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Preliminary Content Evaluation of the North Anna High Burn-Up Sister Fuel Rod Segments for Transportation in the 10-160B and NAC-LWT

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Savannah River National Laboratory
Savannah River Nuclear Solutions
Savannah River Site Aiken, SC 9808

REVIEWS AND APPROVALS

Prepared by:

Edward T. Ketusky
Package Technology and Pressurized Systems
Savannah River National Laboratory

Reviewed by:

Jeffery L. England
Package Technology and Pressurized Systems
Savannah River National Laboratory

Approved by:

Jeffery L. England
Package Technology and Pressurized Systems, Program Manager
Savannah River National Laboratory

ABSTRACT

The U.S. Department of Energy's (DOE's) Used Fuel Disposition Campaign (UFDC) Program has transported high-burnup nuclear sister fuel rods from a commercial nuclear power plant for purposes of evaluation and testing. The evaluation and testing of high-burnup used nuclear fuel is integral to DOE initiatives to collect information useful in determining the integrity of fuel cladding for future safe transportation of the fuel, and for determining the effects of aging, on the integrity of UNF subjected to extended storage and subsequent transportation. The UFDC Program, in collaboration with the U.S. Nuclear Regulatory Commission and the commercial nuclear industry, has obtained individual used nuclear fuel rods for testing. The rods have been received at Oak Ridge National Laboratory (ORNL) for both separate effects testing (SET) and small-scale testing (SST). To meet the research objectives, testing on multiple 6 inch fuel rod pins cut from the rods at ORNL, will be performed at Pacific Northwest National Laboratory (PNNL). Up to 10 rod equivalents will be shipped. Options were evaluated for multiple shipments using the 10-160B (based on 4.5 rod equivalents) and a single shipment using the NAC-LWT.

Based on the original INL/Virginia Power transfer agreement the rods are assumed to 152 inches in length with a 0.374-inch diameter.

This report provides a preliminary content evaluation for use of the 10-160B and NAC-LWT for transporting those fuel rod pins from ORNL to PNNL. This report documents the acceptability of using these packagings to transport the fuel segments from ORNL to PNNL based on the following evaluations:

- Enrichment
- A2 Evaluation
- Pu-239 FGE Evaluation
- Heat Load
- Shielding (both gamma and neutron)
- Content Weight/Structural Evaluation

CONTENTS

ACRONYMS

CoC	Certificate of Compliance
DOE	U.S. Department of Energy
FGE	(Pu-239) Fissile Gram Equivalent
GWd/MTU	GigaWatt-days per Metric Ton of Uranium
HAC	Hypothetical Accident Condition
HBU	High-BurnUp nuclear fuel
HWCSFR	Hypothetical Worst-Case Sister Fuel Rod
ID	Identification
MWd/MTU	MegaWatt-days per Metric Ton of Uranium
NAC-LWT	NAC-Legal Weight Truck
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PNNL	Pacific Northwest National Laboratory
PWR	Pressurized Water Reactor
RH-TRU	Remote Handled Transuranic (waste)
SAR	Safety Analysis Report
SARP	Safety Analysis Report for Packaging
SET	Separate Effects Testing
SST	Small Scale Testing
SRNL	Savannah River National Laboratory
TRU	Transuranic
UFDC	Used Fuel Disposition Campaign
UNF	Used Nuclear Fuel

ASSUMPTIONS

1. ORNL will be shipping 4.5 rod equivalents to PNNL for 10-160B option and 10 rod equivalents for the NAC-LWT option.
2. Each rod equivalent is 152 inches in length.
3. Each pin is 0.374-inch diameter.
4. Each pin is 6.0-inches in length.
5. Each pin is encased in a Swagelok tube not exceeding 0.75-inch in overall diameter.
6. The cooling time for all pins would be as low as 5 years.
7. The initial enrichment of all pins would be 4.45 wt% U-235.
8. The burnup for all pins are 58,000 MWd/MTU.

1. INTRODUCTION

The U.S. Department of Energy's (DOE's) Used Fuel Disposition Campaign (UFDC) Program has transported high-burnup nuclear fuel rods (HBU, burnup exceeding 45 gigawatt-days per metric ton of uranium [GWd/MTU]) from commercial nuclear power plants for purposes of evaluation and testing. The evaluation and testing of high-burnup used nuclear fuel (UNF) is integral to DOE initiatives to collect information useful in determining the integrity of fuel cladding for future safe transportation of the fuel, and for determining the effects of aging on the integrity of UNF subjected to extended storage and subsequent transportation. This need for additional research is based on the limited information currently available on the properties of high-burnup fuel, and because much of the fuel discharged from today's reactors exceeds this burnup threshold. As the burnup of fuel increases, a number of changes occur that may affect the performance of the fuel, cladding, and assembly hardware in storage and transportation. These changes include: 1) increased cladding corrosion layer thickness; 2) increased cladding hydrogen content; 3) increased cladding creep strains; 4) increased fission gas release, and 5) the formation of the high-burnup structure at the surface of the fuel pellets.

The UFDC Program, in collaboration with the U.S. Nuclear Regulatory Commission and the commercial nuclear industry, obtained individual used nuclear fuel rods for testing. HBU-UNF rods have been received at Oak Ridge National Laboratory (ORNL) for both separate effects testing (SET) and small-scale testing (SST). To meet UFD's research objectives, testing on multiple fuel rod pins (i.e., segmented rods) will be performed at Pacific Northwest National Laboratory (PNNL).

This report provides a preliminary content evaluation for use of the 10-160B and NAC-LWT for transporting those HBU UNF segments from ORNL to PNNL.

1.1 Used Fuel Rod Segments

The evaluation of the 10-160B used to transport high-burnup fuel rod segments are based on assuming a hypothetical bounding fuel rod, based on the a 17x17 pressurized water reactor (PWR) fuel assembly from the North Anna. Based on the original INL/Virginia Power transfer agreement the rods are assumed to 152 inches in length with a 0.374-inch diameter, having a maximum diameter of ¾" placed inside additional tubing. The length of the HBU-UNF pins would be approximately 6-inches long.

To bound the radiation and radionuclide contents for a single package in a shipment, the evaluations assumed the burnup of the fuel in each individual segment could be as high as the worst-case pin from the sister pins being evaluated (58,000 MWd/MTU). The cooling time would be as low as 5 years, and the initial enrichment of the fuel would be 4.45 wt% ^{235}U. These characteristics are representative of the "worst-case" sister pin in the testing program.

The current number of rod equivalents planned to be used for evaluation at PNNL is estimated as 4.5 rods.

2. 10-160B CASK DESCRIPTION

This section provides an overview of the 10-160B. Table 2.1 provides an overview of availability and applicable Certificate of Compliance (COC) (Ref. 1) Content Limitations

Table 2.1 10-160B Availability and Content Limits

Cask	Package ID Number	CoC Expiration Date	Packages Available	CoC Content Limits
HBU Fuel Segments				
10-160B	USA/9204/B(U)F–96 (DOE 2012)	DOE, 12/31/2014 - under timely renewal (Ref. 1) NRC, 10/31/2015 (Ref. 2)	2 casks in service one owned by DOE	<5 wt% enriched < 3,000 A2s < 325 of Pu-239 <200 thermal watts <14,250 pounds of content • TRU special form fissile material in secondary containers, • RH-TRU, radioactive sources, • TRU material in secondary containers

CoC = certificate of compliance, DOE = U.S. Department of Energy, ID = identification, NRC = U.S. Nuclear Regulatory Commission, RH=remote handled, TRU = transuranic

The 10-160B cask is a cylindrical carbon steel and lead shielded shipping cask designed to transport fissile and non-fissile radioactive waste material. The cask is transported in the upright position and is equipped with steel encased, rigid polyurethane foam impact limiters on the top and bottom (Figure 2.1). The cask's approximate dimensions, shielding, and weight are listed in Table 2.2.

Figure 2.1. 10-160B Cask

Table 2.2 10-160B Cask Dimensions (Ref. 1&2)

Cask Part Description	Dimension
Cask height	88 in.
Cask outer diameter	78½ in.
Cask cavity height	77 in.
Cask cavity diameter	68 in.
Overall package height, with impact limiters	130 in.
Overall package diameter, with impact limiters	102 in.
Lead shielding thickness	1 7/8 in.
Gross weight (package and contents)	72,000 lb.
Maximum total weight of contents, shoring, secondary containers, and optional shield insert	14,500 lb.

The cask body consists of a 1 1/8-in.-thick carbon steel (ASME SA516 or SA537) inner shell, a 1 7/8-in.-thick lead gamma shield, and a 2-in.-thick carbon steel outer shell (ASME SA516). The inner and outer shells are welded to a 5½-in.-thick carbon steel bottom plate. The cask cavity has an optional 11-gauge stainless steel liner. A 12-gauge stainless steel thermal shield surrounds the cask outer shell in the region between the impact limiters. The impact limiters are secured to each other around the cask by eight ratchet binders.

The cask lid is a 5½-inch thick carbon steel plate, and has a 31-inch diameter opening equipped with a secondary lid. The primary lid is sealed with a double elastomer O-ring and 24 equally spaced 1 3/4-in.-diameter bolts. The secondary lid is 46 inches in diameter, centered within the primary lid, and sealed to the primary lid by a double elastomer O-ring and 12 equally spaced 1 3/4-inch diameter bolts. The space between the double O-ring seals is provided with a test port for leak testing the primary and secondary lid seals. The optional cask drain and vent ports are sealed with a plug and an O-ring seal.

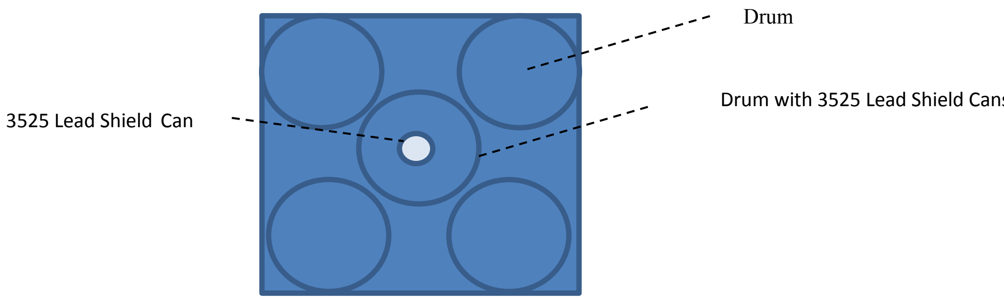
The package is equipped with four tie-down lugs welded to the cask outer shell. Two lifting lugs and two redundant lifting lugs are removed during transport. The lid is equipped with three lifting lugs which are covered by the top impact limiter and rain cover during transport.
An optional carbon steel shield insert may be used within the cask cavity and is installed in the DOE-owned cask (Ref. 1).

Note: The DOE owned 10-160B cask has transported transuranic waste that including irradiated nuclear fuel debris and high-level-radioactive-waste-glass debris from the Battelle West Jefferson, Ohio facility to laboratories in Hanford and Savannah River (Ref. 3).

3. ASSUMED CONFIGURATION

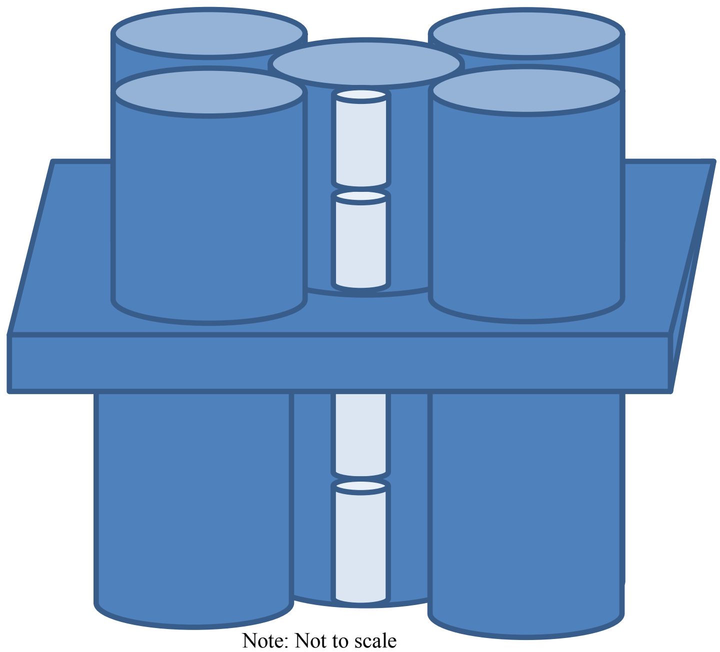
Based on the need to ship 4.5 rod equivalents, the desire to minimize the number of shipments, and the A2 limit, Section 4.2 determines that 4 shipments will be required ((2,570 A2s per rod from Section 4.2 x 4.5 rods)/3,000 A2 per 10-160B ≈3.9, which is rounded to 4 shipments. Optimization of load planning based on actual pin data may minimize shipments to 3.

Inside the 10-160B, the configuration will be a ten pack of 55-gallon drums, with each five pack separated by a pallet. Only the center drum in each five pack is used for holding the pins and the other drums are dunnage to meet structural load requirements. Inside the center drum, two 3525 Lead Shield Cans shown in Attachments 1-3 are stacked on top of each other and used for holding the pins. This equates to a total of four 3525 Lead Shield Cans per 10-160B. See Figure 3.1 for a diagram of the five pack and Figure 3.2 for the ten pack.



Note: not to scale

Figure 3.1 –Five Pack



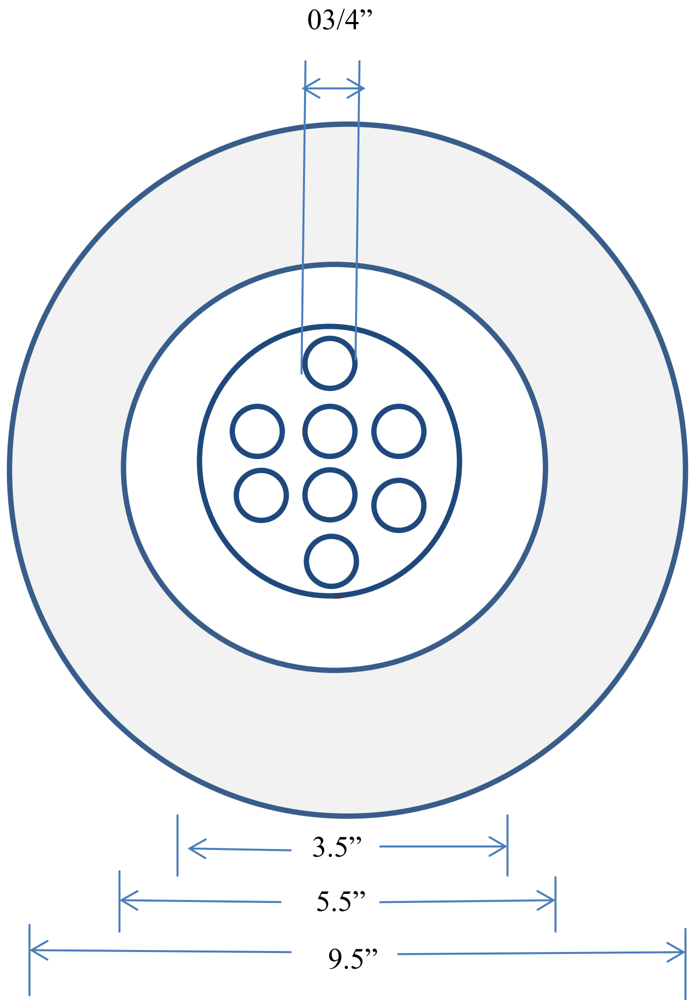
Note: Not to scale

Figure 3.2 – Ten Pack Configuration

The outside dimensions of the 3525 Lead Shield Cans are 15-7/8-inch tall x 9.5-inch diameter (See Attachments 1 through 3). Accounting for the 2" of lead, the inner cavity is approximately 11-7/8 tall by 5.5-inch diameter. Since the pins are in Swagelok tubes, the height of a 3525 Lead Shield Can is assumed to ensure the pins are not vertically stacked on top of each other inside of a 3525 Lead Shield Can. A convenience can be used inside the shield cans to hold the pins.

With 4.5-rod equivalents of 152 inches per rod, a total equivalent length of 684 inches will need to be shipped. With four shipments, an equivalent length of 171 inches (1.125 Rod Equivalents) would need to be shipped per 10-160B transport. With only 23.5 inches of usable length in four 3525 Lead Shield Cans, up to eight pins will be placed into each 3525 Lead Shield Can. With each fuel rod having a 0.374-inch diameter, it is assumed the pins are placed into outer tubes, with an outer ¾" diameter.

Figure 3.3 confirms up to 8 pins with ¾" diameter will fit inside the 3525 Lead Shield Can, while still allowing greater than 1" of additional lead to be added to the inside of the can, surrounding the pins.



Note: Not to Scale

Figure 3.3- Space Available for Additional Lead Shielding if Required

4. PRELIMINARY CONTENT EVALUATIONS

4.1 Enrichment

Criteria: Enrichment must be less than 5%
From the applicable North Anna fuel nuclide spreadsheets, the enrichment for the North Anna Fuel is 4.55-wt%. The COC (Ref. 1) limits the content to 5-wt%, therefore, the enrichment is acceptable.

4.2 A_2 Evaluation

Criteria: Must contain < 3,000 A2s
The active fuel rod height is estimated as 152 inches (Ref. 4). Using the maximum grams of each radionuclide identified as being associated with the North Anna fuel, using years 2016 to 2031 a Hypothetical Worst-Case Fuel Rod (HWCFR) was determined, as shown in Table 3.1.

Table 3.1 Mass of Hypothetical Worst-Case Sister Fuel Rod Constituents

Nuclide	grams	Nuclide	grams	Nuclide	grams	Nuclide	grams	Nuclide	grams	Nuclide	grams	Nuclide	grams
Ge-72	3.13E-5	As-75	3.91E-4	Pu-237	1.96E-19	Ru-106	1.16E-2	Pr-141	3.39E+0	Am-243	8.04E-1	Pa-233	6.22E-8
Sr-89	9.31E-14	Y-90	3.22E-4	Es-253	6.32E-40	Sn-120	1.66E-2	Eu-151	1.01E-2	Kr-85	5.75E-2	Cm-244	2.18E-1
Ru-101	2.37E+0	Rh-103	1.35E+0	Se-77	2.39E-3	Te-130	1.20E+0	Ho-165	5.31E-4	Mo-96	2.05E-1	Sr-86	2.96E-3
Sn-114	2.04E-7	Sn-115	8.71E-4	Zr-91	1.69E+0	Sm-150	9.62E-1	Pu-244	3.50E-4	Pd-110	2.03E-1	Tc-99	2.26E+0
Te-126	2.28E-3	I-127	1.55E-1	Sn-117	1.78E-2	Dy-162	1.16E-3	Kr-82	2.90E-3	Sn-124	3.29E-2	Cd-112	5.04E-2
Ba-135	3.36E-3	Ba-136	9.83E-2	Te-129m	2.39E-23	Pu-241	3.66E+0	Nb-95	9.11E-11	Th-230	4.42E-5	Sb-125	6.55E-3
Pm-147	9.94E-2	Pm-148	1.26E-19	Ba-138	4.01E+0	He-3	1.49E-4	Pd-108	6.00E-1	Cm-241	4.76E-27	Ba-134	8.60E-1
Gd-156	4.48E-1	Gd-158	9.28E-2	Tb-160	1.70E-12	Kr-80	1.33E-6	Sn-123	3.31E-8	Rb-85	3.61E-1	Nd-144	4.12E+0
U-236	1.15E+1	Np-237	1.84E+0	Pu-238	8.95E-1	Zr-94	2.28E+0	Xe-131	1.12E+0	Mo-97	2.45E+0	Gd-154	1.15E-1
Bk-249	6.63E-10	Cf-251	3.84E-8	Se-78	6.98E-3	Pd-106	1.53E+0	Ce-142	3.44E+0	Cd-110	2.33E-1	U-232	1.11E-5
Ge-73	6.85E-5	Ge-76	1.04E-3	Zr-92	1.86E+0	Sb-121	1.59E-2	Sm-152	2.45E-1	Sb-124	2.38E-14	Cm-245	4.22E-2
Y-89	1.28E+0	Zr-90	9.64E-1	Sn-118	1.69E-2	Ce-140	3.81E+0	Er-166	1.88E-4	Cs-133	3.20E+0	Rb-87	7.02E-1
Ru-102	2.68E+0	Ru-104	2.00E+0	I-129	5.19E-1	Dy-163	1.02E-3	Am-241	4.15E+0	Eu-153	3.73E-1	Ru-99	3.97E-4
Cd-115m	3.02E-18	Cd-116	1.98E-2	La-139	3.73E+0	Pu-242	2.23E+0	Kr-83	1.13E-1	Th-232	2.17E-5	Cd-113	4.00E-4
Xe-126	5.08E-7	Te-128	3.01E-1	Sm-149	1.01E-2	He-4	2.20E-2	Mo-95	2.18E+0	Cm-242	2.17E-5	Te-125	4.30E-2
Xe-136	7.10E+0	Cs-137	3.08E+0	Dy-160	1.56E-3	Br-81	6.20E-2	Cd-108	1.60E-6	Kr-86	5.08E-1	Nd-145	1.87E+0
Sm-147	6.13E-1	Pm-148m	2.00E-17	Pu-239	1.46E+1	Nb-94	5.98E-6	Sb-123	2.00E-2	Mo-98	2.57E+0	Eu-155	1.20E-2
U-237	1.14E-7	Tb-159	1.07E-2	Br-79	8.31E-6	Pd-107	8.72E-1	Eu-152	3.55E-5	Te-124	1.67E-3	U-233	2.90E-5
Cf-249	1.84E-7	Pu-236	2.29E-6	Zr-93	2.06E+0	Sn-122	2.11E-2	Er-167	7.48E-6	Xe-134	4.72E+0	Cm-246	5.02E-3
Ge-74	1.77E-4	Cf-252	1.05E-9	Pd-105	1.44E+0	Xe-130	3.04E-2	Am-242m	2.97E-3	Nd-143	2.31E+0	Sr-87	1.82E-5
Sr-90	1.27E+0	Se-76	2.49E-5	Sn-119	1.63E-2	Ce-141	6.24E-20	Kr-84	3.26E-1	Sm-154	1.39E-1	Mo-100	2.92E+0
Ru-103	8.29E-17	Y-91	5.92E-12	Xe-129	2.03E-4	Sm-151	3.73E-2	Zr-96	2.43E+0	Pa-231	1.74E-6	In-113	4.87E-6
In-115	4.51E-3	Pd-104	1.06E+0	Nd-150	5.79E-1	Dy-164	3.43E-4	Ag-109	2.87E-1	Cm-243	1.65E-3	Sn-126	7.19E-2
Te-127m	2.13E-9	Sn-116	8.42E-3	Dy-161	1.76E-3	Pu-243	3.66E-15	Te-123	2.65E-5	Rb-86	1.6E-36	Nd-146	2.31E+0
Nd-148	1.16E+0	Xe-128	1.87E-2	Pu-240	5.75E+0	Se-82	9.88E-2	Xe-132	3.74E+0	Cd-111	1.03E-1	Gd-155	3.01E-2
Gd-157	6.41E-4	Ba-137	2.46E+0	H-3	1.09E-4	Zr-95	7.56E-11	Nd-142	9.21E-2	Cs-134	8.09E-2	U-234	4.94E-1
U-238	1.62E+3	Sm-148	5.69E-1	Se-80	3.85E-2	Ag-107	4.03E-6	Gd-152	1.04E-4	Ce-144	6.46E-3	Cm-247	1.03E-4
Cf-250	1.99E-8	Gd-160	4.57E-3	Nb-93	2.22E-5	Te-122	1.91E-3	Bi-209	2.76E-11	Eu-154	6.39E-2	Sr-88	9.59E-1
Ru-100	5.18E-1	Cd-114	5.73E-2	Sb-126	1.48E-9	Cs-135	1.41E+0	U-235	2.03E+1	Cm-248	1.08E-5		

Table 3.2 shows the corresponding content in terms of A2s.

Table 3.2 A2s In Hypothetical Worst-Case Sister Fuel Rod

Nuclide	A2	Nuclide	A2
Sr-89	1.69E-10	Xe-131	7.85E+2
Pm-147	1.71E+0	Am-241	5.22E+2
U-236	4.66E-3	Eu-152	2.37E-4
Bk-249	1.31E-7	Am-242m	1.10E+0
Cd-115m	5.39E-15	Am-243	5.95E+0
Cf-249	3.42E-5	Kr-85	8.30E-2
Sr-90	2.19E+1	Th-230	3.44E-5
Ru-103	4.91E-13	Cm-241	3.00E-24
Te-127m	1.43E-6	Sb-124	2.53E-11
Cf-250	4.05E-5	Cm-242	2.65E-1
Np-237	2.41E-2	Pa-231	7.44E-6
Cf-251	3.23E-6	Cm-243	3.19E+0
Cs-137	1.67E+1	Rb-86	9.26E-33
Pm-148m	2.21E-14	Cs-134	5.53E+0
Pu-236	1.50E-2	Ce-144	3.83E+0
Cf-252	3.77E-4	Eu-154	1.04E+0
Y-91	9.25E-9	Pa-233	6.87E-5
Pu-237	4.35E-18	Cm-244	3.26E+2
Tb-160	1.17E-9	Tc-99	1.60E-3
Pu-238	5.64E+2	Sb-125	2.42E-1
Pu-239	3.35E+1	U-232	9.00E-3
Pu-240	4.90E+1	Cm-245	2.99E-1
Pu-241	2.29E+2	Eu-155	7.24E-2
Pu-242	3.23E-1	U-233	1.76E-6
Nb-94	5.98E-8	Cm-246	6.49E-2
Ce-141	1.09E-16	Sn-126	1.83E-4
Sm-151	3.59E-3	U-234	1.92E-2
Zr-95	7.22E-8	Cm-247	3.53E-7
Pu-244	2.34E-7	Sb-126	1.13E-5
Nb-95	1.32E-7	Cs-135	6.77E-5
Sn-123	1.70E-5	Cm-248	5.57E-6
		Total A2s	2.57E+3

The total A2s associated with Table 3.2 is approx. 2,570 A2s with the content limited to 3,000 A2s. It is expected four shipments would be required with 1.125 rod equivalents per shipment ((2,570 A2s x 4.5 rod equivalents)/3,000 A2s ≈ 3.9 shipments). Optimization of load planning based on actual pin data may minimize shipments to 3.

4.3 Criticality Evaluation

Criteria: < 325gram equivalent of Pu-239

Based on using the content of Table 3.1, Table 3.3 shows the Pu-239 Fissile Gram Equivalents (FGE) for a sole Hypothetical Worst-Case Fuel Rod.

Table 3.3 Pu-239 FGE for Hypothetical Worst-Case Sister Fuel Rod

Nuclide	Mass grams	Mass eq factor	Pu-239 FGE
Cf-249	1.84E-7	4.50E+1	8.26E-6
Cf-251	3.84E-8	9.00E+1	3.45E-6
Pu-239	1.46E+1	1.00E+0	1.46E+1
Pu-241	3.66E+0	2.25E+0	8.24E+0
Am-241	4.15E+0	2.25E+0	9.33E+0
Am-242m	2.97E-3	3.46E+1	1.03E-1
Cm-243	1.65E-3	5.00E+0	8.27E-3
Cm-245	4.22E-2	1.50E+1	6.33E-1
U-233	2.90E-5	9.00E-1	2.61E-5
Cm-247	1.03E-4	5.00E-1	5.13E-5
U-235	2.03E+1	6.50E-1	1.32E+1
Total Pu-239 FGE Equivalent			4.61E+1

Based on an individual rod having approximately 50 FGE, the 4.5 rods would contain only 225 FGE, which is less than the limit for one 10-160B. The expected FGE for 1.125 rod equivalents is approximately 62.5 FGE. Therefore, the 10-160B Pu-239 FGE limit will not be approached.

4.4 Heat Loads

Criteria: Thermal Loading must be < 200 watts
Thermal load associated with each of the nuclides in the Hypothetical Worst-Case Sister Fuel Rod is shown in Table 3.4, with a total thermal wattage of 4.47 watts determined per rod equivalent.

Table 3.4 Thermal Heat Load Estimate for Hypothetical Worst-Case Sister Fuel Rod

Nuclide	g	Ci	w/Ci	watts	Nuclide	g	Ci	w/Ci	watts
Sr-89	9.31E-14	2.70E-9	3.46E-3	9.34E-12	Sm-151	3.73E-2	9.70E-1	7.41E-4	7.19E-4
Pm-147	9.94E-2	9.24E+1	3.67E-4	3.39E-2	Zr-95	7.56E-11	1.59E-6	5.96E-3	9.47E-9
U-236	1.15E+1	7.46E-4	2.66E-2	1.99E-5	Cm-248	1.08E-5	4.52E-8	2.76E-2	1.25E-9
Bk-249	6.63E-10	1.06E-6	1.96E-04	2.08E-10	U-235	2.03E+1	4.46E-5	2.71E-2	1.21E-6
Sm-147	6.13E-1	1.41E-8	1.37E-2	1.93E-10	Pu-244	3.50E-4	6.31E-9	2.71E-2	1.71E-10
Cf-249	1.84E-7	7.53E-7	3.74E-2	2.82E-8	Nb-95	9.11E-11	3.55E-6	4.79E-3	1.70E-8
Sr-90	1.27E+0	1.78E+2	1.16E-3	2.06E-1	Sn-123	3.31E-8	2.72E-4	3.14E-3	8.53E-7
Ru-103	8.29E-17	2.65E-12	3.53E-3	9.37E-15	Am-241	4.15E+0	1.41E+1	3.43E-2	4.84E-1
Te-127m	2.13E-9	2.00E-5	5.52E-4	1.11E-8	Eu-152	3.55E-5	6.39E-3	7.64E-3	4.88E-5
U-238	1.62E+3	5.52E-4	2.49E-2	1.38E-5	Am-242m	2.97E-3	2.97E-2	4.05E-4	1.20E-5
Cf-250	1.99E-8	2.19E-6	3.63E-2	7.94E-8	Am-243	8.04E-1	1.61E-1	3.15E-2	5.06E-3
Y-90	3.22E-4	1.74E+2	5.54E-3	9.62E-1	Kr-85	5.75E-2	2.24E+1	1.50E-3	3.36E-2
Np-237	1.84E+0	1.30E-3	2.88E-2	3.75E-5	Th-230	4.42E-5	9.29E-7	2.77E-2	2.57E-8
Cf-251	3.84E-8	6.14E-8	3.68E-2	2.26E-9	Sb-124	2.38E-14	4.05E-10	1.33E-2	5.38E-12
Cs-137	3.08E+0	2.68E+2	1.01E-3	2.70E-1	Th-232	2.17E-5	2.39E-12	2.38E-2	5.67E-14
Pm-148m	2.00E-17	4.20E-13	1.28E-2	5.37E-15	Cm-242	2.17E-5	7.15E-2	3.59E-2	2.57E-3
Pu-236	2.29E-6	1.21E-3	3.42E-2	4.15E-5	Pa-231	1.74E-6	8.19E-8	2.97E-2	2.43E-9
Cf-252	1.05E-9	5.69E-7	3.52E-2	2.00E-8	Cm-243	1.65E-3	8.60E-2	3.61E-2	3.10E-3
Y-91	5.92E-12	1.48E-7	3.60E-3	5.33E-10	Cs-134	8.09E-2	1.05E+2	1.02E-2	1.07E+0
Sb-126	1.48E-9	1.24E-4	1.84E-2	2.28E-6	Ce-144	6.46E-3	2.07E+1	6.58E-4	1.36E-2
Te-129m	2.39E-23	7.15E-19	1.80E-3	1.29E-21	Eu-154	6.39E-2	1.66E+1	9.08E-3	1.51E-1
Cs-135	1.41E+0	1.69E-3	3.32E-4	5.61E-7	Pa-233	6.22E-8	1.31E-3	2.36E-3	3.08E-6
Tb-160	1.70E-12	1.87E-8	9.24E+1	1.73E-6	Cm-244	2.18E-1	1.76E+1	3.44E-2	6.06E-1
Pu-238	8.95E-1	1.52E+1	3.26E-0	4.96E-1	Tc-99	2.26E+0	3.85E-2	5.01E-4	1.93E-5
I-129	5.19E-1	9.35E-5	4.77E-4	4.46E-8	Sb-125	6.55E-3	6.55E+0	3.37E-3	2.21E-2
Pu-239	1.46E+1	9.06E-1	3.02E-2	2.74E-2	U-232	1.11E-5	2.43E-4	3.15E-2	7.65E-6
Zr-93	2.06E+0	5.16E-3	7.29E-7	3.76E-9	Cm-245	4.22E-2	7.18E-3	3.33E-2	2.39E-4
Pu-240	5.75E+0	1.32E+0	3.06E-2	4.04E-2	Rb-87	7.02E-1	6.04E-8	6.58E-4	3.97E-11
Ru-106	1.16E-2	3.84E+1	5.95E-4	2.28E-2	Eu-155	1.20E-2	5.87E+0	7.59E-4	4.45E-3
Pu-241	3.66E+0	3.66E+2	3.20E-5	1.17E-2	U-233	2.90E-5	2.81E-7	2.86E-2	8.03E-9
Pu-242	2.23E+0	8.71E-3	2.90E-2	2.53E-4	Cm-246	5.02E-3	1.56E-3	3.18E-2	4.95-5
Nb-94	5.98E-6	1.14E-6	1.02E-2	1.16E-8	Sn-126	7.19E-2	2.01E-3	1.08E-3	2.17E-6
Pd-107	8.72E-1	4.45E-4	5.50E-5	2.45E-8	U-234	4.94E-1	3.07E-3	2.83E-2	8.67E-5
Ce-141	6.24E-20	1.75E-15	1.47E-3	2.57E-18	Cm-247	1.03E-4	9.54E-9	3.12E-2	2.98E-10
							Total Heat Load		4.47 watts

Since 4.5 rods equivalents would contain approximately 21 thermal watts. The expected load will be less than 6 watts. The content meets thermal loading limit for the 10-160B.

4.5 Shielding

The 10-160B will be operated under "exclusive use" with the contents in the cask not creating a dose rate exceeding 200 mrem/hr on the cask surface, or 10 mrem/hr at two meters from the outer lateral surfaces of the vehicle. The package shielding must be sufficient to satisfy the dose rate limit of 10 CFR 71.51 which states any shielding loss resulting from the hypothetical accident will not increase the external dose rate to more than 1000 mrem/hr at one meter from the external surface of the cask.

The cask sidewall consists of an outer 2-inch thick steel shell surrounding 1-7 /8 inches of lead and an inner containment shell wall of 1 1/8-inch thick steel. The primary cask lid consists of two steel layers with a total thickness of 5.5 inches. The lid closure is made in a stepped configuration to eliminate radiation streaming at the lid/cask body interface. A secondary lid is located at the center of the main lid, covering a 31-inch opening. The secondary lid is constructed of steel plates with a total thickness of 5.5 inches with multiple steps machined in its periphery. These steps match those in the primary lid, eliminating radiation-streaming pathways. The cask bottom has an identical shielding effectiveness to the cask lids. It also consists of two layers of steel with a total thickness of 5.5 inches. Foam filled impact limiters cover the top and bottom of the vertically oriented cask. The impact limiters are conservatively ignored for the purpose of the shielding evaluation.

In addition to the cask, each 3525 Lead Shield Can contains 2" of lead.

4.5.1 Gamma

A gamma content of 1.125 rods per shipment with a pin bundle (consisting of up to 8 pins) will be inside 2" of lead shielding in the 3525 Lead Shield Cans, placed inside of the 10-160B. As part of a preliminary gamma evaluation, a comparison of content and shielding to previously analyzed HBU fuel segments was performed with the results showing that the content would meet the transport gamma shielding requirements.

In addition, more than 1 of lead can be added to the inside of each 3525 Lead Shield Can if required.

4.5.2 Neutron

SCALE models of the 10-160B cask were evaluated with a Pu-Be neutron source. A Pu^{239}-Be source produces neutrons at a rate of approximately 1.4E+06 n/sec per Ci (Ref. 10). 325 FGE from a Pu^{239}-Be source will produce approximately 2.8E+07 n/sec. The equivalent neutron source, which produces a dose rate of 9.4 mrem/hr at 2m from the 8' wide trailer, has an emission rate of 1.1 E+08 n/sec. Thus, the equivalent source used for the dose rate calculation is larger than the fissile gram limit imposed by the criticality evaluation and gives a conservative dose rate result. Therefore, the preliminary neutron evaluation the neutron shielding is acceptable.

4.6 Structural Evaluation

<u>Criteria: Net package payload must be less than 14,500 lbs.</u>

The current SARP structural analysis is based on mass (Ref. 10). Section 2 shows that the net package payload is limited to 14,500 lbs. Therefore, as long as the weight of the content mass stays less than 14,500 lbs, the content will be within that that is currently analyzed.

This mass limit will not be exceeded based on the following:

1 A standard 55-gallon drum has a volume of approximately 218.3 liters and weighs a maximum of about 60 lbs. Therefore 10 should weigh a maximum of 600 lbs.
2 Based on the 2” lead shield can shipping configuration, the mass would be bounded by a solid 9.5-inch diameter solid lead cylinder 35 inch tall. This equates to a volume of approx. 2,480 inches cubed or 40,640 cm^3. Since lead has a density of 11.34 $gram/cm^3$, the solid cylinder would weigh 460,858 grams or 1,016 lbs. For 10 solid cylinders, the mass would be 10,160 lbs.
3 The weight of a pallet is assumed to weigh 100 lbs.
4 Adding all of the weight results in 600 lbs + 10,160 lbs + 100 Lbs = 10,860 lbs

Since the calculated content weight is bounded by 10,860 lbs, accounting for additional strap down apparatus or cribbing, the content would still be less than 14,500 lbs (Ref. 9). Therefore the existing SARP structural analyses (Ref. 9) is bounded and shipment of the HBU sister rod segments would be acceptable under the current CoC (Ref. 1).

5. 10-160B SUMMARY

The content evaluation for using the 10-16B Cask for shipping an equivalent to 4.5 rods of HBU North Anna Fuel Rod Segments from ORNL to PNNL shows the following are acceptable:

- Enrichment
- A2s loading
- FGE loading
- Thermal loading
- Shielding
- Structural Evaluation

6. NAC-LWT CASK DESCRIPTION

This section provides an overview of the NAC-LWT. Table 6.1 provides an overview of availability and applicable Certificate of Compliance (COC) (Ref. 12 and 13) Content Limitations

Table 6.1 NAC-LWT Availability and Content Limits

Cask	Package ID Number	CoC Expiration Date	Packages Available	CoC Content Limits
		HBU Fuel Segments		
NAC-LWT	USA/9225/B(U) F–96 (NRC) USA/9225/B(U) F-96 (DOE)	DOE, 09/2017 (Ref.12) NRC, 04/2020 (Ref. 13)	5 casks in service owned by NAC	• 25 full-length, high burnup fuel rods. 14 of the fuel rods can be classified as damaged

CoC = certificate of compliance, DOE = U.S. Department of Energy, ID = identification, NRC = U.S. Nuclear Regulatory Commission

The NAC-LWT cask (USA/9225/B(U)F–96) is a steel-encased, lead-shielded shipping cask with a forged stainless steel lid and 12 closure bolts, tailored payload baskets to the contents being shipped, water-ethylene-glycol neutron shield, and top- and bottom-end impact limiters (Figure 6.1). The cask is designed to transport one PWR assembly or two boiling water reactor (BWR) assemblies by truck. Five of the legal weight truck (LWT) casks are owned and offered for use by NAC International, based in Norcross, Georgia.

The LWT cask CoC authorizes its use to ship research reactor spent fuel, nuclear power plant spent fuel assemblies and fuel rods, and other irradiated materials. When used to ship commercial nuclear power reactor fuel rods, the maximum burnup authorized by the CoC is 80,000 MWd/MTU. For fuel assemblies, the maximum authorized burnup is 35,000 MWd/MTU. The CoC authorizes shipment of up to 25 individual fuel rods with as many as 14 of the rods being classified as damaged. Damaged rods may include fuel debris, particles, loose pellets, and fragmented rods. Damaged fuel rods must be placed in a fuel rod insert or individual failed fuel rod capsules prior to placement in the fuel rod insert. The fuel rod insert must be transported in a PWR/BWR transport canister in the cask's PWR basket.

The cask can be loaded wet in a pool or dry in hot cell facilities or facilities with small pools and limited crane capacities. The NAC-LWT cask can be used for full-length HBU assemblies, or pieces of HBU fuel.

Figure 6.1. NAC-LWT Cask

7. NAC-LWT CONFIGURATION

The maximum decay heat is not to exceed 2.3 kilowatts per package. Intact individual rods may be placed either in an irradiated or unirradiated fuel assembly lattice (skeleton) or in a fuel rod insert. The PWR fuel assembly lattice must be transported in the PWR basket.

Up to 14 of the 25 fuel rods may be classified as damaged. Damaged fuel rods may include fuel debris, particles, loose pellets, and fragmented rods. Damaged fuel rods must be placed in a fuel rod insert. (See Figure 7.1.) Damaged fuel rods may also be placed in individual failed fuel rod capsules, as shown in Figure 1.2.3-11 of the application, prior to placement in the fuel rod insert. Guide/instrument tubes and tube segments may be placed in the fuel rod insert. The fuel rod insert must be transported in a PWR/BWR transport canister, which is positioned in the PWR insert in the PWR basket.

Figure 7.1. PWR Fuel Rod Insert

8. NAC-LWT SUMMARY

Based on a requirement to ship 10-rod equivalents, in a single shipment, of high burnup PWR fuel rods the NAC-LWT is the only current or forecast option.

All 10 fuel rods may be classified as damaged for this shipment. Damaged fuel rods must be placed in a fuel rod insert. Damaged fuel rods may also be placed in individual failed fuel rod capsules. The fuel rod insert must be transported in a PWR/BWR transport canister, which is positioned in the PWR insert in the PWR basket. A Swagelok or similar system is permitted (as a failed fuel rod capsule) but it must fit into the fuel rod canister with the fuel rod insert. The fuel rod insert is a 5x5 array tube sheet composed of 11/16 inch Tube x .028 Wall. There is a 3-inch gap between the top of the canister tube sheet and the underside of the Canister Lid, so a Swagelok connection can be located up at the top and not penetrate the tubes. If the capsules do not penetrate the canister insert tubes, each capsule will be readily accessible to the hot cell manipulator operator without needing a presentation spacer underneath each capsule. It is expected the shipment will have a combination of intact and sectioned fuel rods not to exceed 25 pieces. Presentation spacers may be needed on the sectioned rods. Any overlap with adjacent tubes will be managed to support a shipment of 10-rod equivalents.

9. REFERENCES

1.) United Sates Department of Energy, USA/9204/B(U)F-96, Certificate Number 9204, Rev. 8, January 2011.

2.) United Sates Nuclear Regulatory Commission, USA/9204/B(U)F-96, Docket Number 71-9204, Energy Solutions, Rev. 22, January 24 2011.

3.) Brady Hanson, et al, Used Fuel Testing Transportation Options, *PNNL-24426, Pacific Northwest National Laboratory, June 30, 2015.*

4.) *Integrated Data Base Report -1994: U. S. Spent Nuclear Fuel and Radioactive Waste, Inventories, Projections, and Characteristics, Appendix B, Characteristics of Important Radionuclides*, DOE/RW-0006, Rev. 11, September 1995.

5.) J. L. Varble, *Compilation of Radionuclide Data*, N-CLC-H-00719, Savannah River Site, 11/17/2008.

6.) Savannah River Nuclear Solutions Criticality Safety Methods Manual, SRNS-IM-2009-00035, 9/13/12.

7.) ANSI/ANS-8.1-1998 R2007, Nuclear Criticality Safety in Operations with Fissionable Materials outside Reactors.

8.) ANSI/ANS-8.15-1981R95, Nuclear Criticality Control of Special Actinide Elements.

9.) *Consolidated Safety Analysis Report for Model 10-160B Type B Radwaste Shipping Cask,* Revision 4, July 2012.

10.) 10 CFR71.51, *Additional Requirements for Type B Packages*, 12/02/2015.

11.) *Transfer of Title Agreement for Up to 25 Spent Fuel Rods from Dominion Virginia Power to the United Sates Department of Energy*, Dated January 9, 2015. US_Active-118345792 by Richard B. Provencher, Idaho Operations Office, U.S. Department of Energy to Kerry L. Basehore, Director, Nuclear Analysis and Fuel, Virginia Electric and Power Company.

12.) United Sates Department of Energy, USA/9225/B(U)F-96, Certificate Number 9225, Rev. 11, January 2013.

13.) United Sates Nuclear Regulatory Commission, USA/9225/B(U)F-96, Docket Number 71-9225, NAC International, Rev. 66, December 22, 2015.

ATTACHMENTS

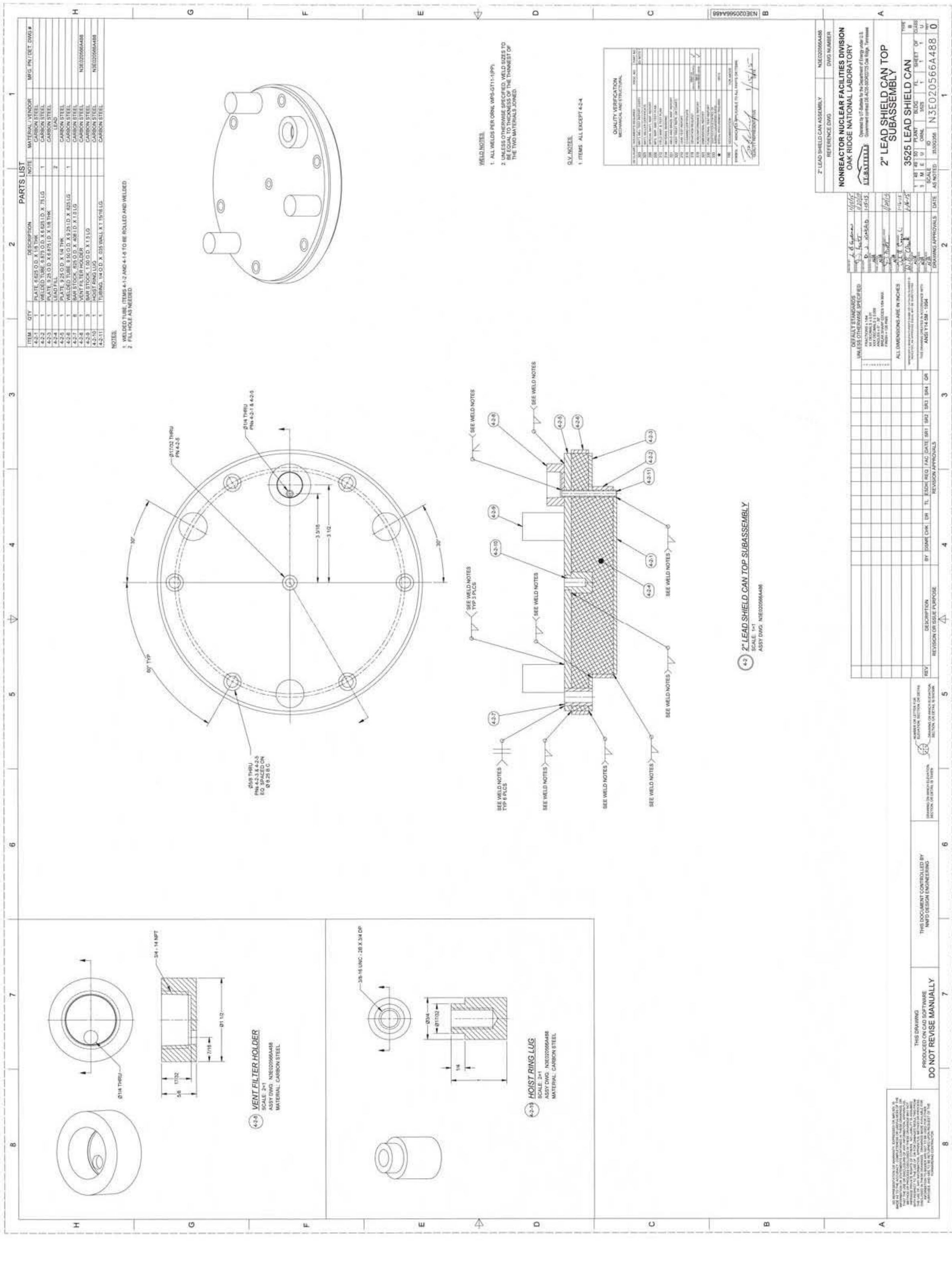
2” Lead Shield Can Top Assembly, 3525 Lead Shield Can, Sheet 1 of 1, N3E020566A488, Rev. 0, Oak Ridge National Laboratory, 1/15/15.

2” Lead Shield Can Top Assembly, 3525 Lead Shield Can, Sheet 1 of 1, N3E020566A485, Rev. 0, Oak Ridge National Laboratory, 1/15/15.

2” Lead Shield Can Top Assembly, 3525 Lead Shield Can, Sheet 1 of 1, N3E020566A486, Rev. 0, Oak Ridge National Laboratory, 1/15/15.

2” Lead Shield Can Top Assembly, 3525 Lead Shield Can, Sheet 1 of 1, N3E020566A487, Rev. 0, Oak Ridge National Laboratory, 1/15/15.

Sample Debris Waste Can Name Plate and TID Modifications, Sheet 1 of 1, B3D020566A516, Rev. 0, Oak Ridge National Laboratory, 4/16/15.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VENT FILTER HOLDER
HOIST RING LUG
NONREACTOR NUCLEAR FACILITIES DIVISION
OAK RIDGE NATIONAL LABORATORY
2" LEAD SHIELD CAN TOP SUBASSEMBLY
3525 LEAD SHIELD CAN
N3E020566A488
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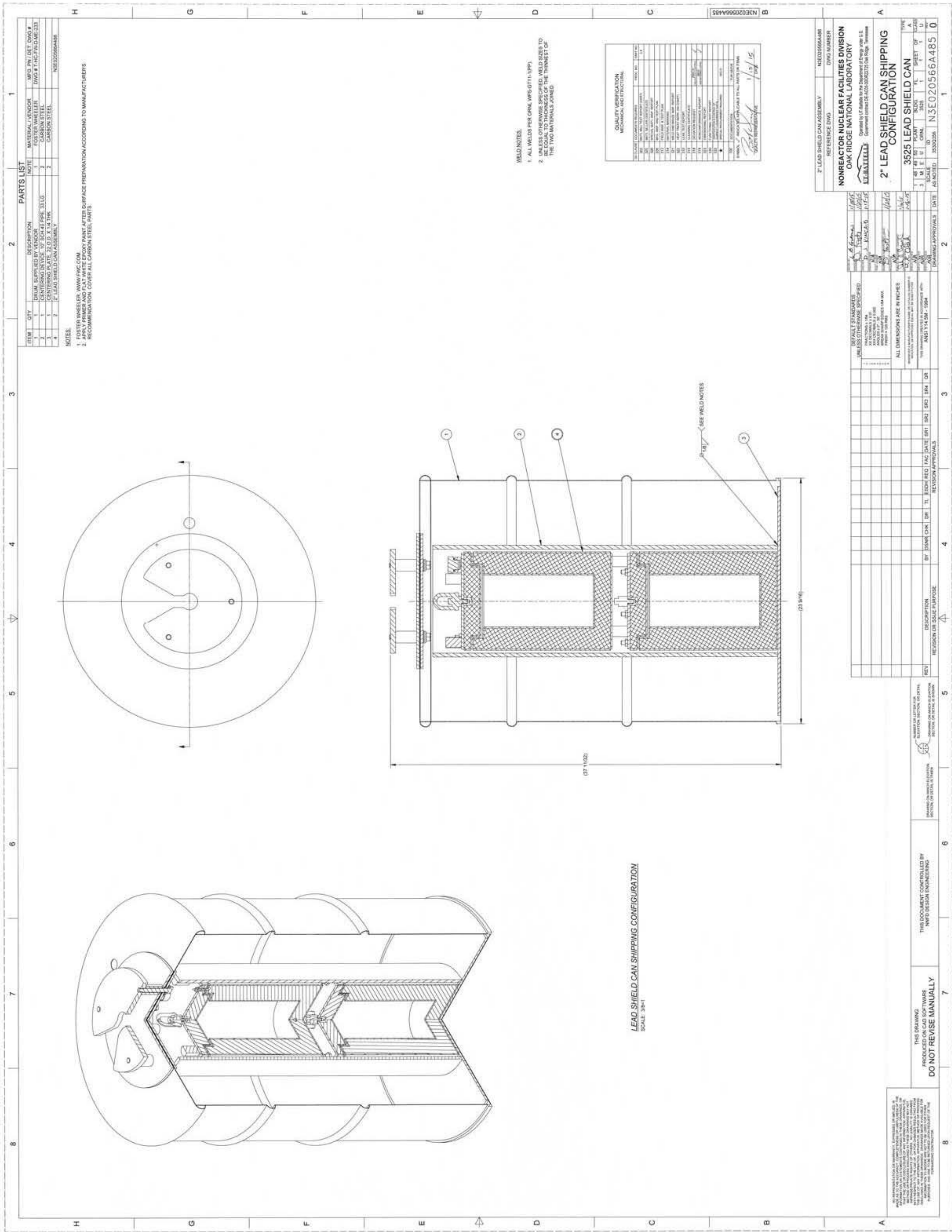
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WELD NOTES
QUALITY VERIFICATION
LEAD SHIELD CAN SHIPPING CONFIGURATION
NONREACTOR NUCLEAR FACILITIES DIVISION
OAK RIDGE NATIONAL LABORATORY
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3525 LEAD SHIELD CAN
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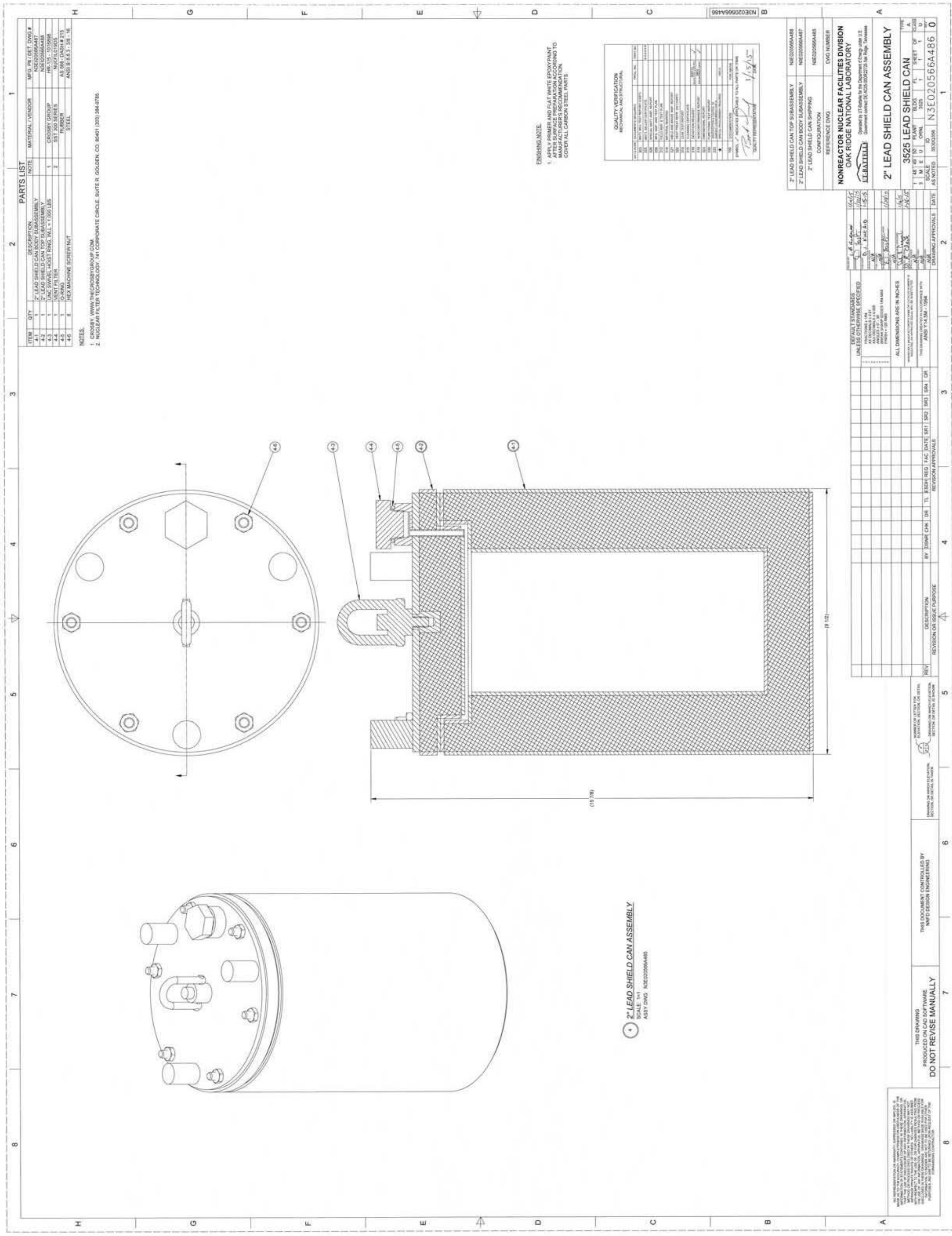
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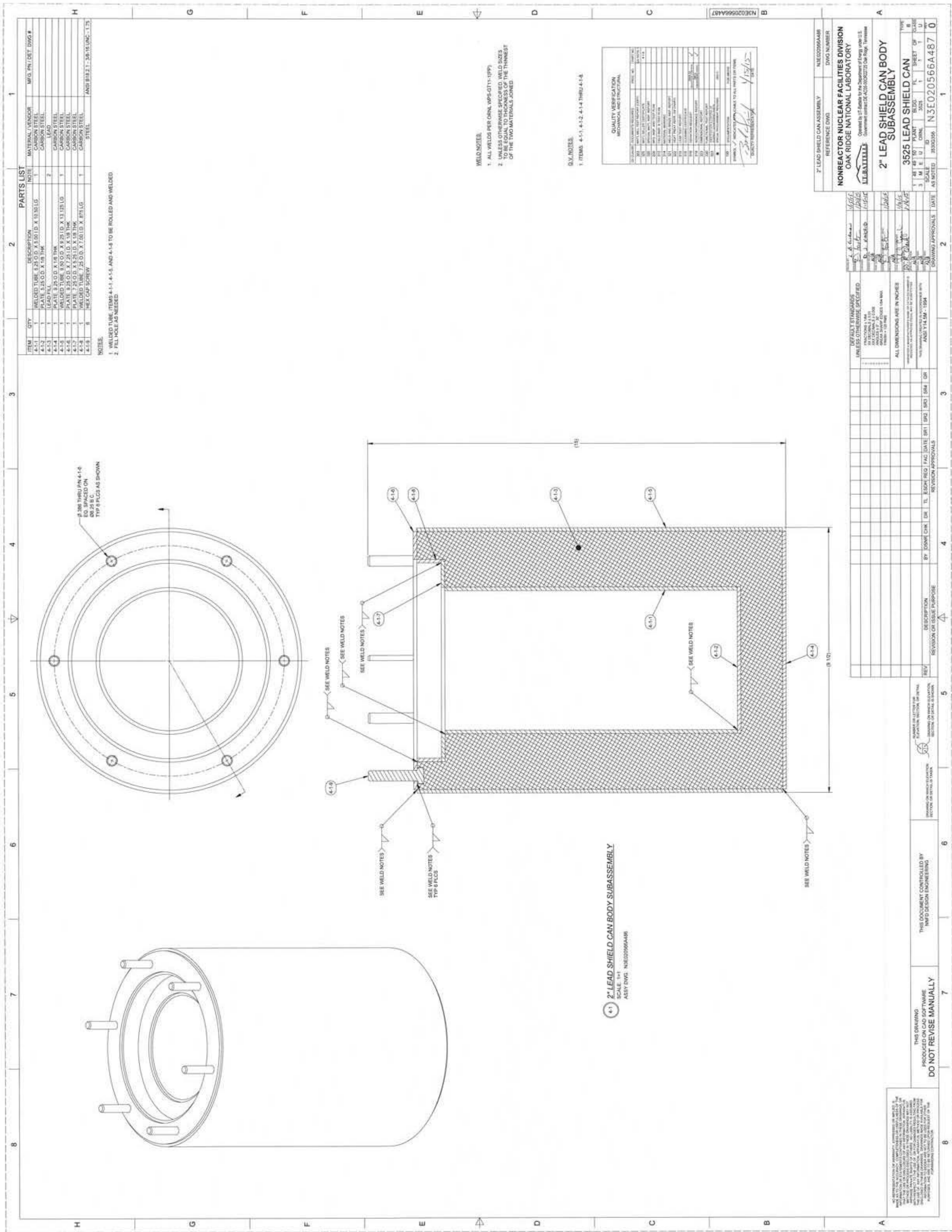
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OAK RIDGE NATIONAL LABORATORY
2" LEAD SHIELD CAN BODY SUBASSEMBLY
3525 LEAD SHIELD CAN
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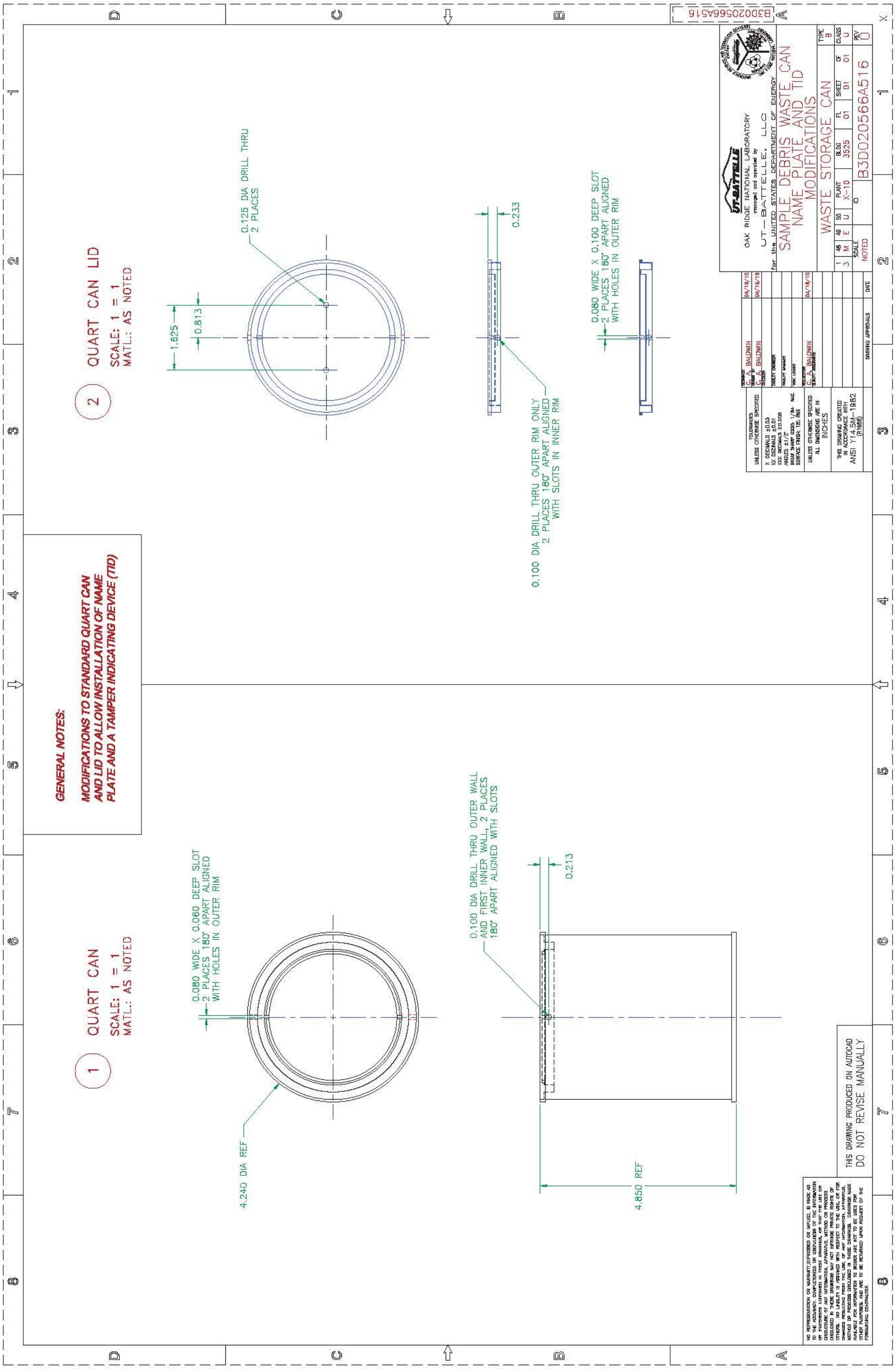
GENERAL NOTES:
MODIFICATIONS TO STANDARD QUART CAN AND LID TO ALLOW INSTALLATION OF NAME PLATE AND A TAMPER INDICATING DEVICE (TID)
1 QUART CAN
SCALE: 1 = 1
MATL.: AS NOTED
2 QUART CAN LID
SCALE: 1 = 1
MATL.: AS NOTED
0.125 DIA DRILL THRU
2 PLACES
1.625
0.813
0.233
0.100 DIA DRILL THRU OUTER RIM ONLY
2 PLACES 180° APART ALIGNED
WITH SLOTS IN INNER RIM
0.080 WIDE X 0.100 DEEP SLOT
2 PLACES 180° APART ALIGNED
WITH HOLES IN OUTER RIM
0.080 WIDE X 0.060 DEEP SLOT
2 PLACES 180° APART ALIGNED
WITH HOLES IN OUTER RIM
0.100 DIA DRILL THRU OUTER WALL
AND FIRST INNER WALL, 2 PLACES
180° APART ALIGNED WITH SLOTS
0.213
4.240 DIA REF
4.850 REF
THIS DRAWING PRODUCED ON AUTOCAD
DO NOT REVISE MANUALLY
OAK RIDGE NATIONAL LABORATORY
UT-BATTELLE, LLC
for the UNITED STATES DEPARTMENT OF ENERGY
SAMPLE DEBRIS WASTE CAN
NAME PLATE AND TID
MODIFICATIONS
WASTE STORAGE CAN
B3D020566A516
C.A. BALDWIN
04/18/15
INCHES
ANSI Y14.5M-1982