS200 TC would be inspected for damage. Any debris would be removed. Any damage resulting from impact with a missile would be evaluated to determine if the system was still within the licensed design basis.

U.11.2.4 Flood

This event is described in UFSAR Section 8.2.4.

U.11.2.4.1 Cause of Accident

No change. See UFSAR Section 8.2.4.1.

U.11.2.4.2 Accident Analysis

The HSM-H / HSM-HS and DSCs are evaluated for flooding in Section U.3.7.3. The DSC is designed and tested to be leak tight to the criteria of ANSI N14.5 [11.2]. The stresses in the DSC due to the design basis flood are well below the allowable stresses for Service Level C of the ASME Code Subsection NB [11.5]. Therefore, the NUHOMS® 32PTH1 DSC will withstand the design basis flood without breach of the confinement boundary.

U.11.2.4.3 Accident Dose Calculations

The radiation dose due to flooding of the HSM-H is negligible. The NUHOMS® 32PTH1 DSC is designed and tested as a leak-tight containment boundary. Flooding does not breach the containment boundary. Therefore radioactive material inside the DSC will remain sealed in the DSC and, therefore, will not contaminate the encroaching flood water.

U.11.2.4.4 Corrective Actions

No change. See UFSAR Section 8.2.4.4.

U.11.2.5 Accidental TC Drop

This event is described in UFSAR Section 8.2.5.

U.11.2.5.1 Cause of Accident

See Section U.3.7.4.

U.11.2.7 Blockage of Air Inlet and Outlet Openings

This accident conservatively postulates the complete blockage of the ventilation air inlet and outlet openings of the HSM-H.

U.11.2.7.1 Cause of Accident

No change. See UFSAR Section 8.2.7.1.

U.11.2.7.2 Accident Analysis

The thermal evaluation of this event is presented in Chapter U.4 for HSM-H and the 32PTH1 DSCs. The temperatures determined in Chapter U.4 are used in the structural evaluation of this event, which is presented in Chapter U.3, Sections U.3.7.6 and U.3.4.4.3.

The section below describes the additional analyses performed to demonstrate the acceptability of the system with the NUHOMS® 32PTH1 DSC.

U.11.2.7.3 Accident Dose Calculations

There are no off-site dose consequences as a result of this accident. The only significant dose increase is that related to the recovery operation. Based on the results presented in Chapter U.5, Table U.5-1, the bounding average dose rate on HSM front or roof is 15.5 mrem/hr.

It is conservatively estimated that the on-site workers will receive an additional dose of no more than *150* (=15.5x8 * *1.18 MTU scaling factor*) mrem during the eight hour period it is estimated may be required for removal of debris from the inlet and outlet vent openings. These exposures are well within the limits of 10CFR72.106 for an accident condition.

U.11.2.7.4 Corrective Action

No change to UFSAR Section 8.2.7.4.

U.11.2.8 DSC Leakage

The NUHOMS® 32PTH1 DSC is designed as a pressure retaining containment boundary to prevent leakage of contaminated materials. The analyses of normal, off-normal, and accident conditions have shown that no credible conditions can breach the DSC shell or fail the double seal welds at each end of the DSC. The NUHOMS® 32PTH1 DSC is designed and tested to be leak tight. Therefore DSC leakage is not considered a credible accident scenario. See Chapter U.7 for additional details on the confinement evaluation.

U.11.2.9 Accident Pressurization of DSC

U.11.2.9.1 Cause of Accident

The bounding internal pressurization of the NUHOMS® 32PTH1 DSC is postulated to result from cladding failure of the spent fuel in combination with the transfer accident case with the loss of sunshield and liquid neutron shield in the transfer cask under extreme ambient

V.1.1 Introduction

This appendix adds the HSM Model 202 to the NUHOMS® system. Only those features that are being revised or added to the NUHOMS® system are addressed and evaluated in this appendix. Sections of this appendix which are not affected by the addition of the HSM Model 202 are indicated in this appendix with "No Change." The various DSCs and Transfer Cask (TC) in the Standardized NUHOMS® system remain unchanged.

The HSM Model 202 is based on the HSM-H. The HSM Model 202 is qualified to store payloads with a maximum decay heat load of 24.0 kW, similar to the Standardized HSM Models 80 and Model 102. The HSM Model 202 is being added to the UFSAR as an alternative to the HSM Model 80/102/152. The primary reason

for adding the HSM Model 202 is to incorporate an HSM design that offers even greater biological shielding compared to Models 80 and 102, and greater heat rejection capabilities than those currently available in Models 80/102/152.

The HSM Model 80 and Model 102 were the original HSM designs included in the UFSAR. Thus, the addition of HSM Model 202 is being accomplished by qualifying it in comparison to HSM Model 80 and HSM Model 102.

This evaluation focuses solely on comparing HSM Model 202 to HSM Model 80 and HSM Model 102. The major differences between HSM Model 202 and the existing HSM Model 80 and HSM Model 102 are listed in *Table V.1-1* and highlighted in Section P.1.2.1.2. Other than a general increase in the overall height of the module to minimize air flow resistance, selected differences in the wall and roof thicknesses, elimination of the gap between adjacent modules, and a change in the vent configuration to facilitate decay heat removal, there are no other significant differences between these three HSM designs -- all of which are passively cooled. Moreover, all three HSM designs still consist of massive reinforced concrete structures that are capable of withstanding all normal condition loads as well as the abnormal condition loads created by earthquakes, tornadoes, flooding, and other natural phenomena hazards, and they still remove decay heat by natural circulation convection and by conduction through the HSM walls and roof.

V.1.2 General Description of the NUHOMS® HSM Model 202

V.1.2.1 NUHOMS® HSM Model 202 Characteristics

An isometric view of the prefabricated NUHOMS® HSM Model 202 is shown in Figure V.1-1. A drawing of the HSM Model 202 is presented in Section V.1.5. Nominal dimensions and the empty weight of the HSM Model 202 are shown in Table V.1-1. The material, geometry and dimensions of the Model 202 are based on the NUHOMS® HSM-H design described in Appendix P.

V.1.2.2 Operational Features

V.1.2.2.1 General Features

The HSM Model 202 is designed to safely store a DSC with a maximum weight up to 110 kips which bounds the weight of the 24P, 52B, 61BT, *61BTH Type 1*, 32PT, 24PHB and 24PTH-S-LC DSCs. The HSM Model 202 protects the DSC from the potentially adverse effects of natural phenomena hazards, such as earthquake, tornado, tornado missiles, and flood. In addition, the HSM Model 202 dissipates decay heat from the spent fuel by a combination of radiation, conduction, and convection. Natural convection air flow enters the bottom side walls of the HSM, circulates around the DSC, and exits through the flow channels along the top side walls of the HSM. The cross-sectional areas of the air inlet and outlet openings, and the interior flow paths are designed to optimize ventilation air flow for decay heat removal. Furthermore, like the HSM Models 80/102/152, a thermal radiation heat shield is used in the Model 202 to reduce the HSM concrete temperatures to acceptable limits for all thermal conditions.

Y.2.1 Spent Fuel to be Stored

Y.2.1.1 Intact or Damaged Fuel

As described in Appendix Y.1, the NUHOMS®-69BTH DSC is designed to store intact (including reconstituted) and/or damaged (boiling water reactor) BWR fuel assemblies as specified in Table Y.2-1 and Table Y.2-2. The fuel to be stored is limited to a maximum lattice average initial enrichment of 5.0 wt. % U-235. The maximum allowable fuel assembly average burnup is limited to 62 GWd/MTU. The minimum required cooling time for fuel to be stored with 170 kgU/FA and 198 kgU/FA is explicitly specified as a function of burnup and enrichment in Tables Y.2-5 through Y.2-17b. For fuel with a kgU/FA loading between these two values, the minimum required cooling time for fuel to be stored as a function of burnup and enrichment is determined by using the instructions provided in the the notes and examples following Table Y.2-16.

The NUHOMS®-69BTH DSC is also authorized to store fuel assemblies containing blended low enriched uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity affects only the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Reconstituted fuel assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or 69 lower enrichment UO_2 rods instead of zircaloy clad enriched UO_2 rods are acceptable for storage in 69BTH DSCs as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel rods or 69 with UO_2 rods or Zr rods or Zr pellets or unirradiated stainless steel rods.

The NUHOMS®-69BTH DSCs can also accommodate up to a maximum of 24 damaged fuel assemblies placed in the four outer "six compartment" arrays located at the outer edge of the DSC as shown in Figure Y.2-7. Damaged BWR fuel assemblies are assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that the fuel assembly, *including non-cladding damage,* is able to be handled by normal means *and the retrievability is assured following the normal and off-normal conditions.* Missing fuel rods are allowed. *The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is ensured following normal and off-normal conditions.* The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps.

A 69BTH DSC containing less than 69 fuel assemblies may contain dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, aluminum blocks that approximate the weight and center of gravity of a fuel assembly.

Table Y.2-1

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-1gg.

Table Y.2-5

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7a.

Table Y.2-6

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7a1.

Table Y.2-7

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7b.

Table Y.2-8

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7c.

The change on this page is a result of the issuance of Amendment 15.

Table Y.2-9

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7d.

Table Y.2-10

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7e.

Table Y.2-11

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7f.

Table Y.2-12

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7g.

Table Y.2-13

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7h.

The change on this page is a result of the issuance of Amendment 15.

Table Y.2-14

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7i.

Table Y.2-15

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7j.

Table Y.2-16

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7k.

The explanatory notes associated with Tables Y.2-5 through Y.2-17b can be found in CoC 1004Amendment *15* Technical Specifications below Table 1-7m.

Table Y.2-17a

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7l.

Table Y.2-17b

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Table 1-7m.

The change on this page is a result of the issuance of Amendment 15.

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications Figure 1-34.

Figure Y.2-4

The detailed information associated with this figure can be found in CoC 1004 Amendment *15* Technical Specifications Figure 1-38.

Figure Y.2-9

Table Y.3.1-2

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, ASME Code Alternatives for the NUHOMS-69BTH DSC Confinement Boundary.

Table Y.3.1-3

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, ASME Code Alternatives for the NUHOMS-69BTH DSC Basket.

Proprietary Information on This Page
Withheld Pursuant to 10 CFR 2.390

Y.5.4.10 Impact on Dose Rates due to Reduced Density Concrete and Gaps between HSMs

A bounding analysis is performed by employing a minimum concrete density of 140 pounds per cubic foot (pcf) in the HSM-H MCNP model combined with a maximum gap of 1.5 inches between adjacent HSM-H modules and shield walls to determine the effect on maximum and average dose rates due to a fully loaded 32PTH1 DSC. These calculations are documented in Appendix U.5, Section U.5.4.10. The ratios shown in Appendix U.5, Table U.5-*18* and Table U.5-*19* can be used as scaling factors to increase the maximum and surface-average dose rates of the 69BTH in the HSM-H to account for low density concrete and 1.5" gaps. Note that the HSM-H concrete contains high density rebar which is not credited in the MCNP models. Further, the modules are installed adjacent to each other such that there will not be a "uniform" gap of 1.5 inches. Ignoring the effect due to increased vent dose rates, the increase in the average dose rates caused by both the maximum postulated uniform gaps and the minimum postulated concrete density is expected to be less than 20% at the front and roof surfaces of the HSM-H module. Dose reduction hardware may be installed to further reduce these dose rates. 72.48

Table Y.5-1
Summary of NUHOMS®-69BTH Bounding DSC (with DB Fuel Loaded as in Bounding Configuration) in HSM-HS, Bounding Maximum and Average Dose Rates

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total[1, 3] (mrem/hr)	Total MCNP 1σ Error
HSM-HS roof bird screen	113.66	0.08	14.74	0.03	**128.40**	0.07
HSM-HS roof (centerline)	13.49	0.18	1.68	0.06	**15.17**	0.16
HSM-HS end (side) shield wall surface	1.48	0.25	0.14	0.06	**1.62**	0.23
HSM-HS door exterior surface (centerline)	0.46	0.36	0.05	0.19	**0.51**	0.32
HSM door exterior surface 1 m (centerline)	0.91	0.31	0.20	0.24	**1.12**	0.25
HSM-HS front bird screen	215.83	0.008	10.07	0.02	**225.90**	0.01

Dose Rate Location	Gamma Average (mrem/hr)	Gamma MCNP 1σ Error	Average Neutron (mrem/hr)	Neutron MCNP 1σ Error	Average Total[3] (mrem/hr)	Total MCNP 1σ Error
HSM-HS roof	10.62	0.04	1.38	0.024	**12.01**	0.04
HSM-HS end (side) shield wall surface	0.38	0.02	0.05	0.015	**0.42**	0.02
HSM-HS front	8.26	0.09	0.60	0.04	**8.86**	0.08
HSM-HS back shield wall	0.04	0.09	0.008	0.03	**0.05**	0.07

Note: Dose is calculated using 69BTH bounding, from the shielding performance stand point, DSC.

(1) Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.
(2) Dose is calculated using 69BTH bounding, from the shielding performance stand point, DSC.
(3) Use the ratios shown in Appendix U.5, Table U.5-*18* and Table U.5-*19* to increase the maximum and surface-average dose rates respectively to account for reduced density concrete and gaps of up to 1.5" as described in Appendix U.5, Section U.5.4.10. 72.48

Y.9 Acceptance Tests and Maintenance Program

Background for this particular UFSAR chapter:

Beginning with CoC 1004 Amendment 13, which was incorporated into UFSAR Revision 14, Chapter Y.9, "Acceptance Tests and Maintenance Program," contained information which was incorporated by reference into the Technical Specifications (TS) associated with a particular amendment. It is known that certain general licensees reconcile the CoC 1004 UFSAR revisions provided to them to their loaded systems, pursuant to 10 CFR 72.48 and 10 CFR 72.212. In doing so they sometimes find the changed UFSAR portions incorporated by reference into the TS to be impossible to reconcile because the 10 CFR 72.48 regulation does not allow proposed activities which involve changes to the TS.

In order to facilitate this reconciliation process by general licensees, the following statements are provided, addressing the licensing basis for certain amendments, as they relate to certain UFSAR chapters which contain TS incorporated by reference. Additionally, so that the actual information is contained in the current CoC 1004 UFSAR, to facilitate the reconciliation by general licensees, the UFSAR Revision 14 and 15 versions of Chapter Y.9 are inserted and annotated in this part of the UFSAR. For clarity, this includes annotating the version of Chapter Y.9 directly associated with the latest UFSAR revision in which a change to Chapter Y.9 occurred.

- Systems loaded to CoC 1004 Amendment 13 have Technical Specifications incorporated by reference from UFSAR Revisions 14 and 15 Chapter Y.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 13.
- Systems loaded to CoC 1004 Amendment 14 have Technical Specifications incorporated by reference by FCN 721004-1575, which will be incorporated into UFSAR Revisions 16 and 17 Chapter Y.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 14.
- *Systems loaded to CoC 1004 Amendment 15 have Technical Specifications incorporated by reference from UFSAR Revision 18 Chapter Y.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 15.*

Y.9 Acceptance Tests and Maintenance Program

Y.9.1 Acceptance Tests

The pre-operational testing requirements for the Standardized NUHOMS® system are given in Chapter 9, with the exceptions described in the following sections. The NUHOMS®-69BTH DSC has been enhanced to provide leaktight confinement and the basket includes an updated poison plate design. Additional acceptance testing of the NUHOMS®-69BTH DSC welds and of the poison plates are described.

Y.9.1.1 Visual Inspection

Visual inspections are performed at the fabricator's facility to ensure that the DSC, the Transfer Cask and the HSM conform to the drawings and specifications. The visual inspections include weld, dimensional, surface finish, and cleanliness inspections. Visual inspections specified by codes applicable to a component are performed in accordance with the requirements and acceptance criteria of those codes.

All weld inspection is performed using qualified processes and qualified personnel according to the applicable code requirements, e.g., ASME or AWS. Non-destructive examination (NDE) requirements for welds are specified on the drawings provided in Appendix Y.1; acceptance criteria are as specified by the governing code. NDE personnel are qualified in accordance with SNT-TC-1A [9.2].

The confinement welds on the DSC are inspected in accordance with ASME B&PV Code Subsection NB [9.1] including alternatives to ASME Code specified in Section Y.3.1.2.3.

DSC non-confinement welds are inspected to the NDE acceptance criteria of ASME B&PV Code Subsection NG or NF, based on the applicable code for the components welded.

Y.9.1.2 Structural

The DSC confinement boundary except the inner top cover/shield plug to the DSC shell weld is pressure tested at the fabricator's shop in accordance with ASME Article NB-6300. The test pressure is set between 16.5 to 18.0 psig for the 69BTH DSC, which bounds the 1.1xDSC design pressure of 15 psig.

The inner top cover/shield plug to the DSC shell weld is also pressure tested between 16.5 to 18.0 psig for 69BTH DSC. This pressure test is performed at the field after the fuel assemblies are loaded in the DSC. This test is in accordance with the alternatives to the ASME Code specified in Section Y.3.1.2.3.

HSM-H reinforcement and concrete are tested as described in Chapter 3, Section 3.4.2.

Y.9.1.3 Leakage Tests

The DSC canister confinement boundary is tested using two procedures described below. Personnel performing the leakage test are qualified in accordance with SNT-TC-1A [9.2].

Procedure 1 is accomplished during fabrication:

Upon completion of all canister shell welding and attachment of the inner bottom cover plate to the shell, a temporary seal plate is placed over the open end of the DSC. A bag or other enclosure is placed around the outside of the entire DSC and it is filled with helium. The DSC cavity is evacuated and a helium leakage test is performed using a port in the seal plate. This test is used to show that the entire DSC confinement boundary tested is leak tight (1×10^{-7} ref cm^3/s).

Procedure 2 of the testing occurs after the DSC has been loaded with fuel assemblies:

The DSC cavity has been dried, back filled with helium and the inner top cover plate and the vent and drain port cover plates have been welded in place. After these welds are completed, a temporary test cover is installed or the outer top cover plate is welded in place with at least the root pass of the full weld. The cavity between inner top cover plate and the temporary test cover or outer top cover plate is evacuated and a helium leakage test is performed using a test port in the temporary test cover or in the outer top cover plate. The leakage test thus includes the weld attaching the inner top cover plate to the canister shell, the vent and drain port cover plate welds and the base metal of the inner top cover plate and vent and drain port cover plates. The vent and drain ports are filled with helium prior to welding the vent and drain port covers. This test verifies that the tested welds and cover plates are leak tight (1×10^{-7} ref cm^3/s).

Y.9.1.4 Components

The Standardized NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. No other components of the NUHOMS® system require testing, except as discussed in this chapter.

Y.9.1.5 Shielding Integrity

The Transfer Cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6″ × 6″ grid, the detector will encompass a 6″ × 6″ square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed. The gamma and neutron

shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

Y.9.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section Y.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section Y.9.1.7.6.

Y.9.1.7 Poison Acceptance

CAUTION

Sections Y.9.1.7.1 through Y.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 6) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated aluminum

(b) Boron carbide/aluminum metal matrix composite (MMC)

(c) BORAL®

The 69BTH DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content of these three types of materials is given in Table Y.9-1.

References to metal matrix composites throughout this chapter are not intended to refer to BORAL®, which is described later in this section.

Y.9.1.7.1 **Borated Aluminum**

See the Caution in Section Y.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating primarily as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy (other boron compounds, such as AlB_{12}, can also occur). For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section Y.9.1.7.7. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Y.9.1.7.2 **Boron Carbide / Aluminum Metal Matrix Composites (MMCs)**

See the Caution in Section Y.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, molten metal infiltration, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding or produced by molten metal infiltration shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B_4C particles in boron carbide shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 69BTH DSC, MMCs shall pass the qualification testing specified in Section Y.9.1.7.8, and shall subsequently be subject to the process controls specified in Section Y.9.1.7.9.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section Y.9.1.7.7. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Y.9.1.7.3 **BORAL®**

See the Caution in Section Y.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns. The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of BORAL®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken from the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

Y.9.1.7.4 **Visual Inspections of Neutron Absorbers**

See the Caution in Section Y.9.1.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. Blisters shall be treated as non-conforming. For clad MMCs and for BORAL®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet. Material that does not meet these criteria shall be reworked, repaired, or scrapped.

Y.9.1.7.5 Other Visual Inspections Criteria (non-Technical Specifications)

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

Y.9.1.7.6 Thermal Conductivity Testing

Acceptance testing shall conform to ASTM E1225[1], ASTM E1461[2], or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, and may be reduced if the first five tests meet the specified minimum thermal conductivity. For cast products, the lot shall be defined by the heat or ingot. For other products, the lot shall be defined as material produced in a single production campaign using the same heat or lots of aluminum and boron carbide feed materials.

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1 ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"

2 ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The measured thermal conductivity values shall satisfy the minimum required conductivities as specified in Section Y.4.3

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section Y.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

Y.9.1.7.7 Specification for Acceptance Testing of Neutron Absorber *Content*

Acceptance testing for neutron absorber content shall be performed by either neutron transmission or by B-10 volume density measurement.

CAUTION

Portions of Section Y.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 6) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Y.9.1.7.7.1 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

a) Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam up to 1.1 inch diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard. Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than the statistically derived minimum thickness from Y.9.1.7.7a) or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness.

Non-conforming material shall be evaluated for acceptance in accordance with the

Certificate Holder's QA procedures.

Y.9.1.7.7.2 Specification for Acceptance Testing of Neutron Absorbers by B-10 Volume Density Measurement

a) B-10 volume density measurement acceptance testing procedures shall be subject to approval by the certificate holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, as long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for B-10 volume density measurements shall be such that there is at least one density measurement for each 2000 square inches of final product in each lot.

Areal density is determined by measuring the B-10 volume density in test samples and converting the measured values to areal density. The method of measurement of B-10 volume density shall be subject to approval by the certificate holder. The method of measurement of B-10 volume density shall be qualified against neutron transmission testing. Results of the two test methods shall be compared and a penalty shall be derived to account for the performance based results of neutron transmission testing.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B-10 areal densities are determined by volume density as described above. The lower tolerance limit of B-10 volume density is then determined, defined as the mean value of B-10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6]. Finally, the minimum specified value of B-10 areal density is divided by the lower tolerance limit of B-10 volume density to arrive at the minimum plate thickness that provides the specified B-10 areal density.

Any plate that is thinner than the statistically derived minimum thickness from Y.9.1.7.7.2 a) or the minimum design thickness, whichever is greater, shall be treated as nonconforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than

90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

Y.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

> ***CAUTION***
>
> *Portions of Section Y.9.1.7.8.4, and all of Section Y.9.1.7.8.5, are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 6) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

Y.9.1.7.8.1 Applicability and Scope

MMCs acceptable for use in the 69BTH DSC are described in Section Y.9.1.7.2.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section Y.9.1.7.9 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system. ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the Certificate Holder.

Y.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in Section Y.9.1.7.8.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section Y.9.1.7.8.5.

Y.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

The need for thermal damage and corrosion (hydrogen generation) testing shall be evaluated case-by-case based on comparison of the material composition and environmental conditions with previous thermal or corrosion testing of MMCs. Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for unclad MMCs consisting only of boron carbide in an

aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842 °F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for full density MMCs (clad or unclad) consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

Y.9.1.7.8.4 **Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity**

At least three samples, one each from approximately the two ends and middle of the qualification material run shall be subject to:

a) **room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:**

- **Minimum yield strength, 0.2% offset: 1.5 ksi**
- **Minimum ultimate strength: 5 ksi**
- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

b) **Testing to verify more than 98% of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a means to be approved by the Certificate Holder. The maximum interconnected porosity is 0.5 volume %.**

c) **Delamination Testing of Clad MMC**

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage. An example of such a test would be: (1) immerse a specimen at least 6 x 6 inches in water under pressure ≥30 psig for at least 24 hours, (2) place the specimen in a vacuum furnace preheated to at least 300 °F and evacuate the furnace. Acceptance criterion: no blistering or delamination of the cladding.

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3 Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998.

4 Boralyn testing submitted to the NRC under docket 71-1027, 1998.

5 ASTM B557, Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.

6 ASTM E290, Standard Methods for Bend Testing of Materials for Ductility

Y.9.1.7.8.5 Required Tests and Examinations to Demonstrate B10 Uniformity

Uniformity of the boron distribution shall be verified either by:

a) Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, *or the boron weight fraction,* on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section Y.9.1.7.7, or by chemical analysis for boron carbide *or boron* content in the composite.

Y.9.1.7.8.6 Qualification Report

Qualification report shall be prepared by, or subject to approval by the Certificate Holder.

Y.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION

Sections Y.9.1.7.9.1 and Y.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 6) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Y.9.1.7.9.1 **Applicability and Scope**

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section Y.9.1.7.8 is required, depending on the characteristics of the material that could be affected by the process change.

Y.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, reduce corrosion resistance, reduce the mechanical strength or ductility of the MMC.

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7 ASTM E94, Recommended Practice for Radiographic Testing

8 ASTM E142, Controlling Quality of Radiographic Testing

9 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

Y.9.1.7.9.3 Identification and Control of Key Process Changes

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section Y.9.1.7.9.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that are established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average (d50) particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a magnesium-alloyed aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, and

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

Y.9.2 Maintenance Program

The NUHOMS®-69BTH system is a totally passive system and therefore will require little, if any, maintenance over the lifetime of the ISFSI. Typical NUHOMS®-69BTH system maintenance tasks will be performed in accordance with the UFSAR.

Y.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, 2004 Edition through 2006 Addenda.

9.2 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

9.3 Not Used.

9.4 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.5 "Aluminum Standards and Data, 2003," The Aluminum Association.

9.6 Natrella, "Experimental Statistics," Dover, 2005.

Table Y.9-1
B10 Specification for the NUHOMS®-69BTH Poison Plates

Poison Type	Basket Type	Specified Minimum B10 Areal Density for 90% Credit (g/cm^2)	% Credit Used in Criticality Analysis
Borated Aluminum Alloy / MMC	A	0.021	90
	B	0.031	
	C	0.039	
	D	0.046	
	E	0.053	
	F	0.061	
BORAL®	A	0.025	75
	B	0.037	
	C	0.047	
	D	0.055	
	E	0.064	
	F	0.073	

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Z.1.1 Introduction

The NUHOMS®-37PTH system is designed to store up to 37 intact (including reconstituted) PWR fuel assemblies with uranium dioxide (UO_2). The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt% U-235, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of *2.0* years. Each of the 37PTH DSC types is designed to store up to 37 control components (CCs) which include control spiders, burnable poison rod assemblies (BPRAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), integral fuel burnable absorber (IFBA) assemblies, and neutron source assemblies (NSAs) and neutron sources. The design characteristics, including physical and radiological parameters of the payload, are described in Appendix Z.2.

Reconstituted fuel assemblies containing up to 10 replacement rods (irradiated stainless steel rods) per assembly or unlimited lower enrichment UO_2 rods instead of zircaloy clad enriched UO_2 rods are acceptable for storage in 37PTH DSCs as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel rods or 37 with UO_2 rods or Zr rods or Zr pellets or unirradiated stainless steel rods.

Provisions have been made for storage of up to 4 damaged fuel assemblies in lieu of an equal number of intact assemblies in the cells located at the periphery of the 37PTH basket as described in Appendix Z.2, Figures Z.2-1 through Z.2-3. Damaged PWR fuel assemblies are defined in Appendix Z.2.

The NUHOMS®-37PTH system consists of the following components:

- A 37PTH DSC, with two alternate configurations, described in detail in Section Z.1.2, provides confinement, an inert environment, structural support, and criticality control for the 37 PWR fuel assemblies.
- A modified HSM-H/HSM-HS module, described in Section Z.1.2, provides environmental protection, shielding, and heat rejection during storage.
- An OS200 or OS200FC TC, described in Appendix U.1, Section U.1.2.1.3, provides for onsite transfer of the 37PTH DSCs.

The NUHOMS®-37PTH system requires the use of non-safety related auxiliary transfer equipment similar to those described in Chapter 1, Section 1.3.2.2 and Appendix U.1, Section U.1.2.1.3 (when using the air circulation feature of the OS200FC TC). There is no change to any of the design features of the auxiliary transfer equipment.

Approval of the NUHOMS®-37PTH system components described in Section Z.1.2 is sought under the provisions of 10 CFR 72, subpart L for use under the general license provisions of 10 CFR 72, subpart K. The 37PTH system components are intended for storage on a reinforced concrete pad.

Z.2.1 Spent Fuel to be Stored

Z.2.1.1 Intact or Damaged Fuel

As described in Appendix Z.1, there are two alternate design configurations for the NUHOMS®-37PTH DSC depending on the canister length: a short (182.00 in.) DSC designated as 37PTH-S, and a medium (189.25 in.) DSC designated as 37PTH-M. Each of the DSC configurations is designed to store intact (including reconstituted) and/or damaged PWR fuel assemblies as specified in Table Z.2-1 and Table Z.2-3. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235. The maximum allowable assembly average burnup is limited to 62 GWd/MTU. The minimum required cooling time for fuel to be stored with *380, 475, and 492* kgU/FA is explicitly specified as a function of burnup and enrichment in *Technical Specifications Table 1-3a through 1-3p.* For fuel with a kgU/FA loading between these values, the minimum required cooling time for fuel to be stored as a function of burnup and enrichment is determined by using the instructions provided in the notes and examples following *Technical Specifications Table 1-3p.*

Each of the DSC types is designed to store control components (CCs) with thermal and radiological characteristics as listed in Table Z.2-2. The CCs include burnable poison rod assemblies (BPRAs), neutron source assemblies (NSAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), and neutron sources. Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as guide tube or instrument tube tie rods or anchors, guide tube inserts, BPRA spacer plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.

The NUHOMS®-37PTH DSC is also authorized to store fuel assemblies containing blended low enriched uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity affects only the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

The WE 17x17 class fuel assembly is authorized for loading at soluble boron concentrations in the range 2000 to 2600 ppm provided that PRAs are also used as an additional criticality control mechanism. Both intact and damaged fuel assemblies are authorized for storage with and without CCs in a five-assembly or nine-assembly PRA configuration, as shown in Figure Z.2-5 and Figure Z.2-6, respectively, with each of the five or nine assemblies containing eight poison rods. Table Z.2-5a provides the minimum required B_4C content per rod.

Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiation and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding and criticality evaluations for these rods are bounded by those of the regular rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Reconstituted assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or unlimited lower enrichment UO_2 rods instead of zircaloy clad enriched UO_2 rods, or Zr rods or Zr pellets, or unirradiated stainless steel rods are acceptable for storage in the 37PTH DSC as intact fuel assemblies.

The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel replacement rods or 37 with UO_2 replacement rods.

The NUHOMS®-37PTH DSCs can also accommodate up to a maximum of four damaged fuel assemblies placed in the outer cells of the DSC as shown in Figure Z.2-2 and Figure Z.2-3. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater than hairline cracks, or pinhole leaks. The extent of damage in the fuel assembly, *including non-cladding damage,* is to be limited such that a fuel assembly will still be able to be handled by normal means *and the retrievability is ensured following the normal and off-normal conditions.* Missing fuel rods are allowed. *The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is ensured following normal and off-normal conditions.* The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps.

A 37PTH DSC containing less than 37 fuel assemblies may contain dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly.

The 37PTH DSC basket is designed with solid aluminum transition rails for support and to facilitate heat removal, since the solid aluminum rails allow a more direct heat conduction path from the basket edge to the DSC shell.

The NUHOMS®-37PTH DSCs may store up to 37 PWR fuel assemblies arranged in any of the two alternate heat load zoning configurations (HLZC) as shown in Figure Z.2-2 and Figure Z.2-3. The maximum decay heat per fuel assembly and the maximum canister heat load allowed is as specified in Figure Z.2-2 and Figure Z.2-3. The maximum allowed heat load for the various 37PTH system configurations are presented in Table Z.2-17.

The NUHOMS®-37PTH DSC configuration is analyzed for three alternate DSC basket designs for criticality control. In Option 1, the poison plate configuration consists of a 0.05-in. nominal thickness aluminum plate in combination with a 0.075 in. nominal thickness neutron absorber plate. Option 2 and Option 3 consist of a single neutron absorber plate of 0.125-in.and 0.105-in. nominal thickness, respectively. Option 1 results in the most reactive configuration.

In addition, the NUHOMS®-37PTH DSC basket is provided with three alternate neutron absorber plate materials (poison material) for criticality control: borated aluminum alloy, boron carbide/aluminum metal matrix composite (MMC) and Boral®. For criticality analysis, 90% of B10 content present in the borated aluminum and MMC poison plates is credited, while only 75% is credited for Boral®. The minimum B-10 poison loadings allowed are presented in Table Z.2-16. The use of Boral is restricted to Option 1 DSC basket designs only.

The selection of the poison material does not have any impact on the thermal analysis, since it is based on the limiting thermal conductivity of Boral® as discussed in Appendix Z.4, Section Z.4.3.

Table Z.2-4 summarizes the maximum assembly average initial enrichment as a function of soluble boron concentration for intact and damaged fuel. *Technical Specification Tables 1-3a through 1-3p* along with the explanatory notes that accompany the tables

define the minimum required cooling time after reactor discharge for a fuel assembly with or without CCs for a given assembly heat load, burnup, and maximum assembly average initial enrichment parameters. These tables ensure that the fuel assembly decay heat load is less than that specified for each table and that the corresponding radiation source term is bounded by that analyzed in Appendix Z.5.

The NUHOMS®-37PTH DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight with CCs that can be accommodated is 1665 lbs in the 37PTH DSC.

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Z.2.1.3 Thermal and Related Design Criteria

The maximum fuel cladding temperature limit of 400 °C (752 °F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-37PTH DSC per NUREG 1536 [2.1]. In addition, [2.1] does not permit repeated thermal cycling of the fuel cladding (limited to less than 10 cycles) with cladding temperature differences greater than 65°C (117°F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570 °C (1058 °F) is applicable to accidents or off-normal storage thermal transients [2.1].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, thermal and confinement. These evaluations are performed in Appendix Z.5, Z.6, Z.4 and Z.7 respectively. The fuel assembly classes considered are listed in Table Z.2-3. Although B&W 15x15 fuel assembly is not authorized for storage in the 37PTH DSC, this fuel assembly design *may be used as a representative fuel assembly for dose rate calculations*. For criticality safety, the WE 17x17 fuel assembly is the most reactive assembly type for a given enrichment. This assembly is used to determine the most reactive configuration in the DSC. Using this most reactive configuration, criticality analysis for all other fuel assembly classes is performed to determine the maximum enrichment allowed as a function of the soluble boron concentration and fixed poison plate loading. For thermal analysis, the WE 14x14 fuel assembly is limiting for the 37PTH DSCs, since it results in the lowest effective fuel thermal conductivity. The confinement analysis, similar to the shielding analysis, is conservatively based on the B&W 15x15 fuel assembly, since it results in a smaller free volume inside the DSC cavity as compared to a 14x14 fuel assembly.

For calculating the maximum internal pressure in the NUHOMS®-37PTH DSC, it is assumed that 1% of the fuel rods are damaged for normal conditions, up to 10% of the fuel rods are damaged for off-normal conditions, and 100% of the fuel rods will be damaged following a design basis accident event [2.1]. A minimum of 100% of the fill gas and 30% of the fission gases within the ruptured fuel rods are assumed to be available for release into the DSC cavity, consistent with NUREG-1536 [2.1].

Table Z.2-1

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-1ll.

Table Z.2-2

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-1qq.

Table Z.2-3

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-1nn.

Table Z.2-4a

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-1pp.

The change on this page is a result of the issuance of Amendment 15.

Table Z.2-6
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Table Z.2-16

The detailed information associated with this table can be found in CoC 1004 *Amendment 15* Technical Specifications Table 1-1rr.

The detailed information associated with this figure can be found in CoC 1004 *Amendment 15* Technical Specifications Figure 1-41.

Figure Z.2-5

The detailed information associated with this figure can be found in CoC 1004 *Amendment 15* Technical Specifications Figure 1-42.

Figure Z.2-6

Table Z.3.1-1

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, Alternatives to the ASME Code for the NUHOMS-37PTH DSC Confinement Boundary.

Table Z.3.1-2

The detailed information associated with this table can be found in CoC 1004 Amendment *15* Technical Specifications, Alternatives to the ASME Code for the NUHOMS-37PTH DSC Basket Assembly.

Z.3.6.2.2 Off-Normal Thermal Loads Analysis

As described in UFSAR Section 8.1.2.2, the NUHOMS® system is designed for use at all reactor sites within the continental United States. Therefore, off-normal ambient temperatures of -40°F (extreme winter) and 117°F (extreme summer) are conservatively chosen. Each licensee should verify that this range of ambient temperatures envelops the design basis ambient temperatures for 72.48
the ISFSI site. The NUHOMS® system components affected by the postulated extreme ambient temperatures are the TC and DSC during transfer from the plant's fuel/reactor building to the ISFSI site, and the HSM-H/HSM-HS during storage of a DSC.

Appendix Z.4 provides the off-normal thermal analyses for storage and transfer modes for the NUHOMS®-37PTH DSC. Maximum DSC shell assembly thermal stress analysis results for the normal and off-normal conditions are summarized in Table Z.3.6-2. Basket assembly thermal results are summarized in Table Z.3.6-3a and Table Z.3.6-4a. The resulting stress intensities for the NUHOMS®-37PTH DSC are acceptable.

The off-normal stress analysis results and thermal loading for the HSM-H/HSM-HS are presented in Appendix U, Sections U.3.6.2.3 and U.3.6.2.4, respectively, and are applicable to the HSM-H/HSM-HS loaded with the 37PTH DSC.

Z.3.6.3 Damaged Fuel Cladding Structural Evaluation for Normal and Off-Normal Loads

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The structural analysis documented in Section U.3.6.3 conservatively evaluates a limiting configuration with a single rod and the spacer grids in designated locations without any support from the fuel compartment to provide assurance of limiting additional cladding damage. The changes to the fuel assembly configuration do not have impact on retrievability due to damage to spacer grids, as long as the assembly is able to be handled by normal means and the retrievability is ensured following the normal and off-normal conditions. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to ensure retrievability. The criticality analysis documented in Section Z.6.4 also considers the impact of damage to the fuel assembly that includes missing and damaged grid spacers, which results in limiting the enrichment of these fuel assemblies. Therefore, additional configurations are not evaluated herein. Licensees can perform specific evaluations to demonstrate retrievability using actual configurations.

Z.5 Shielding Evaluation

The radiation shielding evaluation for the standardized NUHOMS® System (during loading, transfer, and storage) for the other NUHOMS® canisters is discussed in other sections and appendices of the UFSAR. The following radiation shielding evaluation specifically addresses the shielding performance of the NUHOMS®-37PTH System with design-basis PWR fuel assembly (FA) containing UO_2 fuel and control components (CCs) loaded in a NUHOMS®-37PTH DSC.

The radiation shielding evaluation described below is for the NUHOMS®-37PTH DSC transferred in a NUHOMS® OS200/OS200 FC transfer cask (TC) and stored in an HSM-H/HSM-HS module. There are two alternate configurations depending on the canister length applicable to the 37PTH system: (1) 37PTH-S, and (2) 37PTH-M. Each DSC has a different length. The basket layout for these configurations is identical except for the length of the compartments. Note that HSM and HSM-H (or HSM-HS), as well as OS200 and OS200 FC, are used interchangeably throughout this appendix. Also, 37PTH, 37PTH-S, and 37PTH-M nomenclatures are used interchangeably. For each 37PTH DSC there are two heat load zoning configurations (HLZCs) described in Appendix Z.2, Figures Z.2-2 and Figure Z.2-3. The OS200 TC is the same as described in Appendix U.

Each DSC configuration is designed to store up to 37 intact (and up to 4 damaged in the corner fuel compartments, with remaining intact) PWR fuel assemblies with or without CCs. The authorized CCs include burnable poison rod assemblies (BPRAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), thimble plug assemblies (TPAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), neutron sources, and neutron source assemblies (NSAs). Furthermore, non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as guide tube or instrument tube tie rods or anchors, guide tube inserts, BPRA spacer plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.

The design-basis PWR fuel, 0.490 MTU of heavy metal weight, source terms are derived from the bounding fuel assembly design (B&W 15x15 Mark B assembly design), as described in Section Z.5.2; however, B&W 15x15 Mark B assemblies are not authorized for storage or transfer in the 37PTH DSC. The SAS2H\ORIGEN-S depletion model of these assemblies is utilized because it results in radiological sources that are bounding for all PWR assemblies described in Appendix Z.2.

The NUHOMS®-37PTH DSCs are designed to store PWR fuel assemblies with or without CCs with the characteristic sources for CCs described in Table Z.5-13. The 37PTH DSCs have a maximum decay heat of 1.2 kW per assembly and a maximum heat load of 30.0 kW per canister. Note that while the specific fuel designs are listed in Appendix Z.2, storing reload fuel designed by other manufacturers is also allowed provided an analysis is performed to demonstrate that the limiting features listed in Appendix Z.2 bound the specific manufacturer's replacement fuel. The limiting parameters are the design basis radiological and decay heat source terms.

The design-basis fuel source terms, based on 0.490 MTU of heavy metal weight, for this evaluation are defined as the source terms from fuel with the burnup/initial enrichment/cooling time combination that *result in* the maximum dose rate on the surface of the HSM and/or TC. This approach is consistent with the method used to generate the fuel qualification tables for the Standardized NUHOMS®-24P and -52B DSC designs as described in Chapter 7, Section 7.2.3, 32PT DSC design as described in Appendix M, NUHOMS®-24PTH DSC design described in Appendix P, or NUHOMS®-32PTH DSC design described in Appendix U. The design basis fuel source terms in conjunction with the design basis CC source terms (Table Z.5-13) are used to calculate dose rates for the NUHOMS®-37PTH System.

The original 0.490 MTU fuel qualification tables (FQTs) developed for the 32PTH1 DSC using SCALE4.4/SAS2H have been replaced with FQTs for 0.380 MTU, 0.475 MTU, and 0.492 MTU developed by SCALE6.0/ORIGEN-ARP. These FQTs are documented in Section M.5.2.6.

In this chapter, the original design basis source terms developed by SCALE4.4/SAS2H, and the associated dose rate analysis, are retained as the analysis of record. However, SCALE6.0/ORIGEN-ARP design basis source terms are also developed consistent with the unified FQTs. These source terms are documented in Section Z.5.2.5. The CC source terms from Table Z.5-13 are added to the fuel source terms. The MCNP5 37PTH/HSM-H input files are rerun in MCNP5 using the SCALE6.0/ORIGEN-ARP design basis source terms to determine the impact on the dose rates. To minimize rework, the original analysis is maintained as the analysis of record; however, scaling factors are developed that are used to scale up the dose rates of the analysis of record. This updated shielding analysis that determines the effect of a reduced Uranium loading of 0.380 MTU per assembly on dose rates is summarized in Section Z.5.4.11.

The enveloping heat load zoning configuration (HLZC) utilized in the shielding evaluation is shown in Figure Z.5-1. This HLZC produces the highest dose rates on the surfaces of the HSM-H and OS200 TC as compared to the allowable HLZCs because the highest source fuel assemblies are on the outer periphery of the basket region where self-shielding due to adjacent assemblies is limited. This results in a shielding analysis corresponding to a total of 31.2 kW decay heat per DSC which is conservative because the total decay heat in 37PTH DSC is limited to 30 kW. These bounding gamma and neutron source terms are then used in the radiation shielding models to conservatively calculate dose rates on and around the NUHOMS®-37PTH System.

The bounding burnup, minimum initial enrichment, and cooling time combinations for the *0.490 MTU* fuel assemblies used in the shielding analyses of the 37PTH DSC in the HSM-H and the OS200 TC are as follows:

37PTH DSC in HSM-H:

- Zones 1 and 2: 40 GWD/MTU, 1.5 wt % ^{235}U, 40.2 years cooled, 0.40 kW/FA
- Zone 3: 16 GWD/MTU, 0.7 wt % ^{235}U, 3.1 years cooled, 0.70 kW/FA
- Zone 4: 26 GWD/MTU, 1.5 wt % ^{235}U, 3.1 years cooled, 1.2 kW/FA

37PTH DSC in OS200:

- Zones 1 and 2: 40 GWD/MTU, 1.5 wt % ^{235}U, 40.2 years cooled, 0.40 kW/FA
- Zone 3: 41 GWD/MTU, 1.5 wt % ^{235}U, 15.3 years cooled, 0.70 kW/FA
- Zone 4: 41 GWD/MTU, 1.5 wt % 235U, 5.7 years cooled, 1.2 kW/FA. For dry cask and dry DSC.
- Zone 4: 62 GWD/MTU, 3.4 wt % ^{235}U, 13.7 years cooled, 1.2 kW/FA. For wet cask and wet DSC.

The method of selecting the bounding source terms is explained in detail in Appendix Z.5.2.

The design basis CC source term that envelops all CCs allowed in the 37PTH DSCs is taken 72.48
from Appendix J for BPRAs. This is the same CC source term used in the 32PT, 24PTH and 32PTH1 systems as described in Appendix M, Appendix P and Appendix U, respectively. The

source term energy distribution is shown in Table Z.5-13. Any CC to be stored in a 37PTH DSC must be bounded by this source term.

Reconstituted and/or damaged fuel assemblies are also acceptable for storage in the 37PTH DSC. The maximum number of reconstituted fuel assemblies that can be loaded per DSC is 37. Fuel assemblies may contain up to 10 fuel rods that are reconstituted with stainless steel that is irradiated. There is no limit on the number of rods reconstituted with unirradiated stainless steel or zircaloy or low enriched UO_2 or other non-fuel material. There is no effect on the source terms/shielding due to the position of the reconstituted rods in the fuel rod array. Reconstituted fuel *with irradiated stainless steel rods* has a rather small effect on the dose rate such that for cooling times less than 10 years, 1 year of cooling time is added. *Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.* Damaged fuel, under normal conditions, has essentially no impact on the dose rate as the neutron and gamma source terms would not be impacted and gross axial source redistribution is not likely. Therefore, shielding analysis results with intact fuel are also applicable to the damaged fuel under normal conditions. For accident conditions, damaged fuel is assumed to redistribute and is analyzed separately. AMD 15 AMD 15 & 72.48

The methodology, assumptions, and criteria used in this evaluation are summarized in the following subsections.

The 32PTH1 DSC, analyzed in Appendix U, is also authorized for transfer and storage in OS200/OS200 FC and HSM-H/HSM-HS, respectively. The minimum thicknesses of the shielding materials on the bottom and top end of 32PTH1 DSC are 6.5″ and 10.0″, respectively. The minimum thickness of the shielding materials on the bottom and top end of 37PTH DSC are 7.25″ and 9.75″, respectively. The side shell for both the DSCs is 0.5″ thick. Both DSCs have solid aluminum transition rails on side of the baskets, but the 32PTH1 DSC has an option with hollow rails.

The bounding shielding performance evaluation for OS200/OS200 FC and HSM-H/HSM-HS containing 32PTH1 DSC is documented in Section U.5. The enveloping heat load zoning configuration (HLZC) utilized in the shielding evaluation is shown in Appendix U.5, Figure U.5-3. This HLZC produces the highest dose rates on the surfaces of the HSM-H and OS200 TC as compared to the allowable HLZCs because the highest source fuel assemblies are on the outer periphery of the basket region where self-shielding due to adjacent assemblies is limited. To bound the shielding analysis for all HLZCs, fuel assemblies with a decay heat of 1.5 kW at the outer 28 locations are used along with 1.0 kW fuel assemblies in the central 4 compartments. This results in a shielding analysis corresponding to a total of 46 kW decay heat per DSC which is very conservative because the total decay heat in 32PTH1 DSC is limited to 40.8 kW. These bounding gamma and neutron source terms are then used in the radiation shielding models to conservatively calculate dose rates on and around the NUHOMS®-32PTH1 system. Because shielding properties of 32PTH1 DSC materials are lower compared to those of 37PTH DSC, dose rates on and around the HSM and TC containing 32PTH1 DSC with 32PTH1design basis sources are *used to conservatively approximate* the dose rates from 37PTH DSC containing 37PTH design basis sources. Shielding analysis for the HSM-H containing 37PTH DSC has demonstrated this assertion. Therefore, it is *acceptable* to apply dose rates near the transfer cask documented in Appendix U.5 to the current analysis. AMD 15 AMD 15

Since results of the shielding analysis and dose rates near the transfer cask containing 32PTH1 DSC documented in Appendix U.5 are used directly in the current section, *representative* shielding configurations corresponding to 32PTH1 DSC and the cask containing 32PTH1 DSC are referred to in the current section as bounding DSC and bounding cask, respectively. Note, bounding cask/cask as well as bounding DSC/DSCs terms are used interchangeably in the text that follows unless required otherwise. Also, the shielding configuration of 32PTH1 DSC in HSM is referred to as the bounding HSM. Dose rates near the bounding cask and the bounding HSM are due to design basis radiological sources used in Appendix U.5. Those source terms result in dose rates that *conservatively approximate* the dose rates one would obtain using source terms in Table Z.5-7 through Table Z.5-12. Dose rates presented in Table Z.5-1 through Table Z.5-3, Figure Z.5-11 through Figure Z.5-19 correspond to the bounding HSM and the bounding cask. Those dose rates and figures are from Appendix U.5.

The NUHOMS® -37PTH DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. **[**

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Z.5.1 Discussion and Results

All 37PTH DSC, OS200 transfer cask, and HSM-H dose calculations are performed using MCNP5 code [5.2] and a composite (hypothetical) DSC shielding configuration. The axial geometry of the 37PTH-M DSC is used to accommodate the design basis B&W 15x15 Mark B fuel assemblies. Steel rails are used for calculation of long term storage, transfer, welding and accident dose rates. The presence of solid aluminum rails that fill the space between the peripheral fuel compartments of the bounding DSC and the DSC shell results in a more effectively shielded configuration when the bounding DSC is dry. Decontamination dose rates near the bounding cask are calculated with solid aluminum rails which produce conservative results.

Table Z.5-1 summarizes the maximum and average dose rates for the NUHOMS®-37PTH Design Basis (also referred to as "bounding") DSC loaded into the NUHOMS® HSM-H.

Table Z.5-2 provides a summary of the bounding dose rates on and around the OS200 TC for transfer of the 37PTH DSC under normal, off-normal and accident conditions.

Table Z.5-3 provides a summary of the bounding dose rates on and around the OS200 TC for decontamination and welding operations for the 37PTH DSC.

The dose rates reported in Tables Z.5-1 through Z.5-3 are scaled by footnotes to account for dose rate increases due to the 0.380 MTU/FA based source terms. The unified FQTs are documented in Section M.5.2.6, and the corresponding 37PTH source terms are documented in Section Z.5.2.5. The scaling factors are documented in Section Z.5.4.11.

A discussion of the method used to determine the design-basis fuel source terms is included in Appendix Z.5.2. The design basis CC source term, which is from Appendix J, is shown in Table Z.5-13. The shielding material densities are given in Section Z.5.3. The method used to determine the dose rates due to design-basis fuel assemblies with CCs in the various NUHOMS®-37PTH DSC design configurations is provided in Section Z.5.4. *R*adiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The shielding evaluation is performed with the MCNP5 [5.2] code with the ENDF/B-VI cross section library. Sample input files used for calculating neutron and gamma source terms and dose rates are included in Section Z.5.6.

In summary, the shielding evaluation of the 37PTH system is *conservatively approximated* by that of the 32PTH1 system described in Appendix U.5. Therefore, the results from Appendix U.5 are directly utilized herein.

Z.5.2 Source Specification

The 0.490 MTU/FA design basis source terms were generated using the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1]. The development of these source terms are provided in this section. For the addition of 0.380 MTU fuel, the original FQTs are replaced with FQTs developed using SCALE6.0/ORIGEN-ARP [5.20]. The development of these FQTs is documented in Section M.5.2.6. Because the FQTs have changed, SCALE6.0/ORIGEN-ARP design basis source terms are developed to determine the impact on the dose rates. The source terms developed using SCALE6.0/ORIGEN-ARP are documented in Section Z.5.2.5.

The B&W 15x15 assembly is the bounding fuel assembly design for shielding purposes because it has the highest initial heavy metal loading and ^{59}Co content of the hardware regions as compared to fuel assemblies listed in Appendix Z.2 which are authorized contents of the NUHOMS®-37PTH DSC. The neutron flux during reactor operation is peaked in the active fuel or in-core region of the fuel assembly and drops off rapidly outside the active fuel region. Much of the fuel assembly hardware is outside of the active fuel region of the fuel assembly. To account for this reduction in neutron flux, the fuel assembly is divided into four exposure "regions." The four axial regions used in the source term calculation are: the bottom (nozzle) region, the fuel (active fuel) region, the (gas) plenum region, and the top (nozzle) region. The B&W 15x15 fuel assembly masses for each irradiation region are listed in Table Z.5-4. The light elements that make up the various materials for the various fuel assembly materials are taken from Reference [5.4] and are listed in Table Z.5-5. The design-basis heavy metal weight is 0.490 MTU *for the analysis of record.* These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. To account for the reduction in neutron flux outside the active fuel regions neutron flux (fluence) scaling factors are applied to light element composition for each region. The neutron flux scaling factors which are from Reference [5.4] are given in Table Z.5-6.

Evaluations of the existing data with SAS2H and the 44-group ENDF/B-V library used in the analysis are documented in References [5.11] and [5.12]. These comparisons all show generally good agreement between the calculations and measurements, and show no trend as a function of burnup in the data that would suggest that the isotopic predictions, and therefore neutron and gamma source terms, would not be in good agreement. A similar conclusion is also reached by the results documented in JAERI report [5.13]. In fact, for the case with 46,460 MWd/MTU burnup, the isotopic predictions are all within 2% of those measured. Therefore, the uncertainty in the gamma source term, and associated dose rates, is estimated to be within ±5 %.

The above discussion does not include high-burnup data up to 62 GWd/MTU. However, as documented in Reference [5.14] and confirmed in the SAS2H analysis, the total neutron source with increasing burnup is more and more dominated by spontaneous fission neutrons. Reviewing the output from the SAS2H runs, the neutron source term is due almost entirely to the spontaneous fission of ^{244}Cm (~94% of all neutrons both spontaneous fission and (α, n)). After reviewing the measured ^{244}Cm content compared to the ^{244}Cm content predicted by SAS2H and the 44-group ENDF/B-V library documented in References [5.11] and [5.12] for burnups up to 46,460 MWd/MTU, it is readily apparent that the calculated values are within ±11 % of the measured values, with most of the predicted values within ±5% of the measured. Finally, there is no observed trend as a function of burnup in the data that would indicate that the predicted ^{244}Cm

content is significantly different at higher burnups. Therefore, as the ^{244}Cm isotope accounts for more than 94% of the total neutron source term, the uncertainty in the neutron source and associated neutron dose rates is expected to be less than ±11 %.

As documented in Reference [5.14] and as observed in preparing the fuel qualification tables, the gamma radiation source strength increases nearly linearly with burnup relative to the direct gamma component and the neutron radiation source strength increases with burnup to the fourth power. Therefore, as burnups go beyond 45 GWD/MTU, the contribution from neutron (and associated n,γ) components to the total dose rates measured on the surfaces of the DSC, TC and HSM-H increase in relative importance to that of the gamma component. However, this increase in the importance of the neutron source term has a relatively minor effect on the area dose rates on and around the HSM as these are dominated by the gamma component as shown in Table Z.5-1. The surface dose rates on the HSM are dominated by the gamma component because the HSM is constructed of thick reinforced concrete, which is an excellent neutron shield. Therefore, even a postulated substantial increase in the neutron source term would have a relatively minor effect on the site dose rate evaluation presented in Appendix Z.11 of the amendment application.

For the TC, the neutron source term has a relatively minor effect on the area dose rates during most of the cask handling operations, since the DSC cavity and the annulus between the TC and DSC is filled with water and most of the work is done around the top of the cask. The neutron component is of more importance on and around the TC during transfer operations but, in general, only represents a small portion of the total dose rate on the top of the TC. While the neutron dose rate on the bottom of the TC is slightly higher, relatively little occupational dose is received from this area. The dose rates for the design basis fuel on the surfaces of HSM and TC are shown in Tables Z.5-1 through Z.5-3. These tables show that gamma dose rates are substantially higher than neutron dose rates.

The occupational exposure calculations demonstrate that most of the dose received by workers during cask loading and transfer operations is due to the gamma radiation on and around the cask. The only surface of the TC that is dominated by neutrons is at the bottom of the cask. A small fraction of the total occupational exposure is due to the doses around the bottom of the cask because very little work is performed on or around the bottom of the cask with fuel in the TC.

As discussed above, any impact of uncertainties in source terms is expected to be statistically insignificant for the 37PTH system. Therefore, isotopic depletion calculations with SAS2H for fuel burned above 45 GWD/MTU are appropriate.

The above discussion on the applicability of SCALE 4.4/SAS2H to compute gamma and neutron high-burnup source terms is also largely applicable to SCALE 6.0/ORIGEN-ARP because both code systems utilize ORIGEN-S for the depletion calculation. For PWR fuel assemblies, SAS2H and ORIGEN-ARP generate comparable source terms for equivalent program inputs. The TRITON T-DEPL module of SCALE 6.0 is used to generate ORIGEN-ARP libraries applicable to B&W 15x15 fuel assemblies. These libraries are then used by ORIGEN-ARP to compute gamma and neutron source terms (see Section Z.5.2.5).

Oak Ridge National Laboratory has benchmarked TRITON based on measured data from six different PWRs. This benchmarking is documented in NUREG/CR-6968 [5.17], NUREG/CR-7012 [5.18], and NUREG/CR-7013 [5.19] and includes measurement samples up to a burnup of 78.3 GWd/MTU. A summary of experimental samples utilized in the benchmark analysis is provided in Table Z.5-20. The benchmark references show that TRITON computed results agree well with experiments, thus verifying the use of SCALE 6.0/ORIGEN-ARP to compute gamma and neutron source terms for high-burnup fuel (burnup ≤ 62 GWd/MTU). Because SAS2H and ORIGEN-ARP compute similar source terms for PWR fuel, the overall uncertainty in the gamma and neutron source terms developed above for SAS2H (±5% for gammas and ±11% for neutrons) is also applicable to the SCALE 6.0/ORIGEN-ARP generated source terms.

As discussed previously, reconstituted and/or damaged fuel is also acceptable for the DSC payload. Reconstituted fuel may contain up to 10 fuel rods replaced with irradiated solid stainless steel rods. Reconstituted fuel has a rather small effect on the dose rate such that for cooling times less than 10 years, 1 year of cooling time is added if reconstituted irradiated stainless steel rods are present. If the cooling time is greater than 10 years, no additional cooling time is needed. *Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.* Under normal conditions, damaged fuel has essentially no impact on the dose rate as the source term would not be impacted and gross axial source redistribution is not likely. Damaged fuel under accident conditions is addressed by assuming the fuel turns to rubble. This assumption is only applicable to the transfer casks shielding analysis. AMD 15 & 72.48

Parameters that influence the source term calculations are fuel assembly power (expressed in MW/fuel assembly (MW/FA)) and the total time between cycles. Other depletion parameters like cycle length and number of cycles are derived from the target burnup, MTU loading and power. The time between cycles utilized is 30 days and is adequately bounding. AMD 15 AMD 15

The design-basis source terms are defined as the burnup/initial enrichment/cooling time combination given in the fuel qualification tables that result in the maximum dose rate on the surface of the HSM (HSM-H) or TC (OS200). Note that for a given HLZC, the design basis HSM source will not necessarily be the same as the corresponding design basis TC source. For the HSM, the middle of the roof centerline is selected as the dose location, and for the middle of the TC the cask side is selected as the dose location. This approach is consistent with the method used to determine the fuel qualification tables for the Standardized NUHOMS® canister designs described in Chapter 7, Section 7.2.3 and Appendices M.5, P.5, and U.5.

HLZC 3 (Figure Z.2-3 in Appendix Z.2) produced the bounding total surface dose rate for both the HSM-H and OS200 TC containing the 37PTH DSC. The enveloping HLZC selected for the shielding analysis (shown in Figure Z.5-1) of the 37PTH DSC bounds the actual heat load configuration shown in Figure Z.2-3 because 1.2 kW fuel is assumed in all 16 peripheral locations.

A sample SAS2H/ORIGEN-S input file for the active fuel region for the 26 GWD/MTU, 1.5 wt. % U-235, and 3.1-years cooling case is listed in Section Z.5.6.1. Input for reconstituted fuel is similar, except for a reduced number of fuel pins from 208 to 198, light element masses that reflect reconstituted rods, and slightly different power input to maintain the same burnup for a reduced fuel mass.

Z.5.2.1 Gamma Source Term for MCNP

Z.5.2.1.1 Design Basis Gamma Fuel Assembly Source Terms

Once the design basis burnup/enrichment/cooling time combinations have been determined for each shielding configuration of interest, four SAS2H/ORIGEN-S runs are required for each combination to determine gamma source terms for the four fuel assembly regions (i.e., bottom, active fuel, plenum and top). The only difference between the runs is in Block #10 "Light

Elements" of the SAS2H input and the 82$$ card in the ORIGEN-S input. Each run includes the appropriate light elements for the region being evaluated and the 82$$ card is adjusted to have ORIGEN-S output the total gamma source for the active fuel region and only the light element source for the plenum, bottom, and top regions. Gamma source terms for the active fuel region include contributions from actinides, fission products, and activation products. The bottom, plenum and top nozzle regions include the contribution from the activation products in the specified region only. The SAS2H/ORIGEN-S gamma radiation source is output in the CASK-81 energy group structure.

A design basis source is developed for each decay heat (0.4, 0.7, and 1.2 kW) and shielding structure combination used in the shielding analysis. The enveloping configuration evaluated in the shielding analyses is based on four radial zones. Radial zone 1 is comprised of the center fuel compartments of the 37PTH DSC, radial zone 2 is comprised of the inner middle 8 assemblies, and the outer middle 12 assemblies are of radial zone 3. The remaining 16 outer assemblies define radial zone 4. Source terms are generated for the following enveloping (hypothetical) configuration.

(1) Radial zone 1 and 2: 0.4 kW fuel
(2) Radial zone 3: 0.7 kW fuel
(3) Radial zone 4: 1.2 kW fuel

The source terms for radial zone 1 and 2 fuel in a 37PTH DSC loaded in the HSM-H (0.4 kW, 40 GWD/MTU, 1.5 wt. % U-235, and 40.2-years cooling) are shown in Table Z.5-7. The source terms for radial zone 3 fuel in a 37PTH DSC loaded in the HSM-H (0.7 kW, 16 GWD/MTU, 0.7 wt. % U-235 and 3.1-years cooling) are shown in Table Z.5-8. The source terms for radial zone 4 fuel in a 37PTH DSC loaded in the HSM-H (1.2 kW, 26 GWD/MTU, 1.5 wt. % U-235, 3.1-years cooling) are shown in Table Z.5-9. The bounding radiological source terms for 37PTH DSC in the cask from assemblies in the similar zones are shown in Table Z.5-7, Table Z.5-10 through Table Z.5-12. Dose rates on and around HSM containing 37PTH DSC from sources in Table Z.5-7 through Table Z.5-9 are *conservatively approximated* by the dose rates presented in Appendix U.5, Table U.5-1. Dose rates near the bounding cask and the bounding HSM are due to radiological sources in Table U.5-7 through Table U.5-9.

Z.5.2.1.2 Design Basis CC Source Terms

The design basis CC source terms are taken from Appendix J and are listed in Table Z.5-13. All CCs to be stored in the 37PTH DSC must be bounded by this source. The source terms from the fuel assembly and the CCs are utilized in the MCNP shielding models.

Z.5.2.1.3 Uncertainty in Gamma Source Terms

Almost 100% of the gamma spectrum from light elements is in the range of 0.70 to 1.33 MeV which corresponds exactly to two of the most prominent lines of ^{60}Co. As for fission products, the main contributors after six years with a fraction greater then 5% in the range of 0.01 to 0.90 MeV are: ^{90}Sr, ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, ^{154}Eu, and ^{155}Eu. Contributions from ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, and ^{154}Eu are dominant in the range of 0.90 to 1.50 MeV. ^{106}Rh, ^{147}Sm, and ^{142}Ce are the strongest emitters at energies greater then 2.0 MeV. The accuracy of gamma spectrum is

dependent upon the energy. Photon rates computed for fission products tend to be more accurate then those for actinides because the calculation of their inventory has less uncertainty [5.1].

Shortly after discharge the emission at higher energies is dominated by actinides. This is true for energies >4 MeV at all cooling times and energy above 3.5 MeV for cooling times after 10 years [5.1]. The major part of this emission comes from ^{244}Cm. Thus the uncertainty for energy groups of order 3.0 MeV and greater is bounded with the precision with which the inventory of 244Cm is calculated. Per SCALE 4.4 [5.1], reported experimental ^{244}Cm densities are accurate within ± 20%. The gamma emission intensity from Cm, which is proportional to the quantity of Cm in the actinide inventory, is bounded by this value. Uncertainty in the source strength in the gamma energy range 0.5 to 2.5 MeV is approximately 10 to 15 % [5.1].

Z.5.2.2 Neutron Source Term for MCNP

One SAS2H/ORIGEN-S run is required for each burnup/initial enrichment/cooling time combination to determine the total neutron source term for the active fuel regions. At discharge the neutron source is almost equally produced from ^{242}Cm and ^{244}Cm. The other strong contributor is ^{252}Cf, which is approximately 1/10 of the Cm intensity, but its share vanishes after 6 years of cooling time because the half-life of ^{252}Cf is 2.65 years. The half-lives of ^{242}Cm and ^{244}Cm are 163 days and 18 years, respectively. Contributions from the next strongest emitters, ^{238}Pu and ^{240}Pu, are lower by a factor of 1000 and 100, respectively, relative to ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 85% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein.

The magnitude of the neutron source is provided as the final row in the gamma source term tables; see Tables Z.5-7 through Table Z.5-12. Neutron source terms for use in the MCNP shielding models are calculated by multiplying the assembly source by the number of assemblies in heat zones of interest. The magnitude of the neutron source is also increased to account for the axial distribution in the fuel, as explained in Section Z.5.2.3. The neutron source terms for 37PTH DSC in the cask from assemblies in the similar zones are shown in the last row of Table Z.5-7, Table Z.5-10 through Table Z.5-12. Dose rates near the cask containing 37PTH DSC from radiological sources in those tables are *conservatively approximated* by the dose rates presented in Appendix U.5.

The fixed source spectrum in MCNP is assumed to follow a ^{244}Cm spontaneous fission spectrum for all of the calculations in this chapter, except Section Z.5.5. It is based on the following relationship:

$$f(E) = C\exp\left[\left(\frac{-E}{a}\right)\sinh(bE)^{1/2}\right]$$

where input parameters $a = 0.906$ MeV and $b = 3.848$ (MeV)$^{-1}$, as given in the MCNP manual [5.2] and E is energy (MeV).

Z.5.2.3 Axial Peaking

Axial burnup peaking factors for PWR fuel are taken from References [5.6] and [5.16]. These peaking factors are assumed to match the gamma axial source distribution because the gamma

rate with design basis fuel, an additional 0.5 year of cooling time is added to the reconstituted fuel source term. When the reconstituted fuel is examined in this fashion, no more than one additional year of cooling time is required for reconstituted fuel to be bounded by the design basis source if the decay time listed in the fuel qualification table is less than 10 years. After a cooling time greater than 10 years the effects of reconstituted fuel become insignificant.

The SAS2H input files for a reconstituted assembly are very similar to the input files for a standard assembly except for the following changes: (1) The number of fuel rods is reduced from 208 to 198, (2) the POWER input variable is adjusted to maintain the correct burnup for the reduced fuel loading, and (3) the light elements change to reflect that 10 fuel rods have been replaced with steel rods. The constituent masses of the reconstituted fuel assembly required for the SAS2H input is provided in Table Z.5-4.

Note that a reconstituted rod cannot be irradiated for more than two cycles because the first cycle will always contain fresh, undamaged fuel. To accurately model this behavior, two SAS2H models are generated for each transition point. The first SAS2H model is for only one cycle of irradiation of 10 reconstituted rods, while the second SAS2H model is for three cycles of irradiation of 10 reconstituted rods. By subtracting the single cycle source term of the reconstituted rods from the total source term (fuel and reconstituted rods) for three cycles, the source term for three cycle irradiation of fuel and two cycle irradiation of reconstituted rods is generated. *Alternatively, the licensee can qualify fuel assemblies with fewer than the maximum number of irradiated stainless steel rods and reduce cooling time requirements.* AMD 15 & 72.48

A sensitivity study performed using the 0.380 MTU/FA source terms indicates that the additional cooling time requirements for reconstituted fuel remain valid for the unified FQTs (Technical Specifications Tables 1-3a through 1-3p).

Z.5.2.5 SCALE6.0/ORIGEN-ARP Source Terms

Because the system specific FQTs have been replaced with unified FQTs, the design basis source terms developed in Section Z.5.2 are obsolete, as they are based upon burnup, enrichment, and cooling time combinations that are no longer applicable. Therefore, SCALE6.0/ORIGEN-ARP design basis source terms are developed based on the unified FQTs. The FQTs are documented in Section M.5.2.6. AMD 15

The methodology used to develop the SCALE6.0/ORIGEN-ARP design basis source terms is the same as described in Section Z.5.2. HSM response functions are used to evaluate the source terms for each FQT burnup, enrichment, and cooling time (BECT) combination. The BECT combination that results in the maximum dose rate is selected as the design basis source.

Source terms are not needed for the transfer cask because it has been demonstrated that the dose rate behavior of the 32PTH1 DSC in the transfer cask reasonably approximates that of the 37PTH DSC in the transfer cask. Therefore, only the 37PTH DSC HSM source terms are determined.

The heat load zone configuration used in the shielding analysis is provided in Figure Z.5-1. Based on the response function analysis, the SCALE6.0/ORIGEN-ARP design basis source terms for HSM analysis are:

HSM

- *Table Z.5-21: 0.380 MTU, 0.4 kW/FA, 45 GWd/MTU, 1.1 wt.% U-235, 36.4 years cooled*
- *Table Z.5-22: 0.380 MTU, 0.7 kW/FA, 10 GWd/MTU, 0.7 wt.% U-235, 2.4 years cooled*
- *Table Z.5-23: 0.380 MTU, 1.2 kW/FA, 19 GWd/MTU, 0.8 wt.% U-235, 2.4 years cooled*

Because the FQTs in Appendix M, Section M.5.2.6 are developed for uranium loadings of 0.380 MTU, 0.475 MTU, and 0.492 MTU for fixed heat loads, it is observed that 0.380 MTU source terms bound the 0.475 MTU and 0.492 MTU source terms because self-shielding of the sources by the uranium in the fuel lattice is reduced. Therefore, SCALE 6.0/ORIGEN-ARP design basis source terms are developed only for 0.380 MTU.

The MCNP5 neutron models are run with the NONU card to suppress subcritical neutron multiplication. Subcritical neutron multiplication is addressed by multiplying the neutron source computed by ORIGEN-ARP by $1/(1 - k_{eff})$. Values of k_{eff} appropriate for the burnups of the sources are provided in the source term tables (Table Z.5-21 through Z.5-23).

The gamma source in the active fuel region is modeled with the axial burnup profile appropriate for the burnup of the source and is obtained from Table 20 of ORNL/TM-12973 [5.16]. These profiles are summarized in Table M.5-53. The neutron profile is derived as the 4^{th} power of the gamma profile and is also summarized in Table M.5-53. The burnup peaking factor accounts for the increase in the neutron source magnitude due to the axial burnup profile. The burnup peaking factors used in the neutron calculations are provided in the source term tables (Table Z.5-21 through Z.5-23).

The CC source terms provided in Appendix Z, Table Z.5-13 are applicable and may be added to the fuel-only source terms provided in Appendix Z, Table Z.5-21 through Z.5-23.

Z.5.4 Shielding Evaluation

Dose rate contributions from the bottom, active fuel, plenum, and top regions, as appropriate, from 37 *0.490 MTU* fuel assemblies are calculated with the MCNP Code [5.2] at various locations on and around the NUHOMS®-37PTH DSCs within the HSM and OS200 TC.

The following shielding evaluation discussion specifically addresses the NUHOMS®-37PTH DSC in an OS200 TC and the NUHOMS®-37PTH DSC in HSM-H using the *0.490 MTU* design-basis source terms described in the above sections.

Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from 37 0.380 MTU fuel assemblies with CCs are also calculated with the MCNP5 Code [5.2] at various locations on and around the NUHOMS® 37PTH DSCs within the HSM and TC.

The shielding evaluation that determines the effect of loading 0.380 MTU per assembly on the dose rates is described in Appendix Z, Section Z.5.4.11.

Z.5.4.1 Computer Program

MCNP [5.2] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Pointwise (continuous energy) cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation are accounted for in the cross section set. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. Important standard features that make MCNP very versatile and easy to use include a powerful general source; an extensive collection of cross-section data; and an extensive collection of variance reduction techniques that can be employed to track particles through very complex deep penetration problems. MCNP was employed to take advantage of its mesh tallies capabilities in calculating dose rates distributed over the surface of the HSM. It also allows more point detectors to be used in a single run that substantially reduces the number of input/out decks needed to perform ISFSI site dose rate calculations described in Appendix Z.10.

Z.5.4.2 Spatial Source Distribution

The source components are:

the neutron sources due to the active fuel region,
the gamma source due to the active fuel region,
the gamma source due to the plenum,
the gamma source due to the top region,
the gamma source due to the bottom region,
the gamma source due to the CC in the active fuel region,
the gamma source due to the CC in the plenum region, and
the gamma source due to the CC in the top region.

Axial peaking is accounted for in the active fuel region by inputting an axial shape, as discussed in Section Z.5.2.3.

Z.5.4.3 Cross Section Data

The cross-section data used is the continuous energy ENDF/B-VI provided with the MCNP code [5.2]. The cross-section data allows coupled neutron/gamma-ray dose rate evaluation to be made to account for secondary gamma radiation (n, γ), if desired. All of the TC and HSM-H dose rate calculations account for the dose rate due to secondary gamma radiation.

Z.5.4.4 Flux-to-Dose-Rate Conversion

The flux distribution calculated by the MCNP code is converted to dose rates using flux-to-dose rate conversion factors from ANSI/ANS-6.1.1-1977 [5.8] given in Table Z.5-16.

Z.5.4.5 Methodology

The methodology used in the shielding analysis of the bounding cask and HSM is based on the one employed in the 32PTH1 system described in Appendix U.5. The MCNP computer code was utilized to perform the shielding analyses. MCNP allows for explicit 3-D modeling of any shielding configuration. The methodology used herein is summarized below.

1. Sources are developed for all fuel regions using the source term data described in Section Z.5.2. Source regions include the active fuel region, bottom end fitting (including all materials below the active fuel region), plenum, and top end fitting (including all materials above the plenum region). An effect of CC sources on dose rates is also accounted for in these evaluations.
2. The response functions developed for the 37PTH DSC were shown to be lower than the response functions for the 32PTH1 DSC. Differences in shielding properties and bounding radiological sources of 32PTH1 and 37PTH DSCs were analyzed. They are discussed in Section Z.5.2. 32PTH1 bounding sources are stronger than those for 37PTH.This implies the dose rate for the HSM and the cask containing 37PTH DSC will be lower than those for HSM and the cask containing 32PTH1 bounding DSC.
3. An analysis was performed with the 37PTH DSC in the HSM-H. Surface averaged doses were tabulated and are shown in Table Z.5-19. These dose rates are compared to the 32PTH1 DSC in the HSM, as shown in Table Z.5-1 *conservatively approximated.* The dose rates for the 37PTH DSC are bounded by the 32PTH1 DSC when both are loaded with their respective design basis fuel and HSM-H. This indicates that the dose rates near *the* transfer cask containing 37PTH DSC with 37PTH design basis sources are also *conservatively approximated* by dose rates near the transfer cask containing bounding 32PTH1 DSC with 32PTH1 design basis sources. Therefore, transfer cask dose rates presented in Appendix U.5 can be used to represent dose rates near the cask containing 37PTH DSC.
4. MCNP results from 32PTH1 are used to *derive* offsite exposures (see Appendix Z.10).

5. MCNP models are also generated to determine the effects of accident scenarios, such as loss of cask neutron shield for the OS200 TC (Appendix Z.11). Dose rates near the transfer cask documented in Appendix U.5 to describe a shielding performance of the cask containing 32PTH1 design basis sources are conservatively applied for the cask containing 37PTH DSC with the 37PTH design basis sources. The design basis sources for the cask containing 37PTH DSC are presented in Table Z.5-7 and Table Z.5-10 through Table Z.5-12.

Z.5.4.6 Assumptions

The following general assumptions are used in the analyses.

Z.5.4.6.1 Source Term Assumptions

- The primary neutron source in LWR spent fuel is the spontaneous fission of ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 85% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein and is assumed to follow the ^{244}Cm fission spectrum provided in Section Z.5.2.2.
- Surface gamma dose rates are calculated for the HSM and cask surfaces using the actual photon spectrum applicable for each case.
- The PWR heavy metal weight is assumed to be 0.490 MTU per assembly to bound existing PWR fuel designs *in the analysis of record.*

Z.5.4.6.2 HSM-H Dose Rate Analysis Assumptions

- The 37PTH DSC and fuel assemblies are positioned at approximately 30 cm to the HSM-H front door.
- Planes of reflection are used to simulate adjacent HSM-Hs in a side-by-side arrangement.
- Embedments and rebar in the HSM-H concrete are conservatively neglected.
- Penetrations on the exterior of the HSM-H modules for instrumentation and ease of installation are not modeled since they do not result in any significant change in dose rate distribution.
- The borated neutron absorber sheets in the 37PTH DSC are modeled as aluminum.
- Axial peaking factors assumed as shown in Table Z.5-14.
- Fuel is homogenized within the fuel compartment, although the 37PTH DSC basket is modeled explicitly.

Z.5.4.6.3 OS200 TC Dose Rate Analysis Assumptions

For the 37PTH, *dose rates for* the 32PTH1 loaded in the OS200 TC *conservatively approximate the dose rates* for the 37PTH loaded in the OS200 TC. Appropriate assumptions from Appendix U.5 are *applicable* for the 37PTH DSC in the OS200 TC.

Z.5.4.7 Normal Condition Models

Two classes of MCNP models are developed: (1) 37PTH DSC in HSM-H and (2) 32PTH1 DSC in OS200 TC. These models are described in subsequent sections.

Z.5.4.7.1 37PTH DSC in HSM-H

Two three-dimensional MCNP models are developed for the 37PTH bounding DSC within a HSM-H: one model for neutrons and the other for gammas. Note that the DSC is loaded in HSM-H in accordance with the bounding HLZC depicted on Figure Z.5-1. This is a fictitious HLZC but it results in HSM dose rates that are bounding for all DSC/HSM shielding and source terms combinations defined for NUHOMS®-37PTH system. These models are presented in Figures Z.5-2 through Z.5-6. The HSM-H length is designated as the *z* axis, the width as the *x* axis, and the height as the *y* axis. The HSM-H door is designated as the south side and the *–z* direction, with the west wall as the *–x* direction. The roof is the *+y* direction. The west wall is designated as a reflective boundary and an end shield wall (3 ft thick) is attached to the east wall.

The bottom (bottom of bottom fitting) of the fuel assembly is assigned to a z plane at 103.51 cm. The center of the HSM-H inner cavity is at $(x, y, z) = (0, 0, 346.71)$. The 37PTH DSC lid on top end is located at approximately 3″ from the HSM-H rear wall (z = 556.26 cm). The bottom of the DSC is at z = 85.09 cm, about 21.6 cm in from the door interior. The 37PTH DSC support rails are included in the model. The heat shields are modeled as flat plates and horizontal vent "liner" plates (2 cm thick) are modeled in the top side vents. The HSM-H door is modeled with 3 inches of stainless steel and approximately 25 inches of concrete.

The dose rate results calculated from these evaluations are shown in Table Z.5-19. These results confirm that the use of dose rates calculated for the 32PTH1 DSC described in Appendix U.5 are *applicable.*

Dose rates are calculated on thin cells surrounding the HSM-H and are segmented into 20 to 30 cm increments to capture the peak dose rates. Dose rates are also calculated at the inlet and outlet vents. Dose rates *applicable to* the 37PTH are provided in Table Z.5-1. Dose rates for the front, side shield wall and roof surface at DSC centerline of the HSM-H are also plotted as a function of distance in Figures Z.5-11 through Z.5-16, respectively.

A sample MCNP model input file of HSM-H with 37PTH DSC is included in Section Z.5.6.2.

Z.5.4.7.2 32PTH1-L DSC in OS200 TC

For the 37PTH, *dose rates for* the 32PTH1 loaded in the OS200 TC *conservatively approximate dose rates* for the 37PTH loaded in the OS200 TC. The material presented is directly from Appendix U.5, Section U.5.4.7.2.

Two three-dimensional MCNP models are employed for shielding analyses of the 32PTH1 DSC within an OS200 TC: one model for neutrons and the other for gammas. These models are presented in Figures Z.5-6a through Z.5-10. The *z*-axis in the MCNP models coincides with the axis of rotation of the cask and the 32PTH1 DSC. Select features within the cask and on its surface are neglected because they produce only localized effects and have minimal impact on operational dose rates. Examples of neglected features include the relief valves, clevises, and eyebolts. With the exception of the neutron shield support angles and the trunnions, the balance of these items are local features that increase the shielding in a small area without replacing any of the shielding material which is included in the model. The additional shielding material that

Table Z.5-3.

Welding and 32PTH1-DSC Draining: Before the start of welding operation, approximately 60% of the water in the DSC cavity is removed due to hydrogen generation. A dry DSC cavity is assumed in all welding models to be conservative. Temporary shielding consisting of three inches of NS-3 and one inch of steel is assumed to cover the 32PTH1 DSC inner top cover plate. In addition, the DSC outer top cover plate is not present. The cask/32PTH1 DSC annulus is assumed to remain completely filled with water. Results for this case are provided in Table Z.5-3.

Z.5.4.10 Impact on Dose Rates due to Reduced Density Concrete and Gaps between HSMs

A bounding analysis is performed by employing a minimum concrete density of 140 pounds per cubic foot (pcf) in the HSM-H MCNP model combined with a maximum gap of 1.5 inches between adjacent HSM-H modules and shield walls to determine the effect on maximum and average dose rates due to a fully loaded 32PTH1 DSC. These calculations are documented in Appendix U.5, Section U.5.4.10. The ratios shown in Appendix U.5, Table U.5-*18* and Table U.5-*19* can be used as scaling factors to increase the maximum and surface-average dose rates of the 37PTH in the HSM-H to account for low density concrete and 1.5" gaps. Note that the HSM-H concrete contains high density rebar which is not credited in the MCNP models. Further, the modules are installed adjacent to each other such that there will not be a "uniform" gap of 1.5 inches. Ignoring the effect due to increased vent dose rates, the increase in the average dose rates caused by both the maximum postulated uniform gaps and the minimum postulated concrete density is expected to be less than 20% at the front and roof surfaces of the HSM-H module. Dose reduction hardware may be installed to further reduce these dose rates.

Z.5.4.11 Shielding Analysis with a Loading of 0.380 MTU per Fuel Assembly

As discussed in Section Z.5.4, additional shielding analysis is performed with a reduced Uranium loading of 0.380 MTU per fuel assembly. The objective of this analysis is to determine the impact that reduced Uranium loading has on system dose rates. The results of this analysis are employed to scale the dose rate results for the 37PTH System. For this purpose, the MCNP5 models employed for the 0.490 MTU analyses are rerun with updated source terms as described in Section Z.5.2.5, and with updated material specifications to reflect the reduction in MTU. MCNP5 calculations are performed for the 37PTH DSC inside the HSM-H, and dose rate scaling factors are derived using the same methodology as that described in Appendix U, Section U.5.4.12. As described in Section Z.5.4.5, the TC dose rates obtained from Section U.5 for the 32PTH1 DSC are applicable to the 37PTH DSC. The TC analysis for the 0.380 MTU loading is documented in Appendix U, Section U.5.4.12 and includes decontamination, welding, normal and accident conditions of transfer dose rates, and occupational exposure. The resulting dose rates, and occupational exposure are compared to the 0.490 MTU dose rates, and occupational exposure to determine scaling factors for these configurations.

Based on the updated results, six scaling factors are determined and are summarized as follows:

- *The dose rates for the HSM-H front and roof are to be scaled by 1.25.*
- *The dose rates for the HSM-H side and rear are to be scaled by 1.35.*
- *The site dose for the HSM is to be scaled by 1.25.*
- *The dose rates for the TC for normal, welding and decontamination are to be scaled as follows:*
    - *by 1.24 for the side,*
    - *by 1.63 for the top,*
    - *by 1.91 for the bottom.*
- *The dose rates for the TC for accidents are bound by the 0.490 MTU analysis results, and do not require updating.*
- *The occupational exposure for TC loading and storage operations is to be scaled by 1.25.*

These scaling factors are included as footnotes in the dose rate results summarized in Table Z.5-1, Table Z.5-2, Table Z.5-3, and Table Z.5-19.

These scaling factors are also employed to scale the occupational exposure and generic site dose (2X10 back-to-back and front-to-front arrays) results calculated for the 37PTH system in Section Z.10, and to scale the dose rate consequences of accidents for the 37PTH system in Section Z.11.

Z.5.7 References

5.1 Oak Ridge National Laboratory, RSICC Computer Code Collection, "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluations for Workstations and Personal Computers," NUREG/CR-0200, Revision 6, ORNL/NUREG/CSD-2/V2/R6.

5.2 "Monte Carlo N-Particle Transport Code System," CCC-730, Oak Ridge National Laboratory, RSICC Computer Code Collection, August 2001.

5.3 CASK-81 - 22 Neutron, 18 Gamma-Ray Group, P3, Cross Sections for Shipping Cask Analysis," DLC-23, Oak Ridge National Laboratory, RSIC Data Library Collection, August 1987.

5.4 Ludwig, S.B., and J.P. Renier, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018 Oak Ridge National Laboratory, December 1989.

5.5 Not used.

5.6 "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," NUREG/CR-6801, Oak Ridge National Laboratory.

5.7 Jenal, J. P., P. J. Erickson, W. A. Rhoades, D. B. Simpson, and M. L. Williams, "The Generation of a Computer Library for Discrete Ordinates Quadrature Sets," ORNL/TM-6023, Oak Ridge National Laboratory, October 1977.

5.8 "American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors," ANSI/ANS-6.1.1-1977, American Nuclear Society, LaGrange Park, Illinois, March 1977.

5.9 K. Ueki, N. Nariyama, A. Ohashi, A. Yamaji. "Measurement of Dose-Equivalent Rates around a Cask and Monte Carlo Analysis with Actual Configuration of Fuel Basket". Journal of Nuclear Science and Technology, (Supplement 1, p.324-328), Atomic Energy Society of Japan, March 2000, ISSN 0022-3131.

5.10 *Not used.*

5.11 MD DeHart and OW Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, September 1996.

5.12 OW Hermann, SM Bowman, MC Brady, CV Parks, "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, March 1995.

5.13 Japan Atomic Energy Research Institute, "Technical Development on Burn-up Credit for Spent LWR Fuels," JAERI-Tech 2000-071, September 21, 2000.

5.14 U.S. Nuclear Regulatory Commission, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High Burnup LWR Fuel," NUREG/CR-6700, Published January 2001, ORNL/TM-2000/284.

5.15 "Characteristics of Potential Repository Waste," DOE/RW-0184-R21, Volume 1, Oak Ridge National Laboratory, Tennessee, July 1992.

5.16 M. D. DeHart, "Sensitivity and Parametric Evaluations of Significant Aspects of Burn-up Credit for PWR Spent Fuel Packages", ORNL/TM-12973, May 1996.

5.17 U.S. Nuclear Regulatory Commission, "Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors," NUREG/CR-6968, Published February 2010, ORNL_TM-2008-71.

5.18 U.S. Nuclear Regulatory Commission, "Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel," NUREG/CR-7012, Published January 2011, ORNL-TM-2010-41.

5.19 U.S. Nuclear Regulatory Commission, "Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation -- Vandellós II Reactor," NUREG/CR-7013, Published January 2011, ORNL-TM-2009-321.

5.20 ORNL/TM-2005/39, Version 6, SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Oak Ridge National Laboratory, January 2009.

Table Z.5-1
Summary of Bounding Maximum and Average Dose Rates for HSM-H(2) Containing NUHOMS®-37PTH DSC

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1)(3) (mrem/hr)	Total MCNP 1σ Error
HSM Roof (centerline)(4)	14.13	0.04	0.74	0.01	**14.86**	0.04
HSM Roof Bird screen(4)	114.71	0.01	6.47	0.01	***121.17***	0.01
HSM End (Side) Shield Wall Surface(5)	1.49	0.05	0.06	0.01	***1.54***	0.05
HSM Door Exterior Surface (centerline)(4)	0.61	0.07	0.08	0.06	***0.70***	0.06
HSM Front Bird screen(4)	471.28	0.04	5.89	0.02	**477.17**	0.04

Dose Rate Location	Gamma Average (mrem/hr)	Gamma MCNP 1σ Error	Average Neutron (mrem/hr)	Neutron MCNP 1σ Error	Average Total(3) (mrem/hr)	Total MCNP 1σ Error
HSM Roof(4)	11.27	0.01	0.62	<0.01	**11.89**	0.01
HSM End (Side) Shield Wall Surface(5)	0.36	0.01	0.02	<0.01	**0.38**	0.01
HSM Front(4)	14.87	0.06	0.31	0.02	**15.19**	0.06
HSM Back Shield Wall(5)	0.07	0.01	0.004	0.01	**0.07**	0.01

(1) Gamma and Neutron dose rate peaks do not always occur at the same location; therefore, the maximum of total dose rate is not always the sum of the gamma plus neutron dose rate maximums.

(2) Dose rate is calculated using *the* 32PTH1 bounding DSC. This DSC contains the design basis assembly source loaded in accordance with the HSM bounding loading configuration.

(3) Use the ratios shown in Appendix U.5, Table U.5-*18* and Table U.5-*19* to increase the maximum and surface-average dose rates respectively to account for reduced density concrete and gaps of up to 1.5" as described in Appendix U.5, Section U.5.4.10.

(4) These dose rates increase by 25% when loading 0.380 MTU FAs.

(5) These dose rates increase by 35% when loading 0.380 MTU FAs.

Table Z.5-2
Summary of NUHOMS®-37PTH DSC, OS200 TC Maximum Dose Rates During Transfer Operations

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1) (mrem/hr)	Total MCNP 1σ Error
Cask Side Surface (Radial)*(3)*	4.07E+02	0.0048	2.02E+02	0.0085	6.09E+02	0.0043
Cask Top Axial Surface*(4)*	2.32E+02	0.0602	3.86E+01	0.0452	2.51E+02	0.0556
Cask Bottom Axial Surface*(5)*	2.15E+03(2)	0.0181	1.40E+03(2)	0.0136	3.55E+03(2)	0.0122
50 cm from Cask Side (Radial)*(3)*	2.39E+02	0.0047	1.23E+02	0.0082	3.62E+02	0.0042
50 cm from Cask Top Axial Surface*(4)*	4.71E+01	0.0793	2.07E+01	0.0453	5.95E+01	0.0653
50 cm from Cask Bottom Axial Surface*(5)*	9.25E+02	0.0190	3.43E+02	0.0194	1.27E+03	0.0148
1m from Cask Side (Radial)*(3)*	1.61E+02	0.0046	8.39E+01	0.0082	2.45E+02	0.0041
1m from Cask Top Axial Surface*(4)*	2.95E+01	0.0488	1.44E+01	0.0692	3.79E+01	0.0393
1m from Cask Bottom Axial Surface*(5)*	4.65E+02	0.0200	1.40E+02	0.0299	6.05E+02	0.0169
Cask 1 m (Radial) Accident Condition	1.74E+02	0.0280	3.58E+03	0.003	3.76E+03	0.0031
Cask 100 m (Radial) Accident Condition	9.38E-02	0.0175	1.00E+00	0.0029	1.10E+00	0.0030
Cask 500 m (Radial) Accident Condition	5.25E-04	0.0188	3.40E-03	0.0043	3.92E-03	0.0045

(1) Gamma and neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.
(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 113 mrem/hr gamma, 71 mrem/hr neutron for a total average dose rate of 184 mrem/hr.
(3) The Side dose rates increase by 24% when loading 0.380 MTU FAs.
(4) The Top dose rates increase by 63% when loading 0.380 MTU FAs.
(5) The Bottom dose rates increase by 91% when loading 0.380 MTU FAs.

Table Z.5-3
Summary of NUHOMS®-37PTH DSC, OS200 TC Maximum Dose Rates During Decontamination and Welding Operations

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1) (mrem/hr)	Total MCNP 1σ Error
Decontamination						
Cask side surface (radial)*(4)*	3.46E+02	0.0035	3.73E+02	0.0040	7.19E+02	0.0027
Top axial surface*(5)*	8.32E+02	0.0226	9.34E+00	0.0308	8.33E+02	0.0226
Cask bottom axial surface*(6)*	1.70E+03(2)	0.0153	6.83E+01(2)	0.0424	1.77E+03(2)	0.0148
50 cm from cask side (radial)*(4)*	2.02E+02	0.0035	2.29E+02	0.0037	4.31E+02	0.0025
50 cm from top axial surface*(5)*	6.15E+02	0.0298	5.64E+00	0.0443	6.16E+02	0.0298
50 cm from cask bottom axial surface*(6)*	7.31E+02	0.0160	1.82E+01	0.0534	7.49E+02	0.0157
1m from cask side (radial)*(4)*	1.35E+02	0.0034	1.58E+02	0.0037	2.92E+02	0.0025
1m from top axial surface*(5)*	4.27E+02	0.0345	3.61E+00	0.0537	4.28E+02	0.0344
1m from cask bottom axial surface*(6)*	3.68E+02	0.0174	8.06E+00	0.0808	3.76E+02	0.0171
Welding						
Cask side surface (radial)*(4)*	3.09E+02	0.0053	1.47E+02	0.0076	4.56E+02	0.0043
Top axial surface*(5)*	7.20E+02	0.1107	4.21E+01	0.0618	7.62E+02	0.1047
Cask bottom axial surface*(6)*	2.17E+03(3)	0.0164	1.22E+03(3)	0.0110	3.39E+03(3)	0.0112
50 cm from cask side (radial)*(4)*	1.85E+02	0.0051	8.99E+01	0.0072	2.75E+02	0.0042
50 cm from top axial surface*(5)*	4.25E+02	0.0214	1.74E+01	0.0592	4.42E+02	0.0207
50 cm from cask bottom axial surface*(6)*	9.27E+02	0.0171	3.00E+02	0.0157	1.23E+03	0.0135
1 cm from cask side (radial)*(4)*	1.27E+02	0.0050	6.23E+01	0.0077	1.89E+02	0.0042
1 cm from top axial surface*(5)*	2.94E+02	0.0231	1.17E+01	0.0788	3.06E+02	0.0224
1 cm from cask bottom axial surface*(6)*	4.66E+02	0.0182	1.21E+02	0.0244	5.87E+02	0.0153

(1) Gamma and neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 86 mrem/hr gamma, 11 mrem/hr neutron for a total average dose rate of 97 mrem/hr.

(3) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 108 mrem/hr gamma, 57 mrem/hr neutron for a total average dose rate of 165 mrem/hr. Note that this bottom axial dose rate has no impact on the occupational exposure because no operations are performed near bottom axial location.

(4) The Side dose rates increase by 24% when loading 0.380 MTU FAs.

(5) The Top dose rates increase by 63% when loading 0.380 MTU FAs.

(6) The Bottom dose rates increase by 91% when loading 0.380 MTU FAs.

Table Z.5-19
Averaged Dose Rates for HSM-H Containing NUHOMS®-37PTH DSC

Dose Rate Location	Gamma Average (mrem/hr)	Gamma MCNP 1σ Error	Average Neutron (mrem/hr)	Neutron MCNP 1σ Error	Average Total (mrem/hr)	Total MCNP 1σ Error
HSM Roof[(1)]	10.7	0.02	0.14	0.02	10.8	0.02
HSM End (Side) Shield Wall Surface[(2)]	0.3	0.02	0.01	0.01	0.3	0.02
HSM Front[(1)]	11.5	0.09	0.05	0.05	11.5	0.09
HSM Back Shield Wall[(2)]	0.04	0.06	0.001	0.02	0.04	0.05

(1) These dose rates increase by 25% when loading 0.380 MTU FAs.
(2) These dose rates increase by 35% when loading 0.380 MTU FAs.

Table Z.5-20
Summary of Experimental Samples as a Function of Burnup Range

		Low Burnup (B < 45 GWd/MTU)		High Burnup (B > 45 GWd/MTU)	
Power Plant	**Reference**	**No of Samples**	**Range (GWd/MTU)**	**No of Samples**	**Range (GWd/MTU)**
Takahama-3	NUREG/CR-6968 Reference [5.17]	14	8.55 – 42.16	2	47.03 – 47.25
Three Mile Island - 1		10	22.80 – 44.8	9	50.10 – 55.70
Calvert Cliffs		3	27.35 – 44.34	-	-
Vandellós II	NUREG/CR-7013 Reference [5.19]	1	42.50	5	54.85 – 78.30
Gösgen	NUREG/CR-7012 Reference [5.18]	1	31.10	5	46.00 –70.30
GKN II		-	-	1	54.10
Total		29	-	22	-

Table Z.5-21
0.4 kW Design Basis HSM Source Term

Bounding Source at 380 kgU/FA: 45 GWD/MTU, 1.1 wt. %, after 36.4 years of cooling						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle (g/s)**	**In-core (g/s)**	**Plenum (g/s)**	**Top Nozzle (g/s)**
0.00e+00	to	5.00e-02	6.498E+09	3.054E+14	1.666E+10	4.571E+09
5.00e-02	to	1.00e-01	8.963E+08	9.062E+13	2.021E+09	6.245E+08
1.00e-01	to	2.00e-01	2.227E+08	4.896E+13	5.031E+08	1.527E+08
2.00e-01	to	3.00e-01	1.281E+07	1.510E+13	3.019E+07	8.792E+06
3.00e-01	to	4.00e-01	1.623E+07	9.904E+12	3.456E+07	1.058E+07
4.00e-01	to	6.00e-01	2.581E+07	7.212E+12	2.238E+07	2.479E+06
6.00e-01	to	8.00e-01	1.569E+09	6.913E+14	7.920E+09	1.320E+09
8.00e-01	to	1.00e+00	1.513E+09	3.701E+12	7.659E+09	1.281E+09
1.00e+00	to	1.33e+00	2.550E+11	7.817E+12	5.621E+11	1.771E+11
1.33e+00	to	1.66e+00	7.200E+10	1.227E+12	1.587E+11	5.001E+10
1.66e+00	to	2.00e+00	4.915E+01	2.508E+10	3.203E+01	6.144E-03
2.00e+00	to	2.50e+00	1.723E+06	1.353E+09	3.798E+06	1.197E+06
2.50e+00	to	3.00e+00	1.472E+03	1.270E+08	3.245E+03	1.022E+03
3.00e+00	to	4.00e+00	5.694E-06	3.727E+07	2.914E-05	4.690E-06
4.00e+00	to	5.00e+00	0.0	1.258E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	5.050E+06	0.0	0.0
6.50e+00	to	8.00e+00	0.0	9.906E+05	0.0	0.0
8.00e+00	to	1.00e+01	0.0	2.103E+05	0.0	0.0
Total Gamma, g/(sec*FA)			3.377E+11	1.181E+15	7.557E+11	2.351E+11
(1)Total Neutrons, n/(sec*FA)			3.650E+8			

(1) This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-keff) to account for subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where the dry keff = 0.25189 and bpf=1.152.

Table Z.5-22
0.7 kW Design Basis HSM Source Term

Bounding Source at 380 kgU/FA: 10 GWD/MTU, 0.7 wt. %, after 2.4 years of cooling						
E_{min}, MeV	***to***	***E_{max}, MeV***	***Bottom Nozzle (g/s)***	***In-core (g/s)***	***Plenum (g/s)***	***Top Nozzle (g/s)***
0.00e+00	*to*	*5.00e-02*	*3.728E+11*	*1.372E+15*	*4.980E+11*	*1.130E+11*
5.00e-02	*to*	*1.00e-01*	*3.286E+10*	*4.487E+14*	*7.140E+10*	*2.234E+10*
1.00e-01	*to*	*2.00e-01*	*1.658E+10*	*4.307E+14*	*2.284E+10*	*5.436E+09*
2.00e-01	*to*	*3.00e-01*	*1.084E+09*	*1.142E+14*	*1.310E+09*	*2.696E+08*
3.00e-01	*to*	*4.00e-01*	*4.739E+09*	*9.000E+13*	*3.874E+09*	*3.513E+08*
4.00e-01	*to*	*6.00e-01*	*4.634E+10*	*4.215E+14*	*3.066E+10*	*2.245E+08*
6.00e-01	*to*	*8.00e-01*	*2.923E+10*	*6.643E+14*	*2.092E+10*	*3.677E+08*
8.00e-01	*to*	*1.00e+00*	*4.351E+11*	*9.548E+13*	*7.813E+10*	*2.477E+11*
1.00e+00	*to*	*1.33e+00*	*9.341E+12*	*1.429E+14*	*2.069E+13*	*6.497E+12*
1.33e+00	*to*	*1.66e+00*	*2.638E+12*	*4.271E+13*	*5.843E+12*	*1.835E+12*
1.66e+00	*to*	*2.00e+00*	*4.586E+06*	*2.887E+12*	*9.972E+06*	*3.100E+06*
2.00e+00	*to*	*2.50e+00*	*6.313E+07*	*8.624E+12*	*1.398E+08*	*4.390E+07*
2.50e+00	*to*	*3.00e+00*	*5.393E+04*	*2.077E+11*	*1.195E+05*	*3.751E+04*
3.00e+00	*to*	*4.00e+00*	*1.590E-06*	*1.892E+10*	*8.137E-06*	*1.310E-06*
4.00e+00	*to*	*5.00e+00*	*0.0*	*3.327E+05*	*0.0*	*0.0*
5.00e+00	*to*	*6.50e+00*	*0.0*	*1.335E+05*	*0.0*	*0.0*
6.50e+00	*to*	*8.00e+00*	*0.0*	*2.617E+04*	*0.0*	*0.0*
8.00e+00	*to*	*1.00e+01*	*0.0*	*5.554E+03*	*0.0*	*0.0*
*Total Gamma, g/(sec*FA)*			*1.292E+13*	*3.835E+15*	*2.726E+13*	*8.722E+12*
*(1)Total Neutrons, n/(sec*FA)*			*9.631E+6*			

(1) This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where the dry k_{eff} = 0.39328 and bpf=1.414

Table Z.5-23
1.2 kW Design Basis HSM Source Term

Bounding Source at 380 kgU/FA: 19 GWD/MTU, 0.8 wt. %, after 2.4 years of cooling						
E_{min}, MeV	*to*	*E_{max}, MeV*	*Bottom Nozzle (g/s)*	*In-core (g/s)*	*Plenum (g/s)*	*Top Nozzle (g/s)*
0.00e+00	*to*	*5.00e-02*	*5.602E+11*	*2.163E+15*	*7.566E+11*	*1.733E+11*
5.00e-02	*to*	*1.00e-01*	*5.036E+10*	*7.055E+14*	*1.093E+11*	*3.422E+10*
1.00e-01	*to*	*2.00e-01*	*2.655E+10*	*6.610E+14*	*3.571E+10*	*8.322E+09*
2.00e-01	*to*	*3.00e-01*	*1.705E+09*	*1.805E+14*	*2.032E+09*	*4.123E+08*
3.00e-01	*to*	*4.00e-01*	*6.440E+09*	*1.413E+14*	*5.399E+09*	*5.376E+08*
4.00e-01	*to*	*6.00e-01*	*7.706E+10*	*9.031E+14*	*5.071E+10*	*2.451E+08*
6.00e-01	*to*	*8.00e-01*	*4.525E+10*	*1.398E+15*	*3.278E+10*	*6.434E+08*
8.00e-01	*to*	*1.00e+00*	*6.088E+11*	*2.547E+14*	*1.095E+11*	*3.466E+11*
1.00e+00	*to*	*1.33e+00*	*1.432E+13*	*2.397E+14*	*3.167E+13*	*9.955E+12*
1.33e+00	*to*	*1.66e+00*	*4.043E+12*	*7.353E+13*	*8.945E+12*	*2.811E+12*
1.66e+00	*to*	*2.00e+00*	*4.854E+06*	*4.797E+12*	*1.043E+07*	*3.228E+06*
2.00e+00	*to*	*2.50e+00*	*9.675E+07*	*1.216E+13*	*2.140E+08*	*6.726E+07*
2.50e+00	*to*	*3.00e+00*	*8.267E+04*	*3.589E+11*	*1.829E+05*	*5.747E+04*
3.00e+00	*to*	*4.00e+00*	*4.160E-06*	*3.295E+10*	*2.129E-05*	*3.426E-06*
4.00e+00	*to*	*5.00e+00*	*0.0*	*3.323E+06*	*0.0*	*0.0*
5.00e+00	*to*	*6.50e+00*	*0.0*	*1.334E+06*	*0.0*	*0.0*
6.50e+00	*to*	*8.00e+00*	*0.0*	*2.616E+05*	*0.0*	*0.0*
8.00e+00	*to*	*1.00e+01*	*0.0*	*5.554E+04*	*0.0*	*0.0*
*Total Gamma, g/(sec*FA)*			*1.974E+13*	*6.738E+15*	*4.172E+13*	*1.333E+13*
*(1)Total Neutrons, n/(sec*FA)*			*9.531E+7*			

(1)This is a "raw" source calculated with ORIGEN-ARP. Multiply it by bpf/(1-k_{eff}) to account for subcritical multiplication and an axial variation of burn-up profile in the active fuel region, where the dry k_{eff}= 0.34041 and bpf=1.403.

		Z4	Z4	Z4		
	Z4	Z3	Z3	Z3	Z4	
Z4	Z3	Z2	Z2	Z2	Z3	Z4
Z4	Z3	Z2	Z1	Z2	Z3	Z4
Z4	Z3	Z2	Z2	Z2	Z3	Z4
	Z4	Z3	Z3	Z3	Z4	
		Z4	Z4	Z4		

	Zone 1	Zone 2	Zone 3	Zone 4
Max. Decay Heat (kW/FA) (2)	0.40	0.40	0.7	1.2
No. of Fuel Assemblies (1)	1	8	12	16
Max. Decay Heat per Zone (kW) (2)	0.4	3.2	8.4	19.2
Max. Decay Heat per DSC (kW)	31.2(1)			

Note (1): The 37PTH DSC is limited to 30.0 kW.

Figure Z.5-1
37PTH DSC Bounding HLZC Used for Shielding Analysis

Z.9 Acceptance Tests and Maintenance Program

Background for this particular UFSAR chapter:

Beginning with CoC 1004 Amendment 13, which was incorporated into UFSAR Revision 14, Chapter Z.9, "Acceptance Tests and Maintenance Program," contained information which was incorporated by reference into the Technical Specifications (TS) associated with a particular amendment. It is known that certain general licensees reconcile the CoC 1004 UFSAR revisions provided to them to their loaded systems, pursuant to 10 CFR 72.48 and 10 CFR 72.212. In doing so they sometimes find the changed UFSAR portions incorporated by reference into the TS to be impossible to reconcile because the 10 CFR 72.48 regulation does not allow proposed activities which involve changes to the TS.

In order to facilitate this reconciliation process by general licensees, the following statements are provided, addressing the licensing basis for certain amendments, as they relate to certain UFSAR chapters which contain TS incorporated by reference. Additionally, so that the actual information is contained in the current CoC 1004 UFSAR, to facilitate the reconciliation by general licensees, the UFSAR Revision 14 and 15 versions of Chapter Z.9 are inserted and annotated in this part of the UFSAR. For clarity, this includes annotating the version of Chapter Z.9 directly associated with the latest UFSAR revision in which a change to Chapter Z.9 occurred.

- Systems loaded to CoC 1004 Amendment 13 have Technical Specifications incorporated by reference from UFSAR Revisions 14 and 15 Chapter Z.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 13.
- Systems loaded to CoC 1004 Amendment 14 have Technical Specifications incorporated by reference by FCN 721004-1575, which will be incorporated into UFSAR Revisions 16 and 17 Chapter Z.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 14.
- *Systems loaded to CoC 1004 Amendment 15 have Technical Specifications incorporated by reference from UFSAR Revision 18 Chapter Z.9. Changes made to that chapter in subsequent UFSAR revisions do not apply to Amendment 15.*

This version of Chapter Z.9 is associated with CoC 1004 Amendment 15 and is added from UFSAR Revision 18. Please see Page Z.9 Introduction - 1 for a discussion as to why certain versions of Chapter Z.9 are being maintained in the UFSAR.

Z.9 Acceptance Tests and Maintenance Program

Z.9.1 Acceptance Tests

The pre-operational testing requirements for the Standardized NUHOMS® system are given in Chapter 9, with the exceptions described in the following sections. The NUHOMS®-37PTH DSC has been enhanced to provide leaktight confinement and the basket includes an updated poison plate design. Additional acceptance testing of the NUHOMS®-37PTH DSC welds and of the poison plates are described.

Z.9.1.1 Visual Inspection

Visual inspections are performed at the fabricator's facility to ensure that the DSC, the transfer cask and the HSM conform to the drawings and specifications. The visual inspections include weld, dimensional, surface finish, and cleanliness inspections. Visual inspections specified by codes applicable to a component are performed in accordance with the requirements and acceptance criteria of those codes.

All weld inspection is performed using qualified processes and qualified personnel according to the applicable code requirements, e.g., ASME or AWS. Non-destructive examination (NDE) requirements for welds are specified on the drawings provided in Appendix Z.1; acceptance criteria are as specified by the governing code. NDE personnel are qualified in accordance with SNT-TC-1A [9.2].

The confinement welds on the DSC are inspected in accordance with ASME B&PV Code Subsection NB [9.1] including alternatives to ASME Code specified in Section Z.3.1.2.3.

DSC non-confinement welds are inspected to the NDE acceptance criteria of ASME B&PV Code Subsection NG or NF, based on the applicable code for the components welded.

Z.9.1.2 Structural

The DSC confinement boundary except the inner top cover/shield plug to the DSC shell weld is pressure tested at the fabricator's shop in accordance with ASME Article NB-6300. The test pressure is set between 16.5 to 18.0 psig for 37PTH DSC, which bounds the 1.1xDSC design pressure of 15 psig.

The inner top cover/shield plug to the DSC shell weld is also pressure tested between 16.5 to 18.0 psig for 37PTH DSC. This pressure test is performed at the field after the fuel assemblies are loaded in the DSC. This test is in accordance with the alternatives to the ASME Code specified in Section Z.3.1.2.3.

HSM-H reinforcement and concrete are tested as described in Chapter 3, Section 3.4.2.

Z.9.1.3 Leakage Tests

The DSC canister confinement boundary is tested using two procedures described below. Personnel performing the leakage test are qualified in accordance with SNT-TC-1A [9.2].

Procedure 1 is accomplished during fabrication:

Upon completion of all canister shell welding and attachment of the inner bottom cover plate to the shell, a temporary seal plate is placed over the open end of the DSC. A bag or other enclosure is placed around the outside of the entire DSC and it is filled with helium. The DSC cavity is evacuated and a helium leakage test is performed using a port in the seal plate. This test is used to show that the entire DSC confinement boundary tested is leak tight ($1x10^{-7}$ ref cm^3/s).

Procedure 2 of the testing occurs after the DSC has been loaded with fuel assemblies:

The DSC cavity has been dried, back filled with helium and the inner top cover plate and the vent and drain port cover plates have been welded in place. After these welds are completed, a temporary test cover is installed or the outer top cover plate is welded in place with at least the root pass of the full weld. The cavity between inner top cover plate and the temporary test cover or outer top cover plate is evacuated and a helium leakage test is performed using a test port in the temporary test cover or in the outer top cover plate. The leakage test thus includes the weld attaching the inner top cover plate to the canister shell, the vent and drain port cover plate welds and the base metal of the inner top cover plate and vent and drain port cover plates. The vent and drain ports are filled with helium prior to welding the vent and drain port covers. This test verifies that the tested welds and cover plates are leak tight ($1x10^{-7}$ ref cm^3/s).

Z.9.1.4 Components

The Standardized NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. No other components of the NUHOMS® system require testing, except as discussed in this chapter.

Z.9.1.5 Shielding Integrity

The transfer cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6″ × 6″ grid, the detector will encompass a 6″ × 6″ square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written

procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed.

The gamma and neutron shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

Z.9.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section Z.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section Z.9.1.7.6.

Z.9.1.7 Poison Acceptance

CAUTION

Sections Z.9.1.7.1 through Z.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 7) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated aluminum

(b) Boron carbide/aluminum metal matrix composite (MMC)

(c) BORAL®

The 37PTH DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content of these three types of materials is 0.0020 g/cm^2 for Borated Aluminum and MMC, and 0.0025 g/cm^2 for BORAL®.

References to metal matrix composites throughout this chapter are not intended to refer to BORAL®, which is described later in this section.

Z.9.1.7.1 **Borated Aluminum**

See the Caution in Section Z.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating primarily as a uniform fine dispersion of discrete AlB2 or TiB2 particles in the matrix of aluminum or aluminum alloy (other boron compounds, such as AlB12, can also occur). For extruded products, the TiB2 form of the alloy shall be used. For rolled products, either the AlB2, the TiB2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section Z.9.1.7.7. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Z.9.1.7.2 **Boron Carbide/Aluminum Metal Matrix Composites (MMCs)**

See the Caution in Section Z.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, molten metal infiltration, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding or produced by molten metal infiltration shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B4C particles in boron carbide shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 37PTH DSC, MMCs shall pass the qualification testing specified in Section Z.9.1.7.8, and shall subsequently be subject to the process controls specified in Section Z.9.1.7.9.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section Z.9.1.7.7. The specified acceptance testing assures that at any location

in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Z.9.1.7.3 **BORAL®**

See the Caution in Section Z.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns. The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of BORAL®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken from the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

Z.9.1.7.4 **Visual Inspections of Neutron Absorbers**

See the Caution in Section Z.9.1.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. Blisters shall be treated as non-conforming. For clad MMCs and for BORAL®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet. Material that does not meet these criteria shall be reworked, repaired, or scrapped.

Z.9.1.7.5 Other Visual Inspections Criteria (non-Technical Specifications)

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

Z.9.1.7.6 Acceptance Testing

Acceptance testing shall conform to ASTM E1225[1], ASTM E1461[2], or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, and may be reduced if the first five tests meet the specified minimum thermal conductivity. For cast products, the lot shall be defined by the

[1] ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"
[2] ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"

heat or ingot. For other products, the lot shall be defined as material produced in a single production campaign using the same heat or lots of aluminum and boron carbide feed materials.

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The measured thermal conductivity values shall satisfy the minimum required conductivities as specified in Section Z.4.3

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section Z.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

Z.9.1.7.7 Specification for Acceptance Testing of Neutron Absorber *Content*

Acceptance testing for neutron absorber content shall be performed by either neutron transmission or by B-10 volume density measurement.

CAUTION

Portions of Section Z.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 7) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Z.9.1.7.7.1 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

a) Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam up to 1.1 inch diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard. Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than the statistically derived minimum thickness from Z.9.1.7.7 a) or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

Z.9.1.7.7.2 *Specification for Acceptance Testing of Neutron Absorbers by B-10 Volume Density Measurement*

a) B-10 volume density measurement acceptance testing procedures shall be subject to approval by the certificate holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, as long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for B-10 volume density measurements shall be such that there is at least one density measurement for each 2000 square inches of final product in each lot.

Areal density is determined by measuring the B-10 volume density in test samples and converting the measured values to areal density. The method of measurement of B-10 volume density shall be subject to approval by the certificate holder. The method of measurement of B-10 volume density shall be qualified against neutron transmission testing. Results of the two test methods shall be compared and a penalty shall be derived to account for the performance based results of neutron transmission testing.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B-10 areal densities are determined by volume density as described above. The lower tolerance limit of B-10 volume density is then determined, defined as the mean value of B-10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6]. Finally, the minimum specified value of B-10 areal density is divided by the lower tolerance limit of B-10 volume density to arrive at the minimum plate thickness that provides the specified B-10 areal density.

Any plate that is thinner than the statistically derived minimum thickness from Z.9.1.7.7.2 a) or the minimum design thickness, whichever is greater, shall be treated as nonconforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.

Non-conforming material shall be evaluated for acceptance in accordance with the certificate holder's QA procedures.

Z.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

> ***CAUTION***
>
> *Portions of Section Z.9.1.7.8.4, and all of Section Z.9.1.7.8.5, are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 7) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

Z.9.1.7.8.1 Applicability and Scope

Metal matrix composites (MMCs) acceptable for use in the 37PTH DSC are described in Section Z.9.1.7.2.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section Z.9.1.7.9 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system. ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the Certificate Holder.

Z.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in Section Z.9.1.7.8.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section Z.9.1.7.8.5.

Z.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

The need for thermal damage and corrosion (hydrogen generation) testing shall be evaluated case-by-case based on comparison of the material composition and environmental conditions with previous thermal or corrosion testing of MMCs. Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for unclad MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for full density MMCs (clad or unclad) consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

Z.9.1.7.8.4 Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity

At least three samples, one each from approximately the two ends and middle of the qualification material run shall be subject to:

a) room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:

- **Minimum yield strength, 0.2% offset: 1.5 ksi**
- **Minimum ultimate strength: 5 ksi**
- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

b) Testing to verify more than 98% of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a means to be approved by the Certificate Holder. The maximum interconnected porosity is 0.5 volume %.

c) Delamination Testing of Clad MMC

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage. An example

3 Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998.

4 Boralyn testing submitted to the NRC under docket 71-1027, 1998.

5 ASTM B557, Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.

6 ASTM E290, Standard Methods for Bend Testing of Materials for Ductility

of such a test would be: (1) immerse a specimen at least 6 x 6 inches in water under pressure ≥30 psig for at least 24 hours, (2) place the specimen in a vacuum furnace preheated to at least 300°F and evacuate the furnace. Acceptance criterion: no blistering or delamination of the cladding.

Z.9.1.7.8.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

Uniformity of the boron distribution shall be verified either by:

a) **Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or**

b) **Quantitative testing for the B10 areal density, B10 density, the boron carbide weight fraction, *or the boron weight fraction* on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section Z.9.1.7.7, or by chemical analysis for boron carbide *or boron* content in the composite.**

Z.9.1.7.8.6 Qualification Report

Qualification report shall be prepared by, or subject to approval by the Certificate Holder.

Z.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION

Sections Z.9.1.7.9.1 and Z.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 7) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Z.9.1.7.9.1 **Applicability and Scope**

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section Z.9.1.7.8 is required, depending on the characteristics of the material that could be affected by the process change.

7 ASTM E94, Recommended Practice for Radiographic Testing

8 ASTM E142, Controlling Quality of Radiographic Testing

9 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

Z.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, reduce corrosion resistance, reduce the mechanical strength or ductility of the MMC.

Z.9.1.7.9.3 Identification and Control of Key Process Changes

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section Z.9.1.7.9.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that are established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average (d50) particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a magnesium-alloyed aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, and

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

Z.9.1.7.10 B_4C Linear Density Testing for Poison Rod Assemblies (PRAs)

The PRAs are shown in Figure Z.1-1, and additional physical requirements are listed in Table Z.2-5a. The B_4C poison is inserted into the stainless steel tubes shown in Figure Z.1-1. Table Z.2-5a specifies the minimum B_4C content per unit length in the axial direction of the rods for the various PRA designs. The minimum B_4C content per unit length is consistent with the criticality analysis (Section Z.6) with an additional 25% margin.

Pellets or powder representing each powder lot shall be tested per ASTM C751 [9.7] or ASTM C750 (Type 2) [9.8] (or equivalent). Density and diameter shall be measured to verify conformance to the specification requirements.

Deviations from the specified dimensions or density may be accepted, as long as the resulting minimum B_4C mass per unit length is maintained.

Justification for Durability of B_4C Pellets:
B_4C is essentially inert and will not be attacked even by hot hydrofluoric or nitric acids [9.9]. It is insoluble in water [9.10], resistant to steam at temperatures of 200 to 300 °C [9.10] and has a melting point of 2450°C [9.11]. Mechanically, B_4C is extremely hard (Mohs hardness of 9.3 vs. 10 for diamond) and is used in abrasion- and wear-resistant applications and in bullet-proof tiles.

It has a compressive strength of 398,000 psi. In the PRAs, the B_4C pellets are sealed within stainless steel. With this configuration, there is nothing that could cause the material to degrade.

In the unlikely event that a pellet was to crack or break, the total mass would be confined by the steel to the same dimensions.

The irradiation-induced swelling is due to neutron capture by the ^{10}B isotope. Using data from [9.12] and by determining the neutron absorption in the B_4C (^{10}B capture) from the shielding analyses, the swelling is determined to be negligible ≈ 0.00002 %. Finally, according to [9.12], the first intergranular cracks do not start to appear until fluences are 5.5 orders of magnitude greater than those calculated for 50 years operation.

Z.9.2 Maintenance Program

The NUHOMS®-37PTH system is a totally passive system and therefore will require little, if any, maintenance over the lifetime of the ISFSI. Typical NUHOMS®-37PTH system maintenance tasks will be performed in accordance with the UFSAR.

Z.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, 2004 Edition through 2006 Addenda.

9.2 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

9.3 Not Used.

9.4 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.5 "Aluminum Standards and Data, 2003," The Aluminum Association.

9.6 Natrella, "Experimental Statistics," Dover, 2005.

9.7 ASTM C751, "Standard Specification for Nuclear-Grade Boron Carbide Pellets."

9.8 ASTM C750, "Standard Specification for Nuclear-Grade Boron Carbide Powder."

9.9 The Merck Index, 9th edition, Merck & Co., 1976.

9.10 Grant (ed.), Hackh's Chemical Dictionary, 4th edition, McGraw-Hill, 1969.

9.11 Lipp, A., "Boron Carbide: Production, Properties, Application," Reprint from Technische Rundschau, Nos. 14, 28, 33 (1995) and 7 (1966).

9.12 *Stoto, T. et al., "Swelling and Microcracking of Boron Carbide Subjected to Fast Neutron Irradiations," Journal of Applied Physics, Vol. 68, No.7, October 1, 1990, pp. 3198-3206.*

Z.10.1 Occupational Exposure

The occupational exposure results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section Z.5.4.11. Loading 0.380 MTU fuel results in an increase in occupational exposure of 25%.

The expected occupational dose for placing a canister of spent fuel into dry storage is based on the operational steps outlined in Chapter 7, Table 7.4-1 of the UFSAR. The total exposure for the occupational dose due to placing a single NUHOMS® 37PTH DSC loaded with design basis fuel assemblies into storage is estimated to be 2 person-rem as summarized in Table Z.10-1. The calculated exposures are due mainly to the expected gamma dose rate during preparation for welding.

The NUHOMS® 37PTH system loading operations, the number of workers required for each operation, and the amount of time required for each operation are presented in Table Z.10-1. This information is used as the basis for estimating the total occupational exposure associated with one fuel load. The dose rates applicable for each operation are based on the results presented in Appendix U.5, Section U.5.4 for loading operations. Based on the NUHOMS®-37PTH system HSM-H dose rates in Appendix Z.5, Section Z.5.4, it is conservative to bound the NUHOMS®-37PTH system by the NUHOMS®-32PTH1 system. Engineering judgment and operational experience are used to estimate dose rates that were not explicitly evaluated. This evaluation assumes that a transfer trailer/skid with an integral ram is used for the DSC transfer operations. Licensees may elect to use different equipment and/or different procedures. Each licensee must evaluate any such changes in accordance with its ALARA program.

Unique steps are sometimes necessary at the individual site to load the canister, complete closure operations and place the canister in the HSM-H. Specifically, the licensee may choose to modify the sequence of operations in order to achieve reduced dose rates for a larger number of steps, with the end result of reduced total exposure. The only requirement is that the licensee practice ALARA with respect to the total exposure received for a loading campaign. These estimated durations, manloading, and dose rates are not limits.

The amount of time required to complete some operations as identified in Table Z.10-1 may be greater than the actual amount of time spent in a radiation field. The process of vacuum drying the DSC includes setting up the vacuum drying system (VDS), verifying that the VDS is operating correctly, evacuating the DSC cavity, monitoring the DSC pressure, and disconnecting the VDS from the DSC. Of these tasks, only setup and removal of the VDS require a worker to spend time near the DSC. The most time consuming task, evacuating the DSC, does not require anyone to be present near DSC at all. The total exposure calculated for each task is therefore not necessarily equal to the number of workers multiplied by the total time required, multiplied by a dose rate. The exposure estimation for each task correctly accounts for cases such as vacuum drying and assumes that good ALARA practices are followed.

Localized regions of elevated dose rates should be anticipated and minimized with good ALARA practices. Such regions exist due primarily to radiation streaming, including for example, streaming through the cask/DSC annulus, the ventilation paths in OS200 TC lid, and the DSC vent/siphon ports.

The results of the evaluations of the NUHOMS® 37PTH are presented in Table Z.10-1.

calculated by multiplying the current (in particles/cm^2/sec per mrem/hr) by the average dose rate of the face and by the area of the face.

2x10 Back-to-Back Array

A box that envelops the HSM-H array and shield walls, as modeled in MCNP, approximates the 2x10 back-to-back array of HSMs. The dimensions of the box also include the width of the HSM-H end shield walls. As discussed above, the total activity of each face of the box is calculated by multiplying the current by the average dose rate of the face and by the area of the face.

Two 1x10 Front-to-Front Arrays

A box that envelops the HSM-H array and shield walls, as modeled in MCNP, approximates the two 1x10 arrays of HSM-Hs. The dimensions of the box also include the width of the HSM-H end and back shield walls. As discussed above, the total activity of each face of the box is calculated by multiplying the current by the average dose rate of the face and by the area of the face. 72.48

The surface activities are summarized in Table Z.10-6 for the HSM-H.

Z.10.2.2 Dose Rates

Dose rates are calculated for distances of 6.1 meters (20 feet) to 600 meters from the edges of the two ISFSI designs by directly employing the NUHOMS® 32PTH1 dose rates presented in Appendix U.10, Section U.10.2.

The methodology employed in Appendix U.10 (including the use of scaling factors) is directly utilized herein. *The dose rate and annual dose results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section Z.5.4.11. Loading 0.380 MTU fuel results in an increase in annual dose of 25%.* AMD 15

The HSM-H MCNP site dose rate results are summarized in Tables Z.10-7 through Z.10-9. The front dose rates for the 2x10 configuration are provided in Table Z.10-7, the back dose rates for the 2-1x10 configuration are provided in Table Z.10-8, and the side dose rates for both ISFSI configurations are provided in Table Z.10-9.

The preceding analyses and the results provided in Figure Z.10-1 are intended to provide typical dose rates for the generic ISFSI layouts described in Section Z.10.2. They may not be applicable to an actual ISFSI. The written evaluations performed by a licensee for an actual ISFSI must consider the type and number of storage units, layout, characteristics of the irradiated fuel to be stored, site characteristics (e.g., berms, distance to the controlled area boundary, etc.), and reactor operations at the site in order to demonstrate compliance with 10CFR72.104.

Table Z.10-1
Occupational Exposure Summary, 37PTH System

Location	Task Description	# of Workers	Duration (hr)	Area Dose Rate (mrem/hr)	Total Exposure (person-mrem)
Auxiliary Building and Fuel Pool	Place the DSC into the transfer cask.	2	2	2	8
	Fill the cask/DSC annulus with clean water and install the inflatable seal.	3	1	2	6
	Fill the DSC cavity with water.	1	6	2	12
	Place the cask containing the DSC in the fuel pool.	5	0.5	2	5
	Verify and load the candidate fuel assemblies into the DSC.	3	5	2	30
	Place the top shield plug on the DSC.	2	1	2	4
	Remove the cask/DSC from the fuel pool and place them in the decon area.	5	0.5	2	5
		1	0.033	199	7
		1	0.667	146	98
Cask Decontamination Area	Decontaminate the outer surface of the cask.	1	1.75	146	256
		1	1	2	2
	Decontaminate the top region of the cask and DSC.	1	0.5	195	97
		1	0.5	64	32
	Drain water from the DSC.	1	0.083	199	17
	Remove cask/DSC annulus seal and set-up welding machine.	1	0.167	337	56
		1	0.75	87	65
		1	0.5	72	36
	Weld the inner top cover to the DSC shell and perform NDE (PT).	2	6	2	24
		1	0.33	169	56
	Drain the cask/DSC annulus and the DSC cavity.	1	0.25	87	22
		1	0.017	169	3
		1	0.5	2	1
	Vacuum dry and backfill the DSC with helium.	1	0.5	72	36
		2	30	2	120
	Helium leak test the shield plug weld.	2	1	2	4
	Seal weld the prefabricated plugs to the vent and siphon ports and perform NDE (PT).	1	0.5	87	44
	Fit-up the DSC outer top cover plate.	1	0.25	169	42
		1	0.5	87	44
	Weld the outer top cover plate to DSC shell and perform NDE (PT).	1	1	72	72
		1	0.167	169	28
		2	14	2	56
		1	0.333	169	56
	Install the cask lid.	2	0.667	93	124
Reactor/Fuel Building Bay	Ready the cask support skid and transport trailer for service.	2	2	2	8
	Place the cask onto the skid and trailer.	2	0.25	136	68
	Secure the cask to the skid.	1	0.25	136	34
ISFSI Site	Ready the cask support skid and transport trailer for service.	2	2	negligible	0
	Transport the cask to ISFSI.	6	1	negligible	0
	Position the cask in close proximity with the HSM.	3	1	negligible	0
	Remove the cask lid.	2	0.67	42	56
	Align and dock the cask with the HSM.	2	0.25	108	54
	Position and align ram with cask.	2	0.5	174	174
	Remove ram access cover plate.	1	0.083	562	47
	Transfer the DSC from the cask to the HSM.	3	0.5	negligible	0
	Lift the ram back onto the trailer and un-dock the cask from the HSM.	2	0.083	56	9
	Install HSM access door.	2	0.5	15	15
	Totals	**N/A**	**87**	**N/A**	**1934**

Total estimated dose is 2 person-rem per 37PTH canister load.

Total estimated dose increases by 25% when loading 0.380 MTU/FA fuel.

Z.11.2.3.2 Accident Analysis

There is no change to the missile impact evaluation presented in Appendix U, Section U.11.2.3.2.

Z.11.2.3.2.1 HSM-H/HSM-HS Missile Impact Analysis

There is no change to the missile impact evaluations presented in Appendix U, Section U.11.2.3.2.1.

Z.11.2.3.3 Accident Dose Calculations

The increase in the dose rates at the localized impact location following the missile impact accident is expected to be bounded by the dose rates at the HSM-H vents, calculated to be *600* mrem/hour in Appendix Z.5, Table Z.5-1 *(477.17 mrem/hr*1.25 ~ 600 mrem/hr)*, since the structural analysis results demonstrate that there is no full penetration. This represents an increase in the *roof centerline dose rate* by a factor greater than 20 and is conservative.

For the purpose of this calculation, it is conservatively assumed that the affected area is twice the area of impact ~ 1.6 ft^2. The surface area at the HSM-H front is 140 ft^2, at the HSM-H roof is 200 ft^2 and that at the HSM-H side is 280 ft^2. *The impact area, therefore, represents approximately 0.6% to 1.2% of the surface area of the HSM-H, and the average dose rate on the surface of the impacted HSM will not increase appreciably. This increase does not significantly affect the ISFSI site dose rates and the results from Section Z.10.2 for a 2x10 array of undamaged HSMs (specifically Table Z.10-7) can be utilized to determine the exposure from a damaged HSM. This method is conservative because the missile impact will affect at most a single HSM, while a 2x10 array has approximately 20 front and 20 roof vents. The total dose rate is then the dose rate of the damaged HSM summed with the dose rate of the undamaged HSMs in the array, or twice the dose rate of the undamaged array using the conservative assumptions outlined above.*

*The dose received by a person located 100 meters away from the ISFSI for the assumed 8-hour duration would be less than 5 mrem (2*8 hours*dose rate at 100m, 8.75E-02 mrem/hour*1.25 scaling factor) with a 2x10 array of HSMs. The dose to an offsite person located 500 meters away for the assumed 8-hour duration would be less than 0.01 mrem (2*8 hours*dose rate at 500m, 1.83E-04 mrem/hour*1.25 scaling factor) with a 2x10 array of HSMs.*

Z.11.2.3.4 Corrective Actions

There is no change to the corrective actions presented in Appendix U.11, Section U.11.2.3.4.

Z.11.2.4 Flood

This event is described in Chapter 8, Section 8.2.4.

Z.11.2.4.1 Cause of Accident

No change. See Chapter 8, Section 8.2.4.1.

Z.11.2.4.2 Accident Analysis

There is no change to the accident analysis presented for HSM-H and HSM-HS in Appendix U.11, Section U.11.2.4.2. The DSC is designed and tested to be leak tight to the criteria of ANSI N14.5 [11.2]. The stresses in the DSC due to the design basis flood are well below the allowable

The section below describes the additional analyses performed to demonstrate the acceptability of the system with the NUHOMS® 37PTH DSC.

Z.11.2.7.3 Accident Dose Calculations

There are no off-site dose consequences as a result of this accident. The only significant dose increase is that related to the recovery operation. Based on the results presented in Appendix Z.5, Table Z.5-1, the bounding average dose on HSM front or roof is 15.5 mrem/hr.

It is conservatively estimated that the on-site workers will receive an additional dose of no more than *160* (=15.5x8**1.25 MTU scaling factor*) mrem during the eight hour period which is the estimated duration that may be required for removal of debris from the inlet and outlet vent openings. These exposures are well within the limits of 10CFR72.106 for an accident condition.

Z.11.2.7.4 Corrective Action

No change. See Chapter 8, Section 8.2.7.4.

Z.11.2.8 DSC Leakage

The NUHOMS® 37PTH DSC is designed as a pressure retaining containment boundary to prevent leakage of contaminated materials. The analyses of normal, off-normal, and accident conditions have shown that no credible conditions can breach the DSC shell or fail the double seal welds at each end of the DSC. The NUHOMS® 37PTH DSC is designed and tested to be leak tight. Therefore DSC leakage is not considered a credible accident scenario. See Appendix Z.7 for additional details on the confinement evaluation.

Z.11.2.9 Accident Pressurization of DSC

Z.11.2.9.1 Cause of Accident

The bounding internal pressurization of the NUHOMS® 37PTH DSC is postulated to result from cladding failure of the spent fuel in combination with the transfer accident case with the loss of sunshield and liquid neutron shield in the transfer cask under extreme ambient temperature conditions of 117°F and maximum insolation, and the consequent release of spent fuel rod fill gas and free fission gas. The evaluation conservatively assumes that 100% of the fuel rods have failed.

Z.11.2.9.2 Accident Analysis

The pressure due to this case is evaluated in Appendix Z.4, Section Z.4.7. The maximum accident condition pressure calculated is 117.3 psig for the 37PTH DSC. The accident design pressure is conservatively assumed to be 140 psig in the structural load combinations presented in Appendix Z.2, Table Z.2-15 for 37PTH.

Z.11.2.9.3 Accident Dose Calculations

There is no increase in dose rates as a result of this event.