ORNL/TM-2021/2054

Development of an Activation Analysis Methodology to Support the Disposal of the High Flux Isotope Reactor Metal Pool Waste

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OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE US DEPARTMENT OF ENERGY

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Nuclear Energy and Fuel Cycle Division

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Date Published: September 27, 2021

Prepared by
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Oak Ridge, TN 37831-6283
managed by
UT-BATTELLE, LLC
for the
US DEPARTMENT OF ENERGY
under contract DE-AC05-00OR22725

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SUMMARY

During operation and maintenance of the High Flux Isotope Reactor (HFIR), metal waste from reactor components is generated. Due to space limitations in the pool fuel storage, it is necessary to remove the waste and send it to off-site disposal. Most of this waste was in the reactor and is radioactive through neutron activation. Prior to shipping and disposal at a waste disposal facility, the radiological activity of the waste must be determined. Therefore, a conservative methodology to characterize the radiological activity of the waste must be performed. The methodology used was based on being able to determine a conservative overall activity and radionuclide inventory based on radiological survey data and neutron activation products for the materials present.

1. INTRODUCTION

The beryllium change out for High Isotope Flux reactor (HFIR) is scheduled for the year 2023. As part of the preparations needed to ensure that a successful core internal change out can be performed, legacy materials need to be removed from the storage pool and they need to be relocated to a long-term waste facility. The removal of the legacy material is mainly to assure that the operations personnel have enough space to maneuver during the core change out as well as to store new material that will be removed from the reactor. Many of the legacy parts that are in the process of being removed is waste from reactor component maintenance and repair. Prior to the removal and transportation of the waste from the fuel storage pool to a permanent waste site a determination of the activation rate and radionuclide inventory is needed for transportation and disposal site purposes.

2. PROBLEM STATEMENT

The radioactive activity of waste items that have been activated in the reactor can be ascertained using both measured exposure rate(s) and the fraction of radionuclides present in the items. Since the reactor parts are primarily composed of aluminum alloys and stainless steel, the main challenge is to determine the fraction of each radionuclide present in the waste with relative accuracy. To overcome this challenge, a strategy that focused on using computational analysis to determine activation products associated with aluminum and stainless steel in the reactor was used. Exposure rate measurements were used along with the inventory of materials, and the computational models of material activation to calculate a conservative radionuclide characterization and total activity. Specifically, given a radionuclide distribution and a dose-to-curie model the total activity can be derived by estimating the activity of gamma emitting nuclides.

3. WASTE INVENTORY

3.1 MATERIAL OF CONCERN

In preparation for disposal waste from the pool, items were segregated and placed in a large pool storage can. Reactor operators segregated these items based on measured underwater on-contact dose rates, with the parts having an acceptable dose rate being designated for disposal. Items with an excessive dose rate are being held for future disposal to allow for decay to acceptable levels. After the storage can was filled, additional items continued to be segregated for disposal and placed in designated locations in the pool. Part of the segregation process involved determining the part type, construction material, mass, and location in the reactor for each item. Some parts for disposal were not installed in the reactor and should have a relatively low activation. These parts are small, suspected contaminated items that fill voids in the can.

A total of 484.59 lbs. (2.1981E+05 g) consisting of 227.09 lbs. (1.03031E+05 g) of aluminum and 257.50 lbs. of steel (1.1680E+05 g) is slated for disposal. Table 1 summarizes waste material types and masses for items in the reactor and not in the reactor. A detailed inventory of items is shown in Appendix A and Appendix B. Carbon steel is listed in the storage can and pool waste. However, none of the carbon steel waste was in the reactor to be activated. These items only have radioactive activity due to contamination on their surfaces (which is primarily Co-60). This activity is accounted for in the dose-to-curie model resulting in a slight overestimation of the total activity as it adds to the Co-60 activity.

Table 1: Waste Material Summary

Waste Location	In Reactor	Aluminum Mass Pounds (g)	Steel Mass Pounds (g)	Total Mass Pounds (g)
Storage Can	Yes	135.85 (6.1621E+04)	94.00 (4.2638E+04)	229.85 (1.0426E+05)
	No	0	14.00 (6.350E+03)	13.80 (6.260E+03)
	Sub Total	135.85 (6.1621E+04)	108.00 (4.8988E+04)	243.85 (1.1061E+05)
Pool	Yes	82.00 (3.7195E+04)	142.00 (6.4410E+04)	224.00 (1.0160E+04)
	No	9.24 (4.19E+03)	7.50 (3.40E+03)	16.74 (7.593E+03)
	Sub Total	91.24 (4.1386E+04)	149.50 (6.7812E+04)	260.74 (1.0920E+05)
	Grand Total	**227.09 (1.0301E+05)**	**257.50 (1.1680E+05)**	**484.59 (2.1981E+05)**

4. CHARACTERIZATION

4.1 ACTIVATION PRODUCTS

As part of characterization methodology for the beryllium cage using computer modeling, activation products and concentrations in curies (Ci) per cm^3 for 6061 aluminum and 304 stainless steel were determined and shown in Table 2 and Table 3 (Navarro, et al. 2021). This modeling resulted in conservative estimates for radionuclide concentrations since it was based on activation in a maximum flux zone (i.e., reactor midplane) at the outer edge of the beryllium reflector. Further discussion of the impact of neutron flux is contained in section 5.2.

Table 2. (100%) Aluminum Isotopic Distribution for Cage Location (Mar-2021)

Aluminum Isotopic Distribution	Ci/cm^3	Activity Fraction
H-3	9.45E-06	2.29E-02
Al-26	9.76E-09	2.37E-05
Si-32	3.97E-11	9.62E-08
P-32	3.97E-11	9.62E-08
Fe-55	1.77E-04	4.29E-01
Fe-60	3.88E-10	9.40E-07
Co-60	2.24E-04	5.43E-01
Co-60m	3.88E-10	9.40E-07
Ni-63	2.11E-06	5.11E-03
Zn-65	2.71E-11	6.57E-08
Total	**4.13E-04**	

Table 3. (100%) Stainless Steel Isotopic Distribution for Cage Location (Mar-2021)

304 Stainless Steel Isotopic Distribution	**Ci/cm^3**	**Activity Fraction**
H-3	3.46E-06	3.44E-06
C-14	1.22E-08	1.21E-08
Fe-55	5.32E-01	5.29E-01
Co-60	1.12E-02	1.11E-02
Ni-59	1.66E-03	1.65E-03
Ni-63	4.61E-01	4.58E-01
Total	**1.01E+00**	

These distributions show most of the radionuclides other than Co-60 are beta emitters or low energy gamma emitters that are hard to detect with the detector used for the survey. Therefore, the dose-to-curie model will be used to estimate the Co-60 activity and the distributions used to determine the total activity and the resultant activities for each nuclide.

The waste does contain steels other than 304 stainless steels, but the steel items that were in the reactor are all 300 series stainless or have very similar composition to 304 stainless steel meaning the activation products for 304 stainless steel are appropriate for characterization of all the steels in the waste stream.

To establish the overall radionuclide fractions for the waste, the relative volumes of aluminum and stainless steel must be determined. Since the mass of the in-reactor items was measured, the volumes and relative volumetric fractions can be determined by multiplying the mass by the density of aluminum (2.7 g/cm^3) and steel (8.0 g/cm^3). For items in the storage can, this results in an in-reactor waste volume of 2.3E+04 cm^3 of aluminum and 5.3E+03 cm^3 of steel for a relative volumetric fraction of 0.81 for aluminum and 0.19 for steel. For the waste items not in the storage can, the in-reactor waste volume is 1.4E+04 cm^3 of aluminum and 8.1E+03 cm^3 of steel for a relative volumetric fraction of 0.63 for aluminum and 0.37 for steel (See Table 4 and Table 5.)

Table 4: Storage Can In-Reactor Waste

In-Reactor Waste Items - Storage Can		
	Aluminum	**Steel**
mass (lbs)	135.85	94.00
mass (g)	6.1621E+04	4.2638E+04
mass fraction	5.9052E-01	4.0896E-01
Density (g/cm^3)	2.7	8.0
Volume (cm^3)	2.3E+04	5.3E+03
Volume fraction	**0.81**	**0.19**

Table 5: Pool In-Reactor Waste

In-Reactor Waste Items – Pool		
	Aluminum	**Steel**
mass (lbs)	82.00	142.00
mass (g)	3.7195E+04	6.4410E+04
mass fraction	0.366071	0.633929
Density (g/cm^3)	2.7	8.0
Volume (cm^3)	1.4E+04	8.1E+03
Volume fraction	**0.63**	**0.37**

Based on these volume fractions, the radionuclide mix can then be determined for the waste in the storage can and for the waste in the pool.

Table 6: Radionuclide Fractions Waste in Storage Can

Radionuclide	Aluminum Nuclide Concentration at 0.81 Volumetric Fraction (Ci/cm3)	Steel Nuclide Concentration at 0.19 Volumetric Fraction (Ci/cm3)	Nuclide Concentration for Mixture (Ci/cm3)	Nuclide Activity Fraction
H-3	7.65E-06	6.57E-07	8.31E-06	4.34E-05
C-14	0.00E+00	2.32E-09	2.32E-09	1.21E-08
Al-26	7.91E-09	0.00E+00	7.91E-09	4.13E-08
Si-32	3.22E-11	0.00E+00	3.22E-11	1.68E-10
P-32	3.22E-11	0.00E+00	3.22E-11	1.68E-10
Fe-55	1.43E-04	1.01E-01	1.01E-01	5.29E-01
Fe-60	3.14E-10	0.00E+00	3.14E-10	1.64E-09
Co-60	1.81E-04	2.13E-03	2.31E-03	1.21E-02
Co-60m	3.14E-10	0.00E+00	3.14E-10	1.64E-09
Ni-59	0.00E+00	3.15E-04	3.15E-04	1.65E-03
Ni-63	1.71E-06	8.76E-02	8.76E-02	4.58E-01
Zn-65	2.20E-11	0.00E+00	2.20E-11	1.15E-10
		Total Ci/cm3	**1.91E-01**	

Table 7: Radionuclide Fractions Waste in Pool

Radionuclide	Aluminum Nuclide Concentration at 0.63 Volumetric Fraction (Ci/cm3)	Steel Nuclide Concentration at 0.37 Volumetric Fraction (Ci/cm3)	Nuclide Concentration for Mixture (Ci/cm3)	Nuclide Activity Fraction
H-3	5.95E-06	1.28E-06	7.23E-06	1.94E-05
C-14	0.00E+00	4.51E-09	4.51E-09	1.21E-08
Al-26	6.15E-09	0.00E+00	6.15E-09	1.65E-08
Si-32	2.50E-11	0.00E+00	2.50E-11	6.72E-11
P-32	2.50E-11	0.00E+00	2.50E-11	6.72E-11
Fe-55	1.12E-04	1.97E-01	1.97E-01	5.29E-01
Fe-60	2.44E-10	0.00E+00	2.44E-10	6.56E-10
Co-60	1.41E-04	4.14E-03	4.29E-03	1.15E-02
Co-60m	2.44E-10	0.00E+00	2.44E-10	6.56E-10
Ni-59	0.00E+00	6.14E-04	6.14E-04	1.65E-03
Ni-63	1.33E-06	1.71E-01	1.71E-01	4.58E-01
Zn-65	1.71E-11	0.00E+00	1.71E-11	4.58E-11
		Total Ci/cm3	**3.72E-01**	

4.2 MICROSHIELD ANALYSIS DESCRIPTION AND METHODOLOGY

A model of the of the storage can was developed and analyzed using MicroShield™ V 8.03. This model was developed to closely approximate the storage can waste geometry, physical characteristics, and exposure rate profile. The model is used to estimate the expected exposure rate from 1 Ci of Co-60. Based on the modeled exposure rate and the measured exposure rate from the survey results, the amount of Co-60 activity of the storage can then be estimated. Once the Co-60 activity is estimated, the other radionuclides are then scaled based on the isotopic distributions from Section 4.1 to determine the overall activity for each radionuclide. This is known as a "dose-to-curie" methodology. Even though the model is designed to closely approximate the actual storage can, conservatism is built in to slightly overestimate the curie content.

4.3 STORAGE CAN SURVEY DATA

In-air surveys of the can were conducted on 6/17/2020, this survey (Ellis June 16, 2020) consisted of 27 measurements taken at four different heights from the bottom of the storage can (See Figure 1: Storage Can Survey Data from HFIR-537801). The mean, median and maximum exposure rates for each survey elevation (dose point) was determined. The survey data is tabularized and shown in Table 8: Storage Can Survey Data Table. It should be noted that several "on contact" measurements were also taken. These were not used in developing the model since on-contact dose rates are particularly sensitive to the distance the measurement is taken. However, these on-contact measurements were used to assist in evaluating the model after it was developed and discussed further in Section 5.

Figure 1: Storage Can Survey Data from HFIR-537801

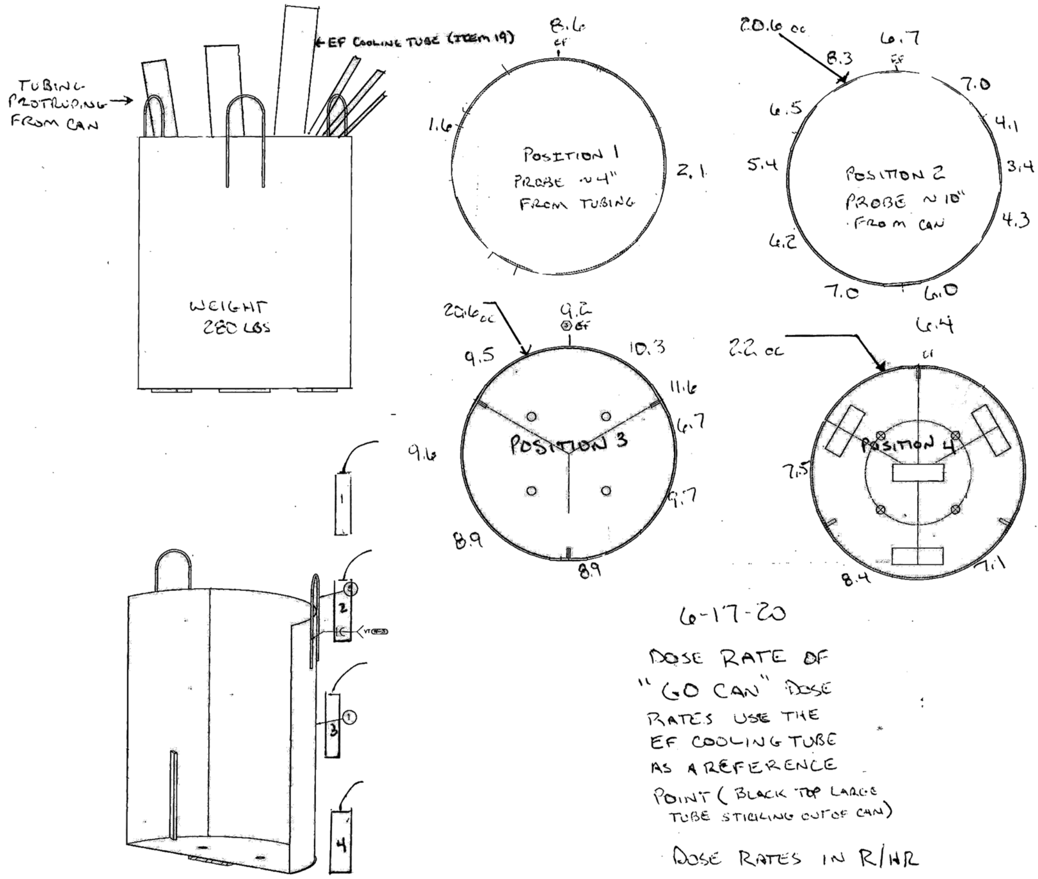


Table 8: Storage Can Survey Data Table

Survey Point	Survey Vertical Location (mR/h)			
	DP1	DP2	DP3	DP4
1	8600	6700	9200	6400
2		7000	10300	
3		4100	11600	
4	2100	3400	6700	
5		4300	9700	7100
6		6000	8900	
7		7000	8900	8400
8		6200		
9		5400	9600	7500
10	1600	6500		
11		8300	9500	
Mean	**4100**	**5900**	**9378**	**7350**
Median	**2100**	**6200**	**9500**	**7300**
Max	**8600**	**8300**	**11600**	**8400**

4.4 MICROSHIELD MODEL

4.4.1 Storage Can Geometry

The dimensions of the storage can are based on current facility drawings (UT-Battelle, Oak Ridge National Laboratory January 4, 2018). The storage can was modeled in MicroShield as a right circular cylinder with an outside radius of 12.625 inches (32.0675 cm), and a height of 30.125 inches (76.5175 cm.) The storage can wall was 14-gauge steel with a thickness of 0.078125 inches (0.198438 cm) from 15 U.S.C. § 206 resulting in a source radius of 12.54688 inches (31.86906 cm). The modeled source volume is 2.4415E+05 cm^3.

4.4.2 Materials and Effective Density

Each piece placed in the storage can was weighed and their mass determined (Table 1). From the source volume calculated above, the relative density for aluminum and steel mixture can thus be determined by dividing their respective masses by the source volume. This method is consistent with the material mixture specification instructions included in the MicroShield User's Manual (Grove Software, Inc. 1992). The source specification based on these masses and volume is an aluminum density of 2.5239E-01 g/cm^3 and 1.7501E-01 g/cm^3 for iron. The density of the wall cladding was 8.0 g/cm^3 iron. Iron was used as the surrogate material for stainless steel in MicroShield as it is a built-in material which very closely approximates the shielding characteristics of both stainless and carbon steels.

4.4.3 Source Term

The source used for the model was assumed to be 1 Ci (3.7E10 Bq) of Co-60. This is based on the isotopic distribution from Table 6 and Table 7. This distribution shows most of the radionuclides other than Co-60 are beta emitters or low energy gamma emitters that are hard to detect with the detector used for the survey. Therefore, the exposure rate from the storage can is conservatively assumed to be solely from Co-60 photons. Use of 1 Ci as the activity allows for scaling the activity with the empirical measurements to estimate the Co-60 activity at the time of the survey and therefore the total radionuclide inventory.

4.4.4 Dose Point Location

During the survey of the cage, the positions the detector was placed from storage container was as follows: dose point 1 was approximately 4" from the tubing that extended approximately 24" above the storage can; dose point 2 was approximately 10 inches at the top of the storage can; dose point 3 was centered between the top and bottom of the storage can at approximately 10 inches away; and dose point 4 was near the bottom of the storage can approximately 10 inches away. Based on the survey measurement locations, the dose points for the MicroShield model are selected as follows: Dose point 1 is 4 inches (10.16 cm) from the side of the can (42.228 cm from the center vertical axis) at a height of 54.125 inches (137.478 cm) from the bottom. Dose Point 2 through 4 are 10 inches (25.4 cm) from the side of the storage can (57.468 cm from the center vertical axis) at 82.518 cm, 38.735 cm, and 4.0 cm from the bottom. Additionally, an "on-contact" dose point was used for further evaluation of the model. This dose point coincided with the dose point 3 height but is 1 inch (2.54 cm) from the side at 38.735 cm from the vertical axis of the storage can.

4.4.5 MicroShield Results

The MicroShield model resulted in the exposure rates for 1 Ci of Co-60 as shown in Table 9 (See Appendix C).

Table 9: MicroShield Results for 1 Ci Co-60

Dose Point	Description	Exposure Rate ($mR\ h^{-1}\ Ci^{-1}$)
DP-1	4” from side above storage can	907.4
DP-2	Top of storage can 10” from side	2032
DP-3	Middle of storage can 10” from side	3082
DP-4	Bottom of storage can 10” from side	2383
DP-5	Middle of storage can 1” from side	8612

5. CALCULATION AND EVALUATION OF MODEL RESULTS

5.1 ACTIVITY ESTIMATION

Using the median exposure rate from the survey for dose point 3 of 9500 mR h^{-1} and the corresponding model exposure rate of 3082 mR h^{-1} Ci^{-1}, the Co-60 activity can be estimated to be 3.082 Ci (Equation 1). The median exposure rate from the survey data was used to result in a conservative activity estimation. It also resulted in a closer agreement with dose points 2 and 4.

Equation 1: Activity Estimation using Dose-to-Curie

$$A = \frac{9500\ mR\ h^{-1}}{3082\ mR\ h^{-1} Ci^{-1}} = 3.082\ Ci\ Co - 60$$

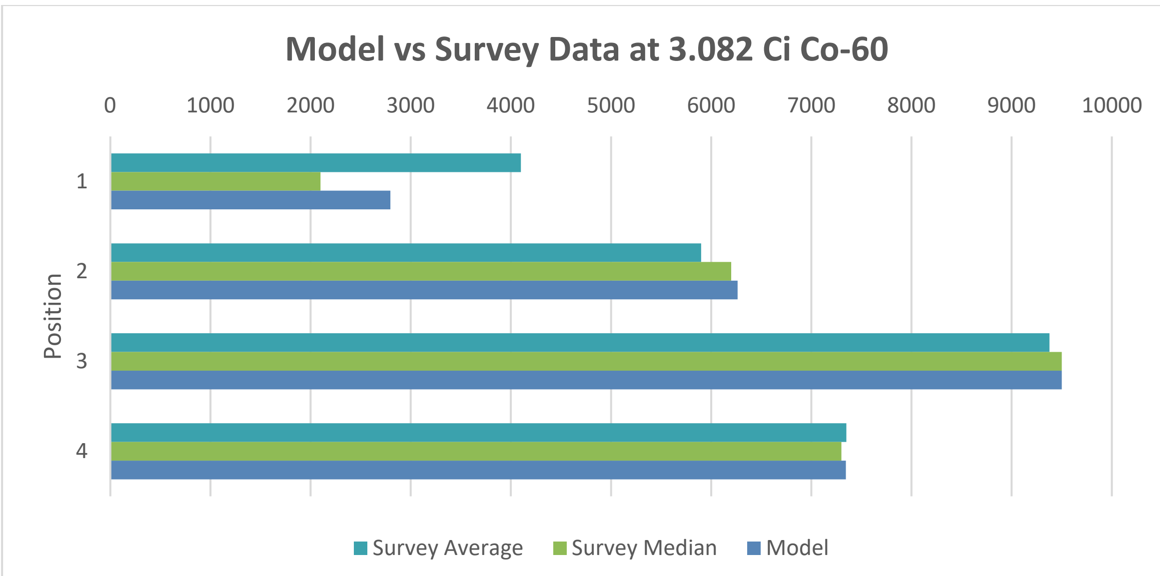
Using this estimate, a comparison of the survey dose profile and model dose profile for all four dose points can be generated. As can be seen in Table 10 and Figure 2, the model closely approximates the survey median and average exposure rates at the dose points except for dose point 1 and is conservative for dose points 2 through 4. Since the survey at dose point 1 was taken very close to tubing protruding from the top of the storage can, an additional model was created to adjust the activity estimation for this tubing.

Additionally, the model indicates that at 3.082 Ci Co-60, the exposure rate on contact with the storage can at dose point 3 would be 2.654E+04 mR/h compared to the on-contact survey result of 2.06E+04 mR/h. This further validates the activity derived from the storage can model is realistic but conservative as far as the waste contained within the storage can.

Table 10: Model vs Survey Data at 3.082 Ci Co-60

Dose Point	Description	Survey Average Exposure Rate (mR h^{-1})	Survey Median Exposure Rate (mR h^{-1})	Modeled Exposure Rate at 3.082 Ci (mR h^{-1})
DP-1	4" from Side Above Storage Can	4100	2100	2797
DP-2	Top of Storage Can 10" from side	5900	6200	6263
DP-3	Middle of Storage Can 10" from side	9378	9500	9500
DP-4	Bottom of Storage Can 10" from side	7350	7300	7345

Figure 2: MicroShield Model and Survey Data Comparison at 3.082 Ci of Co-60

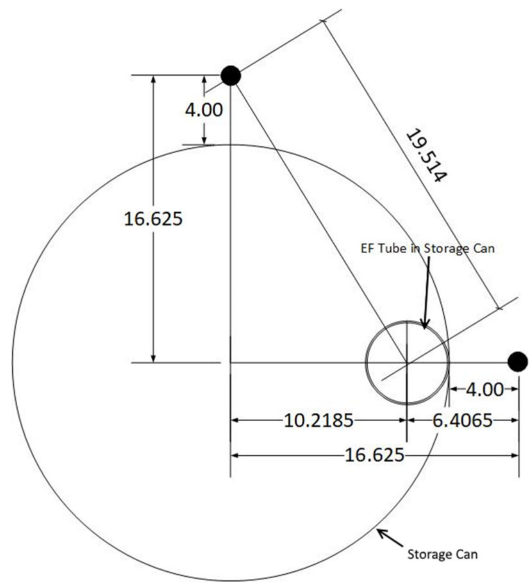


5.1.1 Additional Experiment Facility Tubing Model

Some of the waste, primarily tubing, extended above the storage can. Since the estimated exposure rate at dose point 1 for the model at 3.082 Ci was less than the survey average exposure rate, an additional model for a tube protruding from the storage can was created. According to the survey data, the maximum dose rate at that location coincided with the Engineering Facility Tube Cooling Water Guide Sleeve segment (EF Tube). According to engineering drawings (Union Carbide Nuclear Company, Oak Ridge National Laboratory November 15, 1983), this tube had a maximum inside diameter of 4.625 inches (11.7475 cm) and a wall thickness of 0.094 inches (0.23876 cm). The survey indicated that the exposure rate was taken 4 inches (10.16 cm) from the tubing. For this dose point, there were three measurements, one at the EF Tube and two approximately 90 degrees from the tube.

Therefore, an annular cylinder with inner source radius of 2.3125 inches (5.87375 cm) and a thickness of 0.094 inches (0.23876 cm) with a length of 24 inches (60.96 cm) was modeled. The dose point was placed 4 inches from the wall (6.4065 inches from the vertical axis) at the end of the tubing. This vertical position is worse case since it would result in a lower exposure rate per curie. To model a 90-degree rotation, dose point 2 was modeled at 19.5143 inches (49.56635 cm) from the vertical axis of the EF tube (See Figure 3). Since this would also affect the modeled dose rates at dose points 2 through 4 for the storage can. These were modeled (dose points 3 through 5) at 10 inches from the wall (12.065 inches from the vertical centerline).

Figure 3: EF Tubing Model Dose Points (Plan View in inches)



5.1.2 Adjusted Activity for EF Tubing

The EF tubing model resulted in the exposure rates per curie of Co-60 as shown in Table 11.

Table 11: EF Tube Model Exposure Rate (mR h^{-1} Ci^{-1})

Dose Point	Description	Exposure Rate (mR h^{-1} Ci^{-1})
EF-DP-1	4" from end of EF Tube	1.51E+04
EF-DP-2	19.5143" from the end of EF tube (90-degree rotation)	3.51E+03
EF-DP-3	10" from EF Tube at DP-2	6.68E+03
EF-DP-4	10" from EF Tube at DP-3	2.11E+03
EF-DP-5	10" from EF Tube at DP-4	8.31E+02

Using the maximum dose rate for DP-1 of 8600 mR/h minus the estimated dose rate of 2797 mR/h at that location from 3.082 Ci of Co-60 in the storage can, we can estimate the additional Co-60 activity from the EF tube (Equation 2.)

Equation 2: EF Tube Activity Estimation

$$A = \frac{(8600 - 2797)\ mR\ h^{-1}}{1.512E + 04\ mR\ h^{-1}Ci^{-1}} = 0.384\ Ci\ Co - 60$$

Using this estimate, we can determine the total exposure rate from the storage can at 3.466 Ci (3.082 Ci for the storage can plus 0.384 Ci from the tube) and the maximum exposure rate at each of the dose point locations. These are shown in Table 12. The 90-degree rotation dose rates at this activity also shows it to be very conservative. The comparison from the modeled exposure rates at 3.466 Ci to actual measurements shows the models to be conservative.

Table 12: Exposure Rates for Adjusted Co-60 Activity of 3.466 Ci

Dose Point	Description	From 3.082 Ci Co-60 in Storage Can (mR/h)	From 0.384 Ci from EF Tube Above Storage Can (mR/h)	Total Exposure Rate from model (mR/h)	Corresponding Measured Exposure Rate (mR/h)
DP-1	At EF tube top	2797	5803	8600	8600
DP-1	90-degrees from EF Tube	2797	1346	4143	1600, 2100
DP-2	At top of Storage Can	6263	2565	8829*	6700
DP-3	Middle of Storage Can	9500	808.7	10310*	9200
DP-4	Bottom of Storage Can	7345	318.9	7664*	Not Taken**

*This would be the expected exposure rate at the EF tube location shown on Figure 1 with 3.466 Ci Co-60.
**An exposure rate measurement was not taken adjacent to the EF tube for this dose point

5.1.3 Decay Correction

The survey was conducted on 6/17/2020, however the radionuclide distributions were decay corrected for 3/06/2021. To use the radionuclide distributions, the Co-60 activity must be decay corrected to 3/06/2021. The half-life of Co-60 is 5.2713 years (ICRP-107) and the decay time is 0.717 years. This results in a decay corrected Co-60 activity of 3.154 Ci on 3/06/2021 (Equation 3.)

Equation 3: Decay Correction Equation

$$A_t = A_0 e^{-\lambda t} = 3.466\ Ci\ e^{-\left(\frac{\ln(2)}{5.2713y}\right)0.7173y} = 3.154\ Ci\ Co - 60$$

5.1.4 Pool Waste Activity Estimation

Assuming the waste in the storage can and the waste outside the can (pool waste) comes from relatively similar processes, the Co-60 concentrations for aluminum and steel would be similar. In other words, the storage can waste should be a good sample of the entirety of the waste slated for disposal and its activity can be used to infer the activity for all the waste from similar processes slated for disposal. Therefore, the pool waste activity can then be conservatively determined using the Co-60 concentrations for both the aluminum and steel components of the waste in the storage can.

Based on the isotopic distributions assumed for the 0.81 to 0.19 aluminum to steel volumetric ratio (Table 6), 7.86% of the Co-60 activity comes from aluminum and 92.14% of the Co-60 activity comes from stainless steel (1.814E-04 Ci cm^{-3} / 2.309E-03 Ci cm^{-3} = 7.856E-02 for aluminum, and 2.128E-03 Ci cm^{-3} / 2.309E-03 Ci cm^{-3} = 9.214E-01 for stainless steel). At a Co-60 activity of 3.154 Ci, that equates to an activity of 2.478E-01 Ci for the aluminum components and 2.906E+00 Ci for the stainless steel components. With an in-reactor aluminum volume of 2.282E+04 cm^3 and steel volume of 5.330E+03 cm^3 in the storage can, that results in volumetric Co-60 concentrations in the storage can of 1.086E-05 Ci/cm^3 for aluminum and 5.453E-04 Ci/cm^3 for stainless steel (Table 13). It should be noted that these concentrations are lower than the modeled concentrations, and are indicative of the model's conservatism (e.g., material modeled in the highest flux region of the reactor.) However, these concentrations are based on actual measurements of the waste in the storage can and represent a better estimate of the activity. A more detailed discussion concerning the impact of neutron flux is included in Section 5.2.

The waste volumes for the pool waste are 1.378E+04 cm^3 aluminum and 8.051E+03 cm^3 steel. Using the Co-60 volumetric concentrations from the storage can waste results in an activity of 1.5E-01 Ci for the aluminum components and 4.4E+00 Ci for the steel components with a total of 4.5E+00 Ci Co-60 (See Table 13).

Table 13: Pool Waste Co-60 Activity Calculation

	Aluminum	Steel	Total
Storage can Co-60 activity fraction by material type (determined from Table 6)	7.856E-02	9.214E-01	1.000E+00
Storage can Co-60 activity on 3/6/21 (Ci from 5.1.3)	2.48E-01	2.91E+00	3.154E+00
In-reactor waste volume in storage can (Table **4**)	2.282E+04	5.330E+03	2.815E+04
Storage can waste Co-60 concentration (Ci/cm^3)	5.554E-06	5.680E-04	1.120E-04
Pool waste volume (cm^3)	1.378E+04	8.051E+03	2.183E+04
Pool waste Co-60 activity on 3/6/21* (Ci)	**1.5E-01**	**4.4E+00**	**4.5E+00**

*All calculations carried through without rounding, final activity reported at 2 significant digits.

5.1.5 Pool Waste Activity Assessment from On-Contact Exposure Rates

The author performed an evaluation of Co-60 activity based on underwater on-contact survey measurements taken on the pool waste items during segregation activities. The MicroShield model conservatively assumed a 3" diameter steel rod with a mass of 40 lbs. (the maximum mass in the inventory). The on-contact measurement distance was assumed to be 2.5 inches which is very conservative when the actual distance would be with the probe being as close to the material as possible. The estimated exposure rate of 2.331E+04 mR h^{-1} Ci^{-1} from the MicroShield model results in a conservatively estimated Co-60 activity of 3.2 Ci. This activity is lower than the 4.5 Ci estimated from 5.1.4 and provides added confidence that the method used in 5.1.4 is conservative. Additionally, all items had contact dose rate measurement performed underwater (storage can and pool waste items). This information is shown in Appendix A and B. The maximum, average and median dose rates for the storage can exceeds the average underwater on-contact exposure rates for the pool waste items providing additional evidence the estimated activity for the pool waste items is conservative.

5.1.6 Total Radionuclide Activities

Using the Co-60 activities from 5.1.3 and 5.1.4, and the radionuclide fractions for the storage can and pool waste in Table 6 and Table 7, the activities for each radionuclide can be determined. These activities are shown in Table 14. The total activity of the reactor waste in the storage can and pool is estimated to be 660 curies.

Table 14: Waste Radionuclide Activities

Radionuclide	Nuclide activity fraction storage can waste (0.81 aluminum to 0.19 steel volumetric fraction)	Nuclide activity for storage can (Ci)	Nuclide activity fraction pool waste (0.68 aluminum to 0.32 steel volumetric fraction)	Nuclide activity for pool waste (Ci)	Total activity storage can + pool waste (Ci)
H-3	4.34E-05	1.14E-02	1.94E-05	7.66E-03	1.90E-02
C-14	1.21E-08	3.17E-06	1.21E-08	4.78E-06	7.95E-06
Al-26	4.13E-08	1.08E-05	1.65E-08	6.51E-06	1.73E-05
Si-32	1.68E-10	4.39E-08	6.72E-11	2.65E-08	7.04E-08
P-32	1.68E-10	4.39E-08	6.72E-11	2.65E-08	7.04E-08
Fe-55	5.29E-01	1.38E+02	5.29E-01	2.09E+02	3.47E+02
Fe-60	1.64E-09	4.29E-07	6.56E-10	2.59E-07	6.88E-07
Co-60	1.21E-02	3.15E+00	1.15E-02	4.54E+00	7.69E+00
Co-60m	1.64E-09	4.29E-07	6.56E-10	2.59E-07	6.88E-07
Ni-59	1.65E-03	4.31E-01	1.65E-03	6.51E-01	1.08E+00
Ni-63	4.58E-01	1.20E+02	4.58E-01	1.81E+02	3.00E+02
Zn-65	1.15E-10	3.00E-08	4.58E-11	1.81E-08	4.81E-08
Total		**2.61E+02**		**3.95E+02**	**6.6E+02**

*All calculations carried through without rounding, final activity rounded properly to 2 significant digits.

5.2 EFFECT OF IRRADIATION TIME, NEUTRON FLUX AND DECAY ON NUCLIDE FRACTION AND ESTIMATED ACTIVITY

5.2.1 Irradiation Time and Decay

The activation process resulting in the production of the radionuclides in the mixture is dependent on the amount (mass) of each element present, the target element's neutron cross section, neutron flux, half-life of the activation product, time exposed to the flux and decay time after irradiation (Equation 4).

Equation 4: Neutron Activation Equation

$$\mathbf{A = N(\sigma_{th}\phi_{th} + I_0\phi_{res})(1 - e^{-\lambda\tau_{irr}})(e^{-\lambda t_w})}$$

Where: A = activity
N = number of atoms of target element
σ_{th} = thermal neutron cross section
ϕ_{th} = thermal neutron flux
I_0 = resonance integral
ϕ_{res} = epithermal flux
λ = decay constant
t_{irr} = irradiation time
t_w = decay time

The characterization methodology for the beryllium cage used for developing the source term distribution had four irradiation and decay periods from 1966 to 2021 (Navarro, et al. 2021). Using these results and Equation 4, the isotopic distribution after each irradiation and decay period were calculated to determine how the storage can distribution changes for each irradiation and decay period (Table 17).

Table 15: Nuclide Fraction After Irradiation and Decay Periods

Irradiation / Decay Period	1	2	3	4
Irradiation Period (years)	7.62E+00	7.40E+00	2.50E+00	6.08E+00
Decay Period (years)	6.30E-02	2.48E-01	3.51E+00	2.04E+01
H-3	4.26E-05	4.28E-05	4.36E-05	4.34E-05
Al-26	9.74E-09	9.83E-09	1.01E-08	1.21E-08
C-14	3.33E-08	3.35E-08	3.44E-08	4.13E-08
Si-32	1.39E-10	1.40E-10	1.43E-10	1.68E-10
P-32	1.39E-10	1.40E-10	1.43E-10	1.68E-10
Fe-55	6.05E-01	6.01E-01	5.92E-01	5.29E-01
Fe-60	1.32E-09	1.33E-09	1.37E-09	1.64E-09
Co-60	1.31E-02	1.31E-02	1.32E-02	1.21E-02
Co-60m	1.32E-09	1.33E-09	1.37E-09	1.64E-09
Ni-59	1.33E-03	1.34E-03	1.37E-03	1.65E-03
Ni-63	3.81E-01	3.84E-01	3.94E-01	4.58E-01
Zn-65	1.54E-10	1.44E-10	1.29E-10	1.15E-10
Total Activity for 3.154 Ci Co-60	**2.41E+02**	**2.41E+02**	**2.39E+02**	**2.61E+02**

As can be seen in Table 15, the calculated activity determined from 3.154 Ci of Co-60 is the highest after the fourth irradiation and decay period which was used to estimate the activity in 5.1.6. This shows conservatism in using the isotopic distributions from the previously performed modeling.

5.2.2 Neutron Flux

Some of the material in the pool were in different reactor locations than the cage. This results in a different proportion of thermal, epithermal, and fast neutron fluxes and could impact the radionuclide distribution from activation as shown in Equation 4. As was stated earlier, the model used to determine the nuclide activity distribution was based on material at the maximum flux location at the reflector cage position in the reactor. This position has a high thermal neutron flux. To determine the effect of different neutron flux energies, additional modeling was conducted for placement of aluminum and stainless steel at the center of the flux trap which has the worst case fast and epithermal neutron flux. This allows for comparison of the isotopic distributions as it pertains to the location in the reactor. Table 16 and Table 17 show the isotopic distributions for aluminum and 304 stainless steel placed in the reactor flux trap.

Table 16. (100%) Aluminum Isotopic Distribution for Flux Trap Location (Mar-2021)

Aluminum Isotopic Distribution	Ci/cm^3	Activity Fraction
H-3	3.15E-03	3.19E-02
C-14	7.08E-09	7.16E-08
Al-26	2.75E-06	2.78E-05
Si-32	4.56E-08	4.61E-07
P-32	4.56E-08	4.61E-07
Fe-55	8.91E-05	9.02E-04
Fe-60	1.61E-06	1.63E-05
Co-60	8.98E-02	9.09E-01
Co-60m	1.61E-06	1.63E-05
Ni-63	5.78E-03	5.85E-02
Se-79	3.80E-09	3.84E-08
Kr-85	1.93E-08	1.95E-07
Total	**9.89E-02**	

Table 17. (100%) Stainless Steel Isotopic Distribution for Flux Trap Location (Mar-2021)

304 Stainless Steel Isotopic Distribution	Ci/cm^3	Activity Fraction
H-3	2.99E-03	1.42E-04
C-14	6.22E-07	2.96E-08
Si-32	6.17E-07	2.94E-08
P-32	6.17E-07	2.94E-08
Mn-54	3.88E-07	1.85E-08
Fe-55	1.07E+00	5.08E-02
Fe-60	2.18E-04	1.04E-05
Co-60	1.84E+01	8.74E-01
Co-60m	2.18E-04	1.04E-05
Ni-59	2.12E-05	1.01E-06
Ni-63	1.58E+00	7.51E-02
Total	**2.10E+01**	

As can be seen in Table 16 and Table 17, the Co-60 activity fraction is significantly higher for aluminum alloys (9.09E-01 vs 5.43-01) and stainless steels (8.74E-01 vs 1.11E-02) activated in the flux trap than at the reflector cage location (Table 2 and Table 3). Since total activity is inversely proportional to the Co-60 activity fraction, using the isotopic distributions from the reflector cage location is conservative since it would result in a higher total activity.

Equation 4 also shows that the activity is directly proportional to neutron flux. However, since the cross sections are constant and specific to the material, the relative ratios of the activities will also remain constant. Therefore, it is appropriate to use the relative activity ratios in the dose-to-curie method to determine the total and isotopic activities for items that may have been in a different reactor location and having a different (i.e., lower) neutron flux.

6. CONCLUSIONS AND RECOMMENDATIONS

An approach to determine conservative radionuclide activity using expected neutron activation products, current volumetric fractions, and field radiological measurements was developed and applied. Based on the field measurements and computer modeling of the storage can, the total radionuclide activity content of all pool waste for disposal is estimated to be 660 Ci. The methodology presented in this report was based on using different sources of experimental and simulation methods to obtain a better state of the system. The author also determined a conservative isotopic inventory corresponding to the metal volumetric fractions for both categories of waste (Table 14). In conclusion these activities can be used for shipping and disposal of this waste.

7. REFERENCES

15 U.S.C. § 206. n.d. *Standard Gauge for Sheet and Plate Iron and Steel.*

Ellis, Mark. June 16, 2020. *Radiation / Contamination Job Coverage Survey for Operations Flashing Beryllium Cage, HFIR-537801.*

Grove Software, Inc. 1992. *MicroShield User's Manual.*

Navarro, Jorge, Scott A. Byers, Randal E. Pudelek, Geoffrey G. Deichert, Young Soo Kwon, Russ Wools, Mathew A. Grooms, and David G Blanchard. 2021. *Development of an Activation Analysis Methodology to Support the Disposal of the High Flux Isotope Reactor Original Reflector Container.*

Union Carbide Nuclear Company, Oak Ridge National Laboratory. November 15, 1983. *Engineering Facility Tubes Misc. Details.*

UT-Battelle, Oak Ridge National Laboratory. January 4, 2018. *Large Pool Storage Can, M-10070-0A-344, Rev 1.*

8. APPENDICES

8.1 APPENDIX A – STORAGE CAN INVENTORY

Item Description	Weight (lbs)	Material Group	Material	In Rx	Max Exp. Rate (mR/h)	Vessel location
Snap ring screwdriver (notches on side of blade)	0.25	SS	Carbon Steel	N	100	never in core
Ratchet adapter 3/8" to 1/2"	0.1	SS	Carbon Steel	N	100	never in core
Broken piece with stud	0.25	SS	SS	N	100	never in core
Hole Saw	2	SS	Carbon Steel	N	11,000	never in core
Hole Saw	2	SS	Carbon Steel	N	100	never in core
Hole Saw	2	SS	Carbon Steel	N	100	never in core
Hole Saw	2	SS	Carbon Steel	N	7,000	never in core
Hole Saw	2	SS	Carbon Steel	N	15,000	never in core
Hole Saw	2	SS	Carbon Steel	N	11,000	never in core
tool hanging chain	0.25	SS	Carbon Steel	N	100	never in core
Hook tool	0.25	SS	300 series SS	N	100	never in core
Hook tool	0.25	SS	300 series SS	N	100	never in core
Hook tool	0.25	SS	300 series SS	N	100	never in core
Socket	0.1	SS	Carbon Steel	N	100	never in core
Screwdriver handle	0.1	SS	Carbon Steel	N	100	never in core
Hole Saw	0.2	SS	Carbon Steel	N	8,000	never in core
Control Plate (truncated) 10-3	7	Al	Al 6061-T6	Y	16,000	slightly below midplane just outside of fuel
Exp. In Inpile	2	Al	Al 6061-T6	Y	28,000	midplane in core
Exp. In Inpile	2	Al	Al 6061-T6	Y	500	midplane in core
Vessel Specimen holder Key Rack	7	SS	304 SS	Y	2,000	inside vessel on outer wall midplane
EF Water Guide Sleeve	10	Al	Al 6061-T6	Y	1,000	outside perm beryllium and below
EF Tube Cooling Water Guide Sleeve segment	5	Al	Al 6061-T6	Y	4,000	outside perm beryllium
EF Cooling Tube Water Guide Sleeve with NAPA Hose	7	Al	Al 6061-T6	Y	500	outside perm beryllium and below
RB Liner Tube Assembly	5	Al	Al 6061-T6	Y	2,000	in rb midplane
Bracket pulled out from Item 21	1	Al	Al	Y	1,000	not in vessel
VH-12 to VXF-18 guide Tube with reducer on end, cut open on side	12	SS	304 SS	Y	1,500	above perm beryllium
VH-1 to VXF-5 guide Tube with reducer on end, cut open on side (band of rust)	12	SS	304 SS	Y	2,000	above perm beryllium
Crap plug sample rod	1	Al	Al 6061-T6	Y	4,000	in crap midplane
Target Bundle Rod (88-20)	1	Al	Al 6061-T6	Y	4,000	in center of target bundle
CRAP holddown bolt, nut & washer	1	Al	Al 6061-T6	Y	30,000	in middle of crap midplane
Dummy Exp	2	Al	Al 6061-T6	Y	3,000	in rb midplane
Inpile (cutoff section end)	2	Al	Al 6061-T6	Y	3,500	midplane in core

RB Baffle Plate	1	Al	Al 6061-T6	Y	8,500	top edge of rb
Dummy Target Element	1	Al	Al 6061-T6	Y	11,000	in center of target bundle
VXF Liner with Lead Tube.	4	SS	300 series SS	Y	6,000	in permanent beryllium midplane
VXF Liner with Lead Tube (78-1)	4	SS	300 series SS	Y	300	in permanent beryllium midplane
VXF Liner with Lead Tube (78-2)	4	SS	300 series SS	Y	200	in permanent beryllium midplane
RB thru bolt and nut	1	Al	Al 6061-T6	Y	2,000	in rb midplane
Inpile end section	1	Al	Al 6061-T6	Y	400	midplane in core
RB thru bolt and nut	1	Al	Al 6061-T6	Y	1,800	in rb midplane
Bottom of VXF Liner	1	SS	300 series SS	Y	100	in permanent beryllium midplane
RB Liner tube	1	Al	Al 6061-T6	Y	600	in rb midplane
VXF Liner top broke off	1	SS	304 SS	Y	100	in permanent beryllium midplane
XE production Facility target 76-2	1	Al	Al 6061-T6	Y	100	midplane in core
XE production Facility target 76-1	1	Al	Al 6061-T6	Y	500	midplane in core
Cutoff target rod, H-27	0.5	Al	Al 6061-T6	Y	200	in center of target bundle
RB thru bolt and nut	1	Al	Al 6061-T6	Y	4,600	in rb midplane
CRAP Hanger	0.5	Al	Al 6061-T6	Y	7,400	midplane in core
CRAP Hanger	0.5	Al	Al 6061-T6	Y	7,200	midplane in core
CRAP Hanger	0.5	Al	Al 6061-T6	Y	3,500	midplane in core
CRAP Hanger	0.5	Al	Al 6061-T6	Y	6,100	midplane in core
RB thru bolt and nut	1	Al	Al 6061-T6	Y	2,100	in rb midplane
RB thru bolt and nut	1	Al	Al 6061-T6	Y	2,500	in rb midplane
RB thru bolt and nut	1	Al	Al 6061-T6	Y	2,500	in rb midplane
RB thru bolt and nut	1	Al	Al 6061-T6	Y	2,500	in rb midplane
Inpile end cutoff	0.25	Al	Al 6061-T6	Y	2,000	in center of target bundle
RB thru bolt and nut	1	Al	Al 6061-T6	Y	2,600	in rb midplane
Broken shrouded target rod	1	Al	Al 6061-T6	Y	17,000	in center of target bundle
VXF exp liner- VH to VXF	2.5	SS	304 SS	Y	5,000	above perm beryllium
Pneumatic tube section (old PT-1)	10	SS	304 SS	Y	25,000	above perm beryllium
RB flow baffle	0.25	Al	Al 6061-T6	Y	300	top edge of rb
RB flow baffle	0.25	Al	Al 6061-T6	Y	300	top edge of rb
RB flow baffle	0.25	Al	Al 6061-T6	Y	300	top edge of rb
RB flow baffle	0.25	Al	Al 6061-T6	Y	600	top edge of rb
RB flow baffle	0.25	Al	Al 6061-T6	Y	300	top edge of rb
RB flow baffle	0.25	Al	Al 6061-T6	Y	200	top edge of rb
RB flow baffle	0.25	Al	Al 6061-T6	Y	500	top edge of rb
RB flow baffle	0.25	Al	Al 6061-T6	Y	400	top edge of rb
Removable Reflector upper baffle ring segment	0.25	Al	Al 6061-T6	Y	16,000	top edge of rb
Removable Reflector lower baffle ring segment	0.25	Al	Al 6061-T6	Y	300	top edge of rb

Removable Reflector upper baffle ring segment	0.25	Al	Al 6061-T6	Y	28,000	top edge of rb
Inpile mid section broke off of item 92	2	Al	Al 6061-T6	Y	7,000	in center of target bundle
Dummy rabbit	0.1	Al	Al 6061-T6	Y	3,200	midplane in core
Control Plate coupling special washer(2 washers in barrel)	0.1	SS	17-4 ph SS	Y	300	bottom of control plate
Bearing press tool yoke for plate work	0.25	SS	300 series SS	Y	2,000	used in mandrel box never in core
Control Plate coupling Bellville spring(2 springs in barrel)	0.1	SS	17-4 ph SS	Y	100	bottom of control plate
Beam tube end	2	SS	304 SS	Y	300	midplane in core
CRAP lifting eye	0.2	Al	Al 6061-T6	Y	1,200	top edge of rb
CRAP lifting eye	0.2	Al	Al 6061-T6	Y	800	top edge of rb
Reflector Cage cutoff segment	1	Al	Al 6061-T6	Y	500	top edge of cage
Control Plate coupling retainer washer (only 1)	0.1	SS	17-4 ph SS	Y	600	bottom of control plate
Control Plate coupling Snap ring (8 rings in barrel)	0.25	SS	302 SS	Y	100	bottom of control plate
Track Bearing Shaft	0.1	SS	304 SS	Y	2,900	above or below midplane
Removable Reflector upper baffle ring segment	0.2	Al	Al 6061-T6	Y	29,000	top edge of rb
CRAP lifting eye	0.2	Al	Al 6061-T6	Y	2,100	top edge of rb
Dummy rabbit	0.1	Al	Al 6061-T6	Y	6,300	in center of target bundle
Piece of Dummy Rabbit	0.1	Al	Al 6061-T6	Y	4,500	in center of target bundle
Removable Reflector lower baffle ring segment	0.1	Al	Al 6061-T6	Y	10,000	top edge of rb
tubing	1	SS	SS	Y	1,200	side of cage
Collector Cooling Piping	5	SS	304 SS	Y	7,500	above beam tube outside of permanent be
Target Bundle Rod dummy	1	Al	Al 6061-T6	Y	22,000	in center of target bundle
Target Bundle Rod dummy	1	Al	Al 6061-T6	Y	8,900	in center of target bundle
PTP Rod	1	Al	Al 6061-T6	Y	4,500	in target bundle
tubing	0.5	SS	SS	Y	800	side of cage
Broken shrouded target rod	1	Al	Al 6061-T6	Y	32,000	in center of target bundle
RB thru bolt and nut	1	Al	Al 6061-T6	Y	4,500	in rb midplane
VXF liner	1	Al	Al 6061-T6	Y	12,700	in permanent beryllium midplane
Broken PTP rod	0.5	Al	Al 6061-T6	Y	30,000	in target bundle
Broken PTP rod	0.5	Al	Al 6061-T6	Y	24,000	in target bundle
PT-1 Supply-Exhaust tube	0.5	SS	304 SS	Y	2,400	above perm beryllium
PT-1 Supply-Exhaust tube	0.5	SS	304 SS	Y	600	above perm beryllium
Top Adapter PT-1	1	SS	304 SS	Y	2,200	above perm beryllium

RB thru bolt and nut	1	Al	Al 6061-T6	Y	3,500	in rb midplane
PT-1 Supply-Exhaust tube	0.5	SS	304 SS	Y	1,000	above perm beryllium
Pintle head slotted rod	0.5	SS	304 SS	Y	15,000	midplane in core
PT-1 Supply-Exhaust tube	0.5	SS	304 SS	Y	1,300	above perm beryllium
PT-1 Supply-Exhaust tube	0.5	SS	304 SS	Y	400	above perm beryllium
PT-1 Supply-Exhaust tube	0.5	SS	304 SS	Y	2,800	above perm beryllium
PT-1 Supply-Exhaust tube	0.5	SS	304 SS	Y	2,200	above perm beryllium
RB thru bolt and nut	1	Al	Al 6061-T6	Y	2,400	in rb midplane
CRAP Hanger	0.5	Al	Al 6061-T6	Y	5,300	top edge of rb
RB flow baffle (qty 3)	1.5	Al	Al 6061-T6	Y	600	top edge of rb
RB inner cylinder spacer strip	1	Al	Al 6061-T6	Y	3,500	in rb midplane
RB thru bolt and nut	15	Al	Al 6061-T6	Y	13,800	in rb midplane
RB Baffle Plate lower segment	1	Al	Al 6061-T6	Y	1,000	top edge of rb
item cutoff of cage	5	Al	Al 6061-T6	Y	38,500	cage bottom of rb area
item cutoff of cage	5	Al	Al 6061-T6	Y	20,000	cage bottom of rb area
item cutoff of cage	5	Al	Al 6061-T6	Y	29,500	cage bottom of rb area
item cutoff of cage	5	Al	Al 6061-T6	Y	20,000	cage bottom of rb area
Pressure vessel specimen cover retainer	10	SS	304 SS	Y	900	around beam tube outside permanent be
RB flow baffle pc 11 (qty 9)	2.25	Al	Al 6061-T6	Y	7,500	top edge of rb
RB thru bolt top head only(drilled thru bolt) (qty 11)	2.2	Al	Al 6061-T6	Y	1,500	top edge of rb
RB lower baffle plate	1	Al	Al 6061-T6	Y	2,000	bottom edge of rb
Extension tube segment (6.5 holes)	5	SS	347 SS	Y	100	below permanent be
Extension tube segment (5 holes)	5	SS	347 SS	Y	200	below permanent be
Inner cyclinder nut	1	SS	303 SS	Y	1,000	bottom of control cylinder
Inner cyclinder nut	1	SS	303 SS	Y	100	bottom of control cylinder
RB flow baffle	0.25	Al	Al 6061-T6	Y	1,100	top edge of rb
RB thru bolt top head only	0.2	Al	Al 6061-T6	Y	600	in rb midplane
Broken dummy target rod holder	0.25	Al	Al 6061-T6	Y	100	in center of target bundle
item cutoff of cage	10	Al	Al 6061-T6	Y	20,500	cage bottom of rb area
key for perm Be to cage	1	Al	Al 6061-T6	Y	2,000	outer bottom edge of permanent be on cage
Spring retainer (for coupling)	0.1	SS	304 SS	Y	1,000	bottom of control plate
In Rx Aluminum Weight (lbs)	135.85					
In Rx Steel Weight (lbs)	94.0					
Total In-Rx Weight (lbs)	229.85					

8.2 APPENDIX B – POOL WASTE INVENTORY

Item Description	Weight (lbs)	Material Group	Material	In Rx	Max Exp. Rate (mR/h)	Vessel location
Straight Tubing Section	1	Al	Al 6061-T6	N	15	not in vessel
Straight Tubing Section ¼" (bent end)	1	Al	Al 6061-T6	N	60	not in vessel
RB Liner Tube Assymbly (4 holes @ top)	5	Al	Al 6061-T6	N	10	placed into pool but not in vessel
Rad Trash bag with Misc waste items (Hook tool, saw blades,etc)	7	SS	Carbon Steel	N	45	not in vessel
spring - conduit	0.5	SS	SS	N	35	not in vessel
Clamp Bolt	0.24	Al	Al 6061-T6	N	1,000	Unknown
tool part with 2 pins and 1 bolt	1	Al	Al 6061-T6	N	30	put in rb but never in core
RB Liner Tube	1	Al	Al 6061-T6	N	35	not in vessel
spacer	1	SS	17-4 ph SS	Y	10	tip of cage
Al Disc	1	SS	SS	Y	50	Unknown
RB lower containment ring	20	Al	Al 6061-T6	Y	30,000	bottom of RB
RB lower containment ring	20	Al	Al 6061-T6	Y	8,000	bottom of RB
RB upper containment ring	20	Al	Al 6061-T6	Y	12,000	top of RB
RB upper containment ring	20	Al	Al 6061-T6	Y	4,000	top of RB
Inpile end	1	Al	Al 6061-T6	Y	6,000	above centerline of core
Inpile end	1	Al	Al 6061-T6	Y	2,000	above centerline of core
Extension tube (69-3) bottom end	40	SS	347 ss	Y	500	below permanent Be
Extension Tube (8 holes remaining)	25	SS	347 SS	Y	700	below permanent Be
Extension Tube (3 holes remaining)	25	SS	347 SS	Y	500	below permanent Be
Extension tube (1 hole remaining)	25	SS	347 SS	Y	200	below permanent Be
Extension tube (1 hole remaining)	25	SS	347 SS	Y	10,000	below permanent Be
In Rx Aluminum Weight (lbs)	82					
In Rx Steel Weight (lbs)	142					
Total In-Rx Weight (lbs)	224					

8.3 APPENDIX C – STORAGE CAN MICROSHIELD MODEL

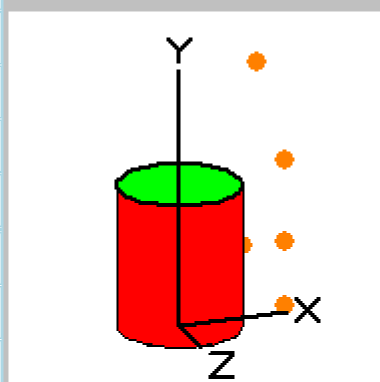
MicroShield 8.03 ORNL (8.03-0000)			
Date	**By**	**Checked**	

Filename	Run Date	Run Time	Duration
GoCan dose profile 2.msd	May 3, 2021	9:29:28 AM	00:00:00

Project Info	
Case Title	Storage Can
Description	Storage Can matching dose profile 1-Ci Co-60
Geometry	7 - Cylinder Volume - Side Shields

Source Dimensions	
Height	76.518 cm (2 ft 6.1 in)
Radius	31.869 cm (1 ft 0.5 in)

Dose Points			
A	**X**	**Y**	**Z**
#1	42.228 cm (1 ft 4.6 in)	137.478 cm (4 ft 6.1 in)	0.0 cm (0 in)
#2	57.468 cm (1 ft 10.6 in)	82.518 cm (2 ft 8.5 in)	0.0 cm (0 in)
#3	57.468 cm (1 ft 10.6 in)	38.735 cm (1 ft 3.3 in)	0.0 cm (0 in)
#4	57.468 cm (1 ft 10.6 in)	4.0 cm (1.6 in)	0.0 cm (0 in)
#5	34.607 cm (1 ft 1.6 in)	38.735 cm (1 ft 3.3 in)	0.0 cm (0 in)



Shields			
Shield N	**Dimension**	**Material**	**Density**
Source	2.44e+05 cm³	Mixed ->	0.4274
		Aluminum	0.25239
		Iron	0.17501
Transition		Air	0.00122
Air Gap		Air	0.00122
Wall Clad	.198 cm	Iron	8
Immersion		Air	0.00122

Source Input: Grouping Method - Actual Photon Energies				
Nuclide	**Ci**	**Bq**	**μCi/cm³**	**Bq/cm³**
Co-60	1.0000e+000	3.7000e+010	4.0959e+000	1.5155e+005

Buildup: The material reference is Source Integration Parameters	
Radial	10
Circumferential	10
Y Direction (axial)	20

Results - Dose Point # 1 - (42.2275,137.4775,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	1.347e+01	2.271e+01	2.600e-02	4.384e-02
1.1732	3.700e+10	1.622e+05	2.406e+05	2.899e+02	4.299e+02
1.3325	3.700e+10	1.908e+05	2.752e+05	3.310e+02	4.774e+02
Totals	**7.401e+10**	**3.530e+05**	**5.158e+05**	**6.209e+02**	**9.074e+02**

Results - Dose Point # 2 - (57.4675,82.5175,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	2.765e+01	5.095e+01	5.339e-02	9.836e-02
1.1732	3.700e+10	3.436e+05	5.388e+05	6.141e+02	9.628e+02
1.3325	3.700e+10	4.065e+05	6.161e+05	7.053e+02	1.069e+03
Totals	**7.401e+10**	**7.502e+05**	**1.155e+06**	**1.319e+03**	**2.032e+03**

Results - Dose Point # 3 - (57.4675,38.735,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	4.469e+01	7.806e+01	8.628e-02	1.507e-01
1.1732	3.700e+10	5.454e+05	8.182e+05	9.747e+02	1.462e+03
1.3325	3.700e+10	6.425e+05	9.338e+05	1.115e+03	1.620e+03

Totals	**7.401e+10**	**1.188e+06**	**1.752e+06**	**2.089e+03**	**3.082e+03**

Results - Dose Point # 4 - (57.4675,4,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	3.265e+01	5.991e+01	6.303e-02	1.157e-01
1.1732	3.700e+10	4.052e+05	6.320e+05	7.241e+02	1.129e+03
1.3325	3.700e+10	4.792e+05	7.224e+05	8.313e+02	1.253e+03
Totals	**7.401e+10**	**8.844e+05**	**1.354e+06**	**1.555e+03**	**2.383e+03**

Results - Dose Point # 5 - (3.46e+01,38.735,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	1.187e+02	2.173e+02	2.292e-01	4.195e-01
1.1732	3.700e+10	1.472e+06	2.285e+06	2.631e+03	4.084e+03
1.3325	3.700e+10	1.740e+06	2.610e+06	3.019e+03	4.528e+03
Totals	**7.401e+10**	**3.213e+06**	**4.895e+06**	**5.651e+03**	**8.612e+03**

8.4 APPENDIX D – EF TUBE MICROSHIELD MODEL

MicroShield 8.03 ORNL (8.03-0000)

Date	By	Checked

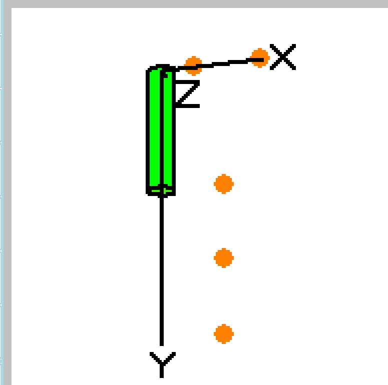
Filename	Run Date	Run Time	Duration
EF Tube.msd	May 3, 2021	9:17:40 AM	00:00:01

Project Info	
Case Title	EF Tube
Description	Tubing Protruding from Storage Can 1 Ci Co-60
Geometry	12 - Annular Cylinder - External Dose Point

Source Dimensions	
Height	60.96 cm (2 ft)
Inner Cyl Radius	5.874 cm (2.3 in)
Inner Cyl Thickness	0.0 cm (0 in)
Outer Cyl Thickness	0.0 cm (0 in)
Source	0.239 cm (0.1 in)

Dose Points

A	X	Y	Z
#1	16.273 cm (6.4 in)	0.0 cm (0 in)	0.0 cm (0 in)
#2	49.566 cm (1 ft 7.5 in)	0.0 cm (0 in)	0.0 cm (0 in)
#3	31.513 cm (1 ft 0.4 in)	60.96 cm (2 ft)	0.0 cm (0 in)
#4	31.513 cm (1 ft 0.4 in)	99.219 cm (3 ft 3.1 in)	0.0 cm (0 in)
#5	31.513 cm (1 ft 0.4 in)	137.478 cm (4 ft 6.1 in)	0.0 cm (0 in)



Shields

Shield N	Dimension	Material	Density
Cyl. Radius	5.874 cm	Air	0.00122
Source	548.075 cm³	Iron	7.86
Transition		Air	0.00122
Air Gap		Air	0.00122
Immersion		Air	0.00122

Source Input: Grouping Method - Actual Photon Energies				
Nuclide	**Ci**	**Bq**	**μCi/cm³**	**Bq/cm³**
Co-60	1.0000e+000	3.7000e+010	1.8246e+003	6.7509e+007

Buildup: The material reference is Source Integration Parameters	
Radial	10
Circumferential	20
Y Direction (axial)	20

Results - Dose Point # 1 - (16.27251,0,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	3.068e+02	3.808e+02	5.922e-01	7.353e-01
1.1732	3.700e+10	3.398e+06	4.015e+06	6.072e+03	7.174e+03
1.3325	3.700e+10	3.917e+06	4.578e+06	6.796e+03	7.942e+03
Totals	**7.401e+10**	**7.315e+06**	**8.593e+06**	**1.287e+04**	**1.512e+04**

Results - Dose Point # 2 - (4.96e+01,0,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	7.328e+01	8.923e+01	1.415e-01	1.723e-01
1.1732	3.700e+10	8.064e+05	9.325e+05	1.441e+03	1.666e+03
1.3325	3.700e+10	9.280e+05	1.061e+06	1.610e+03	1.841e+03
Totals	**7.401e+10**	**1.734e+06**	**1.994e+06**	**3.051e+03**	**3.508e+03**

Results - Dose Point # 3 - (31.51251,60.96,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	1.379e+02	1.692e+02	2.662e-01	3.267e-01
1.1732	3.700e+10	1.522e+06	1.776e+06	2.719e+03	3.174e+03
1.3325	3.700e+10	1.752e+06	2.023e+06	3.040e+03	3.509e+03

Totals	**7.401e+10**	**3.274e+06**	**3.799e+06**	**5.759e+03**	**6.684e+03**
Results - Dose Point # 4 - (31.51251,99.21875,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	3.910e+01	5.276e+01	7.549e-02	1.019e-01
1.1732	3.700e+10	4.449e+05	5.593e+05	7.951e+02	9.995e+02
1.3325	3.700e+10	5.160e+05	6.385e+05	8.952e+02	1.108e+03
Totals	**7.401e+10**	**9.609e+05**	**1.198e+06**	**1.690e+03**	**2.107e+03**
Results - Dose Point # 5 - (31.51251,137.4775,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	1.380e+01	2.054e+01	2.665e-02	3.965e-02
1.1732	3.700e+10	1.624e+05	2.202e+05	2.903e+02	3.936e+02
1.3325	3.700e+10	1.898e+05	2.521e+05	3.293e+02	4.373e+02
Totals	**7.401e+10**	**3.523e+05**	**4.723e+05**	**6.196e+02**	**8.309e+02**

8.5 APPENDIX E – UNDERWATER ON CONTACT MICROSHIELD MODEL

MicroShield 8.03 ORNL (8.03-0000)

Date	By	Checked

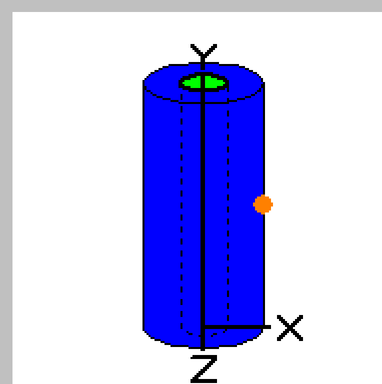
Filename	Run Date	Run Time	Duration
25lbunderwaterDTC.msd	May 27, 2021	10:57:40 AM	00:00:00

Project Info	
Case Title	Underwater DTC
Description	On Contact (2.5") 40 lb Steel Rod
Geometry	7 - Cylinder Volume - Side Shields

Source Dimensions	
Height	45.12 cm (1 ft 5.8 in)
Radius	4.0 cm (1.6 in)

Dose Points			
A	X	Y	Z
#1	10.35 cm (4.1 in)	22.56 cm (8.9 in)	0.0 cm (0 in)

Shields			
Shield N	Dimension	Material	Density
Source	2267.962 cm³	Iron	8
Shield 1	6.35 cm	Water	1
Transition		Air	0.00122
Air Gap		Air	0.00122



Source Input: Grouping Method - Actual Photon Energies				
Nuclide	Ci	Bq	μCi/cm³	Bq/cm³
Co-60	1.0000e+000	3.7000e+010	4.4092e+002	1.6314e+007

Buildup: The material reference is Source Integration Parameters	
Radial	10
Circumferential	10
Y Direction (axial)	20

Results					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	6.035e+06	1.793e+02	4.753e+02	3.463e-01	9.177e-01
1.1732	3.700e+10	2.730e+06	6.061e+06	4.879e+03	1.083e+04
1.3325	3.700e+10	3.389e+06	7.195e+06	5.880e+03	1.248e+04
Totals	**7.401e+10**	**6.119e+06**	**1.326e+07**	**1.076e+04**	**2.331e+04**