UNIVERSITY OF ARIZONA
ACTIVATION ANALYSIS
AND COMPONENT CHARACTERIZATION

Report 08-125D-RE-122
Revision 0

May 2009

WMG Project
08-125D

Prepared for:
Enercon

Prepared by:

WMG

16 Bank Street
Peekskill, NY 10566

FOREWORD

This report summarizes the activation analysis work performed by WMG, Inc. to support Enercon in the preparation of the decommissioning plan for the University of Arizona's TRIGA Reactor at the University's Nuclear Reactor Library. This work was performed by WMG, Inc. under Enercon' Work Order No. WMG001.

TABLE OF CONTENTS

APPENDICES

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

In February 2009, Enercon Services Inc engaged WMG to support preparation of the Decommissioning Plan for the University of Arizona's Nuclear Reactor Laboratory by performing an activation analysis of the Training, Research, Isotope production – General Atomic (TRIGA) research reactor. Phase 1 of the contract awarded to WMG includes the activated component characterization of the University of Arizona TRIGA research reactor using demonstrated and approved methodologies. This report summarizes the results of WMG's analytical methodology used to complete this phase of work and presents the results in terms of activation product content within the activated reactor components.

The preliminary results presented in this report will require further normalization after representative surveys have been obtained from the facility in accordance with the survey plan to be developed with the decommissioning plan. The reported results are considered preliminary until the calculated activation activities are normalization to measured dose rates, and surface contamination, which covers the surfaces of the reactor components, is evaluated.

Based on WMG's extensive experience, the preliminary activation results are conservative and provide a reasonable basis for decommissioning planning. In addition, surface contamination typically does not dictate the NRC waste classification of highly activated reactor components. However, the surface contaminant activity (which can contain activation products, fission products and transuranics) can be significant for reactor components with relatively low activation activities and must be considered prior to classification as Low Level Radioactive Waste (LLRW) for disposal in accordance with 10 CFR Part 61.

The relevant activation product radionuclide concentrations determined from these results are used to classify the activated reactor components in accordance with 10 CFR Part 61. The activation product scaling factors are also presented in this report and are used to quantify the hard-to-detect radionuclide concentrations important to final classification under 10 CFR Part 61. The reported scaling factors are Co-60 based and considered final since the ratios of impurities in the initial material compositions are assumed to be fixed and consistent throughout the components, as shown in Appendix B. Therefore, the individual radionuclide concentrations from neutron activation are directly proportional to the Co-60 concentration.

The neutron transport and activation analysis methods used for this project are discussed in Section 2. The component specific results in terms of estimated activation activity and 10 CFR Part 61 classification as of July 1, 2009 (6 months cooling time) are presented in Section 3. The packaging requirements for the activated components are discussed in Section 4. References are presented in Section 5.

Historical summary

The University of Arizona research reactor is a TRIGA swimming pool type reactor designed and constructed by General Atomic (GA). The reactor was constructed at the University of Arizona in 1958 and first went into operation in December of that year. The initial core loading consisted of 61 aluminum clad fuel elements and the licensed power of 10 kW thermal with operation at 30 kW possible for short durations. Subsequently, two additional aluminum clad fuel elements were obtained to allow operation at higher power and the licensed power was increased to 100kw.

In May 1972, upgrades to the control console, control rod drives and bridge were installed. In December of 1972, 87 partially used stainless steel clad TRIGA fuel elements were obtained through a grant from the AEC which allowed for operation in the pulse mode. The original equipment and aluminum clad fuel were transferred to the University of Utah.

The University of Arizona TRIGA reactor has continued operating since that time and the facility operating history forms the basis for the activation analysis presented in this document. Facility operating records through December of 2008 are included in this report.

2.0   NEUTRON TRANSPORT AND ACTIVATION METHODOLOGY

The ANISN (Ref. 1) computer program is used to estimate the neutron flux levels and energy spectrum at various radial and axial locations of interest in the reactor and reactor systems. ANISN solves the one dimensional Boltzmann transport equation for slab, cylindrical and spherical geometries. The BUGLE-96 (Ref. 2) cross-section library was used in the ANISN calculations. BUGLE-96 is a coupled 47 neutron, 20 gamma-ray group cross section library derived from ENDF/B-VI.   The resultant fluxes are normalized to the reference neutron flux levels provided by the University in the following references:

- General Atomics Report "Calculated Fluxes and Cross Sections for TRIGA Reactors", GA-4361, August 1963. (Ref. 3)

- TRIGA Reactor Description Report, University of Arizona TRIGA Mark I Research Reactor, revised January 1977. (Ref. 4)

- University of Arizona Flux Test Report, "Measurement of Absolute Thermal Neutron Flux at the Pool Perimeter of the University of Arizona Research Reactor," dated Dec 1, 2008. (Ref. 8)

A summary of the neutron flux normalization points is included in Section 2.2. The normalized fluxes and initial material compositions shown in Appendix B are used as inputs to the ORIGEN2.2 (Ref. 5) computer program to perform activation analysis on individual components.

The same ANISN/ORIGEN methodology has been used to support decommissioning activities at 15 reactors, including Zion, Rancho Seco, San Onofre Nuclear Generating Station (SONGS) Unit 1, Maine Yankee, Connecticut Yankee, Millstone 1, the Brookhaven High Flux Beam Reactor (HFBR) and Medical Research Reactor (BMRR), Shoreham, Yankee Rowe, Trojan, and Saxton. The methodology has been refined and benchmarked over the years and has been found to provide reasonable characterization results for the components of interest.

Several assumptions were made for the activation analysis, based upon information provided by the University or available from similar projects performed by WMG in the past. These assumptions are stated in the subsequent sections describing each portion of the activation analysis methodology.

2.1 Input Parameters

Use of the ANISN and ORIGEN2 computer programs require accurate modeling of the reactor components in terms of their physical characteristics and neutron exposure histories.

2.1.1 Material Compositions

Elemental material compositions are required inputs for both the ANISN and ORIGEN2 computer programs. A summary of the initial material compositions is included in the appendices. Most reactor components are Aluminum Alloy 6061-T6, stainless steel (assumed Type 304) fasteners. The reactor tank is constructed of a carbon steel liner interior to a poured concrete pit, with a layer of gunite sprayed on the inner surfaces of the carbon steel shell. The initial material composition data for SS304, carbon steel and concrete are taken from NUREG CR 3474 (Ref. 6).

Other material types present in the reactor core components were also incorporated into the neutron transport and activation models. These additional materials include standard AGOT graphite, zirconium hydride and boron carbide. The graphite composition is compiled from work performed for the Brookhaven National Laboratories' research reactors. The Aluminum Alloy 6061 is compiled from work performed at the University of Michigan and the NASA Plumbrook facility. The boron carbide composition is taken from information provided for analysis of components from commercial US power plants.

2.1.2 Operating History

A detailed operating and power history was provided by the University for the TRIGA reactor. The information was provided in terms of kW-hrs produced per academic year (July 1st to June 30th) from start-up until June 30, 2000, then on a monthly basis through December 2008. The cumulative power produced over the lifetime of the reactor through December 2008 was also provided (243,054.6 kW-hrs).

An irradiation power history was compiled from this information for each academic year of operation, which is considered a "cycle of operation" in this analysis. The total number of Effective Full Power Days (EFPDs) per cycle was calculated using a full power of 100 kW. Furthermore, an 8 hour per day use is assumed to be "full use". Finally, the number of days per cycle is determined by multiplying the calculated EFPDs (power in Kw-hrs divided by 100 kW) by three. The cycle start date for each cycle is July 1st and the cycle stop date is determined by adding the number of cycle days. The decay time between cycles is from the cycle stop date to July 1st of the following year. The operating history is summarized in Table 2-1 below.

Table 2-1
University of Arizona Operating History

Cycle	Start (Date)	Stop (Date)	EFPD	EFPY	Net kW-hrs T
1	07/01/71	09/01/71	21.0	0.058	50388.0
2	07/01/72	07/17/72	5.5	0.015	13130.4
3	07/01/73	07/08/73	2.4	0.006	5660.3
4	07/01/74	07/12/74	3.9	0.011	9398.9
5	07/01/75	07/12/75	3.8	0.010	9023.0
6	07/01/76	07/22/76	7.1	0.019	17070.9
7	07/01/77	07/09/77	2.8	0.008	6831.8
8	07/01/78	07/12/78	4.0	0.011	9574.8
9	07/01/79	07/16/79	5.1	0.014	12163.2
10	07/01/80	07/14/80	4.3	0.012	10402.0
11	07/01/81	07/08/81	2.6	0.007	6220.0
12	07/01/82	07/08/82	2.4	0.007	5817.0
13	07/01/83	07/11/83	3.6	0.010	8585.0
14	07/01/84	07/06/84	1.8	0.005	4327.0
15	07/01/85	07/05/85	1.6	0.004	3794.0
16	07/01/86	07/10/86	3.1	0.008	7333.9
17	07/01/87	07/11/87	3.3	0.009	8011.7
18	07/01/88	07/08/88	2.5	0.007	6031.2
19	07/01/89	07/06/89	1.8	0.005	4273.0
20	07/01/90	07/09/90	3.0	0.008	7184.3
21	07/01/91	07/06/91	1.9	0.005	4496.7
22	07/01/92	07/06/92	1.7	0.005	4141.6
23	07/01/93	07/05/93	1.6	0.004	3722.3
24	07/01/94	07/04/94	1.2	0.003	2928.9
25	07/01/95	07/04/95	1.3	0.004	3179.7
26	07/01/96	07/03/96	0.9	0.002	2188.4
27	07/01/97	07/03/97	0.9	0.003	2259.2
28	07/01/98	07/02/98	0.6	0.002	1491.7
29	07/01/99	07/04/99	1.0	0.003	2471.9
30	07/01/01	07/04/01	1.1	0.003	2552.0
31	07/01/02	07/02/02	0.4	0.001	952.9
32	07/01/03	07/02/03	0.5	0.001	1161.5
33	07/01/05	07/02/05	0.6	0.002	1474.2
34	08/01/06	08/03/06	1.0	0.003	2337.1
35	12/27/08	12/31/08	1.0	0.003	2476.1
Char Date	07/01/09				
Totals	07/01/71	12/31/08	101.3	0.28	243054.6

Note: "Cycle 1" includes the integrated kW-hrs for the years of operation from 1958 up to the refueling and refurbishment at the end of 1971. This approach is provides a conservative estimate of the activity attributable to the initial core.

2.2 ANISN Discrete Ordinates Neutron Transport Calculations

The ANISN code was used to develop one-dimensional radial and axial neutron transport models. The ANISN cylindrical source geometry was used for the radial models, and the slab geometry was used for the axial models. Both radial and axial models were forward solutions using a P_3 maximum order of scatter and an S_8 order of angular quadrature.

2.2.1 ANISN Radial Models

The ANISN radial model consists of the reactor core, graphite reflector, reactor tank gunite, reactor tank carbon steel liner, and outer poured concrete. All air and water gaps were included in the model.

The reactor core section of the radial ANISN model is broken into two homogenized regions. The first region consists of the 71 positions in the "A-Ring" through "E-Ring", with 56 fueled positions, the transient control rod (C10), the regulating control rod (C4), the shim control rod (D10), the fueled follower control rod (D1), and associated guide tubes. The B_4C absorber from the control rods is not included in the transport models. The B_4C absorber material has a localized impact on the thermal neutron flux in the core. Inclusion of the absorber in the homogenized regions of the transport models results in excessively low calculated thermal flux levels in the core regions. The second core region represents the 30 positions in the "F-Ring" with fueled elements in 28 locations and hollow elements in 2 locations to simulate the locations occupied by the Fast Flux Irradiation (FIr) facility and rabbit tube specimen irradiation facility.

The radial models do not include components such as the thermal neutron irradiation facility and the radiography beam tube. These components do have a localized impact on the neutron flux and spectrum incident on the reactor tank and concrete. However, the activation levels in these regions are low enough that localized hotspots do not change the overall approach to removal and disposal of these components, nor do they approach any regulatory limits.

The ANISN radial flux results are normalized to published flux values from the General Atomics Report (Ref. 3), the fast flux (greater than 10 keV) specified for the Fast Irradiation Facility) Fir in the TRIGA Reactor Description Report (Ref. 4), and measured thermal flux levels from the Flux Test Report (Ref. 8). A comparison of calculated to reference flux values is provided in Table 2-4. The ANISN fluxes are normalized to the core thermal fluxes General Atomics Report and the measured thermal flux at the reactor tank wall. Other reference fluxes, included for information only, show the flux levels in other regions to be conservative. The radial thermal flux distribution is shown in Figure 2-1. The calculated neutron flux in each region is provided in Table 2-2 below.

Table 2-2
University of Arizona ANISN Radial Model
Calculated Flux Levels per Region

Zone #	Region Description	Radius (cm)	Total Thermal Flux (n/cm2-sec)	Total Flux (n/cm2-sec)
1	Core Region 1	17.812	1.29E+12	5.485E+12
2	Core Region 2	21.755	1.27E+12	4.551E+12
3	Water Gap 1	22.225	1.72E+12	4.432E+12
4	Reflector Inner Wall	22.86	1.77E+12	4.403E+12
5	Air Gap 1	23.178	1.77E+12	4.374E+12
6	Reflector Graphite	53.658	1.47E+12	2.420E+12
7	Air Gap 2	53.975	8.54E+11	1.002E+12
8	Reflector Outer Wall	55.245	8.44E+11	9.821E+11
9	Water Gap 2	99.06	5.20E+10	5.771E+10
10	Tank Gunite Coating	104.14	1.79E+07	3.975E+07
11	Steel Tank Wall	104.775	1.10E+07	3.050E+07
12	Poured Concrete	125.095	5.58E+06	1.706E+07

Figure 2-1
University of Arizona ANISN Radial Thermal Flux Distribution
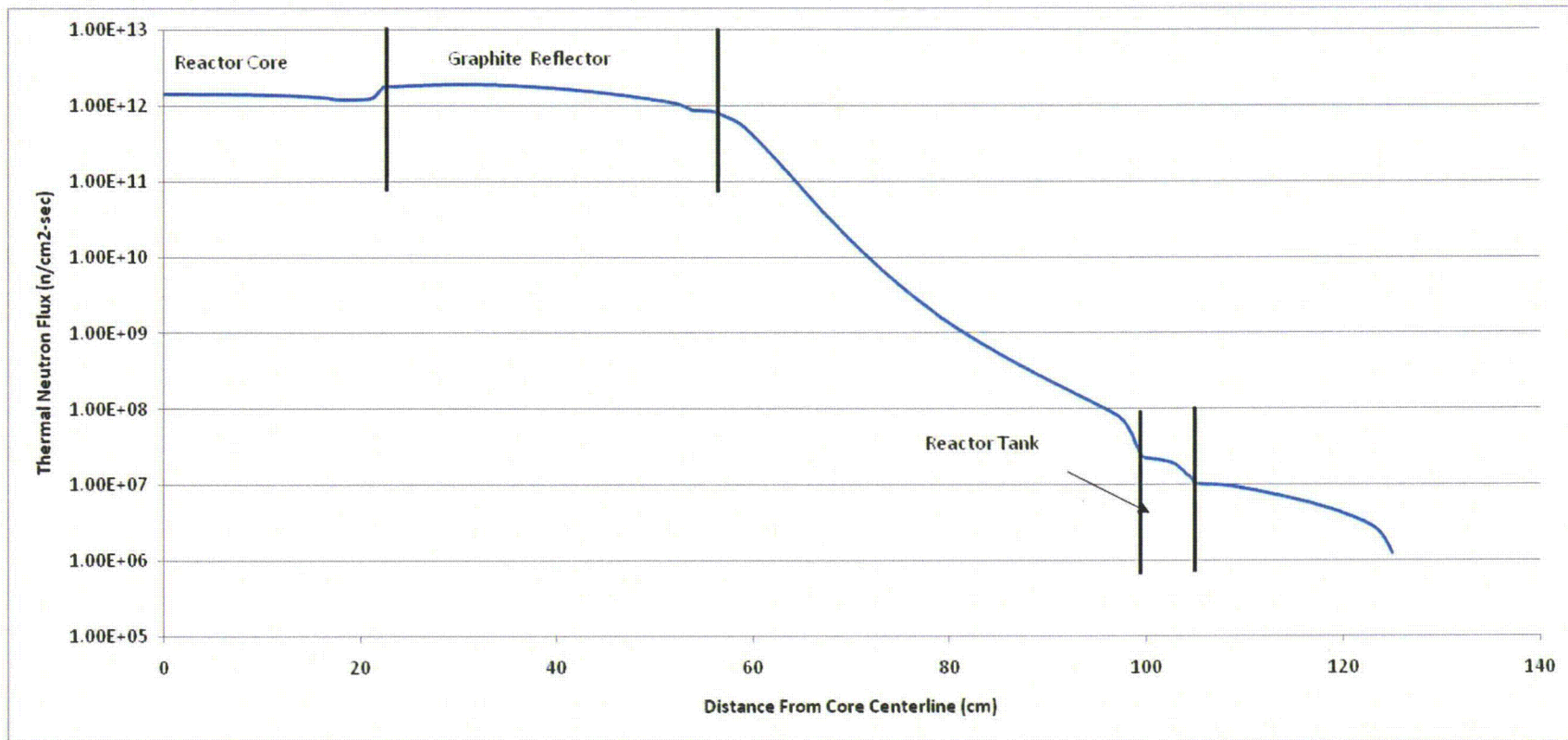
2.2.2 ANISN Axial Models

The ANISN axial component model is more complex than the radial models because of the number of components present and the significant variations in water to metal volume fractions both above and below the active the reactor core. All void and water gaps were included in the model.

The volume averaged fuel parameters of the two radial core regions were used as a baseline for the core fuel regions in the axial models. Each region is homogenized to include the portions of the reactor components within the core equivalent radius and coolant water. The components included the fuel elements, transient control rod and guide tube, fuel follower control rods (regulating and shim rods), the top and bottom grids, and the thimbles. Please note, that the axial model does include the B_4C absorber from the control rods, leading to a reduced thermal flux in the core region and regions above the core. For this reason, the activation calculations for core components must use the radial model results, and not the axial model results.

The ANISN axial flux results are normalized to the average measured flux along the pool floor (Ref. 8). The axial thermal flux distribution is shown in Figure 2-2. The calculated neutron flux in each region is provided in Table 2-3 below. A comparison of calculated to reference flux values is provided below in Table 2-4.

Table 2-3
University of Arizona ANISN Axial Model
Calculated Flux Levels per Region

Zone #	Region Description	Radius (cm)	Total Thermal Flux (n/cm2-sec)	Total Flux (n/cm2-sec)
1	Upper Water Region 1	60.96	9.66E+05	1.432E+06
2	Upper Water Region 2	91.44	8.56E+07	1.216E+08
3	Upper Water Region 3	106.68	1.39E+09	1.895E+09
4	Upper Water Region 4	121.92	1.08E+10	1.666E+10
5	Upper Poison	127.635	1.68E+09	3.349E+10
6	Fuel Element Top End Fixture	132.099	3.39E+09	1.005E+11
7	Top Grid Plate	134.004	1.55E+09	1.856E+11
8	Upper Water Gap	134.798	3.39E+09	2.350E+11
9	Fuel Element Top Plug	136.068	3.34E+09	2.897E+11
10	Upper Graphite Reflector	144.958	8.93E+09	6.713E+11
11	Active Fuel Region	183.058	1.16E+11	2.082E+12
12	Lower Graphite Reflector	191.948	3.92E+11	1.116E+12
13	Fuel Element Bottom Plug	193.218	1.80E+11	4.830E+11
14	Fuel Element Bottom End Fixture	197.663	1.41E+11	3.025E+11
15	Bottom Grid Plate	199.568	1.03E+11	1.928E+11
16	Lower Thimble Tube Region 1	216.713	5.09E+10	7.276E+10
17	Lower Thimble Tube Region 2	240.843	2.49E+09	3.382E+09
18	Lower Thimble Bottoms	242.43	2.25E+08	3.184E+08
19	Lower Water Gap	251.32	1.16E+08	1.689E+08
20	Tank Base Gunite Coating	261.48	2.85E+07	6.081E+07
21	Steel Tank Base	262.115	1.59E+07	4.543E+07
22	Concrete Tank Base	292.595	1.56E+07	3.815E+07

Table 2-4
University of Arizona ANISN Results
Calculated & Reference Neutron Flux Comparison

Table of Normalization Points		Ref	WMG	Ref / WMG
1	Thermal in Susan (33cm)	7.000E+11	1.117E+12	6.26E-01
2	> 10KeV Flux @ Fir	1.305E+12	2.835E+12	4.60E-01
GA Report, Table 8				
3	Core Center Thermal (r = 0cm)	1.954E+12	1.405E+12	1.39E+00
4	D Ring Thermal (r = 12cm)	1.411E+12	1.350E+12	1.05E+00
5	Fueled F Ring Thermal (r = 20 cm)	9.871E+11	1.187E+12	8.32E-01
6	Unfueled F Ring Thermal (r = 20 cm)	1.486E+12	1.724E+12	8.62E-01
7	Lazy Susan Thermal (r = 32.5cm)	7.308E+11	1.866E+12	3.92E-01
8	Outside Graphite Total (r = 54.2cm)	3.188E+11	1.002E+12	3.18E-01
9	Outside Graphite Thermal (r = 54.2cm)	2.711E+11	8.539E+11	3.18E-01
Flux Test Report				
10	Reactor Tank Side Wall	2.022E+07	2.322E+07	8.71E-01
11	Reactor Tank Base	4.009E+07	4.009E+07	1.00E+00

Note: The average ratio of reference to calculated flux for points 3 to 6 & 10 are used for normalization of radial results. Point 11 used for normalization of axial fluxes. The other points are included for information only.

Figure 2-2
University of Arizona ANISN Axial Thermal Flux Distribution
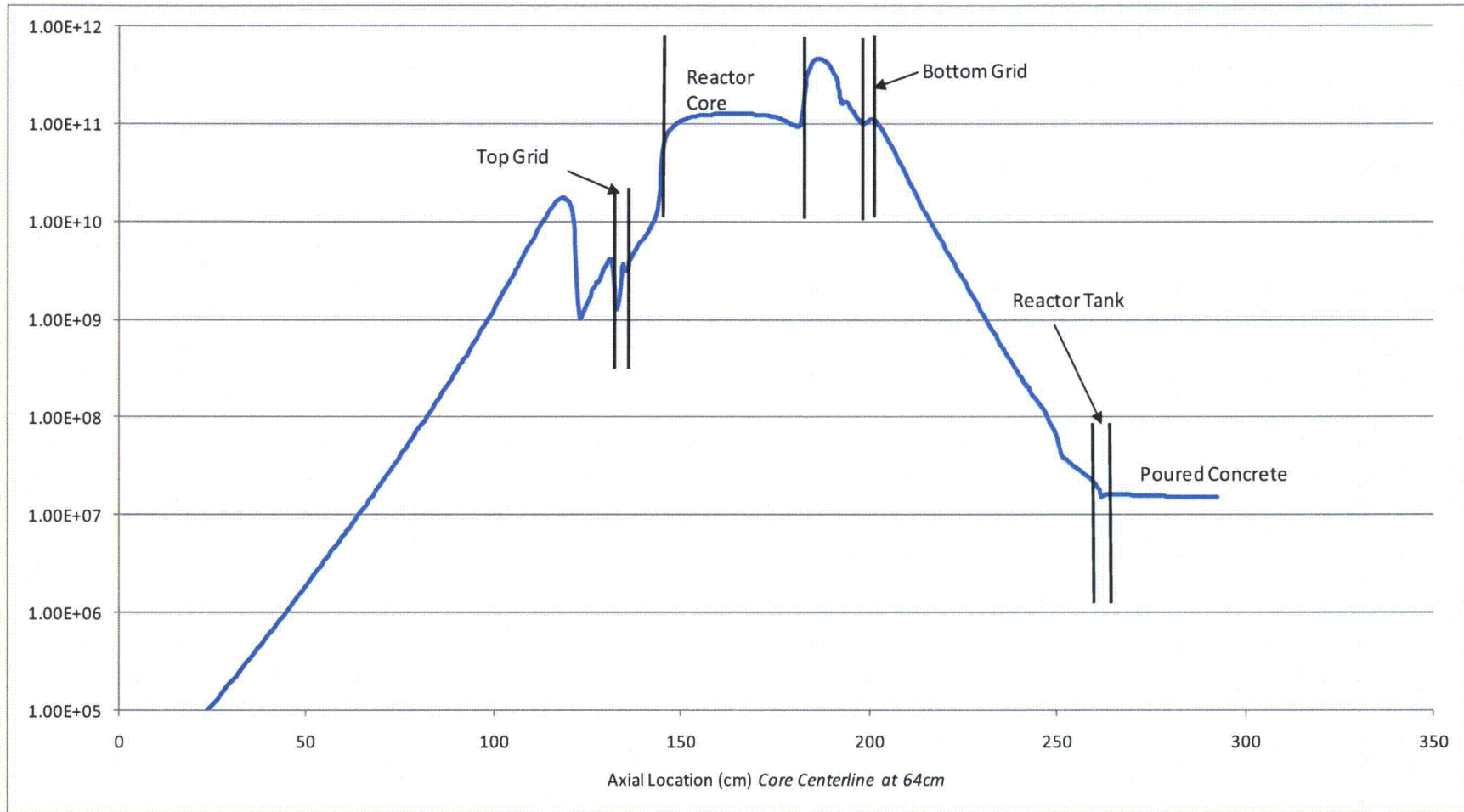
2.3 ORIGEN 2 Neutron Activation Calculations

The ORIGEN2 computer code, version 2.2, was used to calculate the activation and depletion of radionuclides in components exposed to neutron flux. Each component was irradiated based on the reactor operating history summarized in Table 2-1 using the appropriate flux as determined from the ANISN transport models and the initial material compositions shown in Appendix B.

The ORIGEN specific activity is calculated in terms of curies per gram for radial components using the core centerline flux levels and spectrum. The ANISN axial flux results are used to calculate and weight the specific activity as a function of axial distance from the core, when components reside in more than one region in the transport model.

Reactor components such as the reactor tank and graphite reflector are constructed of different materials. For example, the reactor tank assembly consists of a carbon steel liner with an interior 2-inches layer of gunite and a poured concrete on the exterior. The neutron activation differs significantly for each material and due to the relative distance from the reactor core. The results shown in Section 3 for the assembly are a summation of the material constituents of each component.

3.0 ESTIMATED COMPONENT RADIOACTIVITY

This section presents the estimated activity for each reactor component of interest as of July 1, 2009 (6 months cooling time). These preliminary estimates are based on activation analysis results and, as stated previously in this report, are normalized to the reference neutron flux levels provided by the University. The activities presented in this section represent the activation products only.

3.1 Overview

The activation results of all reactor components presented in this section are based on ORIGEN output using ANISN generated flux levels and energy spectrum. The ORIGEN output is in terms of specific activity (Ci/g) and the individual nuclide activities are calculated using the activated component weight and nuclide specific activity. Normalization to measured dose rates will change the activation activity for each component but will not change the component-specific scaling factors. Based on WMG's past experience, normalization to measured dose rates at the time of actual decommissioning is expected to reduce component total activity.

Table 3-1 presents a summary of the preliminary results in terms of waste weight, waste volume, total activation activity and materials of construction. The estimated activity for all reactor components is approximately 5,160 millicuries as of July 1, 2009. Component waste weights, waste volumes and materials of construction were determined from the drawings and data provided by the University. Components located at relatively large distances from the reactor core, such as the cooling coils, control rod drives and control rod extension shafts, are not included in the activation. Please note that the waste volumes included in this section of the analysis are displacement volumes and do not reflect the "as-packaged" for disposal volumes. This will be addressed further in Section 4.0.

All reactor components are NRC Class A waste and suitable for disposal as LLRW. The Class A LLRW components can be largely packaged intact and segmented for packaging efficiency to minimize the total disposal volume at an existing commercial disposal site. This approach will be discussed in greater detail in Section 4.0.

TABLE 3-1
University of Arizona TRIGA
Component Characterization Summary

Component	Weight (lbs)	Waste Volume (ft³)	Total Activity (Ci)	Materials of Construction
Reactor Structural/Peripheral Components				
Reactor Concrete Pit	28,078	195.6	8.90E-04	Concrete
Reactor Tank*	10,174	56.2	1.36E-03	Carbon Steel & Gunite
Reflector Assembly	3,007	17.8	4.00E+00	Al (6061-T6) Frame w/ Graphite Filler & 4 large SS bolts
Thimble Tubes (4 Tubes)	4.3	0.025	3.12E-03	Al (6061-T6) w/ 2 screws each
Core Support**	39.2	0.233	6.24E-04	Aluminum (assumed)
Subtotals	*~41,300*	*~270.0*	*4.01E+00*	
Reactor Core Components				
Bottom Grid Plate (w/ 4 "Captive Screws")	14.2	0.083	1.35E-02	Al (6061-T6) w/ SS Fasteners
Top Grid Plate (w/ 4 "Captive Screws")	11.7	0.068	3.65E-03	Al (6061-T6) w/ SS Fasteners
Transient Rod Guide Tube	1.7	0.010	2.75E-03	Al (6061-T6) w/ SS Fasteners
Subtotals	*27.6*	*0.161*	*2.26E-02*	
Experimental Facilities				
Thermal Neutron Irradiation Facility	785.0	6.30	3.77E-02	Aluminum & Graphite
Isotope Production Facility (a.k.a. Lazy Susan)	139.7	0.826	9.75E-01	Aluminum & Stellite
Neutron Radiography Unit	350.0	2.08	1.14E-01	Aluminum w/ Lead & High Boron Glass
Fast Flux Irradiation Facility	9.1	0.043	1.30E-03	Aluminum w/ Gold, Cadmium & Sodium Borate
Rabbit Tube Assembly	16.5	0.098	3.27E-03	Aluminum
Subtotals	*1,309*	*9.39*	*1.13E+00*	
Totals	**~42,630**	**~280.0**	**5.16E+00**	

* Reactor Tank consists of 1/4" Carbon Steel Wall with Gunite Liner
** Core Support consists of "Platform" and "Pad - Reflector"

3.2 Component Radioactivity

Individual component estimated activation activities are decay corrected to July 1, 2009. The reactor components are divided into three groups: the Reactor Structural/Peripheral Components, the Experimental Facilities and the Reactor Core Components. The Reactor Core Components hold the fuel elements in place (Grid Plates) or reside within the reactor core (Transient Rod Guide Tube). These components have relatively low specific activities and may be disposed of intact or segmented and selectively packaged for disposal. Based on the empirical data from previous decommissioning activities, the additional effect of surface contaminants could increase component dose rates marginally but is unlikely to change waste classification.

The radionuclide activities presented in this report are those considered "significant" per the guidance presented in NUREG/BR-0204 "Instructions for Completing NRC's Uniform Low-Level Radioactive Waste Manifest" (Ref. 7). In summary, a radionuclide is considered significant when it is contained in the waste in concentrations:

- Greater than 1% of the total activity of the component
- Greater than 1% of the Class A limit for nuclides listed in 10 CFCR 61.55
- Greater than 7.0 μCi/cc for nuclides not listed in 10 CFR 61.55

As discussed previously, surveys of the reactor components are necessary to update the preliminary results presented herein to provide final characterization results.

3.2.1 Reactor Structural/Peripheral Components

The reactor structural and peripheral components include the Reactor Concrete Pit, Reactor Tank, Reflector Assembly, Thimbles and Core Support. The Reactor Tank consists of the ¼-inch carbon steel wall and the 2-inch thick gunite layer. The Core Support consists of the structural aluminum platform and pads connecting the reactor and reflector assemblies to the floor of the reactor tank. The activation analysis results for these five (5) components regions are presented below.

Reactor Concrete Pit

The reactor concrete pit consists of a 12-inch thick concrete slab upon which the reactor tank rests and an approximately 8-inch poured concrete cylinder formed by a corrugated steel outer form and the ¼-inch thick steel reactor tank as an inner form. The total volume of the concrete reactor pit is calculated to be 376.4 ft^3 with an associated waste weight of 54,200 lbs. Based upon the measured axial flux data (Ref. 8), the activated region of the reactor concrete pit has a height of 7-feet high, +/- 3-½ feet from the core mid-plane. The total weight of concrete in the activated region is 28,078 lbs with displacement volume of 195.6 ft^3. The component activities and scaling factors are summarized in Table 3-2 below.

Table 3-2
Reactor Concrete Pit
Characterization Results

Component: **Reactor Concrete Pit**				
Waste Weight/Volume:	28,078 lbs			195.6 ft3
	1.27E+07 g			5.54E+06 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	5.75E-04	1.04E-04	4.31E+01		2.60E-06
C-14	4.73E-07	8.54E-08	3.54E-02	1.07E-07	
Ca-45	1.51E-04	2.73E-05	1.13E+01		3.90E-08
Cr-51	2.99E-07	5.40E-08	2.24E-02		7.71E-11
Mn-54	1.63E-06	2.94E-07	1.22E-01		4.20E-10
Fe-55	1.13E-04	2.03E-05	8.44E+00		2.91E-08
Fe-59	1.70E-06	3.07E-07	1.27E-01		4.38E-10
Co-58	0.00E+00	0.00E+00	0.00E+00		0.00E+00
Co-60	1.34E-05	2.41E-06	1.00E+00		3.44E-09
Ni-59	3.68E-09	6.64E-10	2.76E-04	3.02E-11	
Ni-63	4.29E-07	7.74E-08	3.21E-02		2.21E-08
Zn-65	6.73E-07	1.22E-07	5.04E-02		1.74E-10
Nb-94	5.99E-10	1.08E-10	4.49E-05	5.41E-09	
Tc-99	6.30E-12	1.14E-12	4.72E-07	3.79E-12	
Eu-152	3.23E-05	5.83E-06	2.42E+00		
Total	8.90E-04			1.12E-07	2.69E-06

Reactor Tank

The reactor tank consists of a 6-foot, 10-inch diameter carbon steel tank with ¼-inch thick walls. The inside of the steel tank is covered on the sides by a layer of Gunite approximately 2-inches thick and on the bottom by a layer approximately 4–inches thick. The entire inner Gunite surface is coated with Amercoat epoxy based paint. The total weight of the carbon steel reactor tank liner is calculated as 4,950 lbs. The volume of the Gunite layer is calculated to be approximately 92.63 ft³ and the waste weight is estimated to be 10,400 lbs, using a density of 126 pounds per cubic foot. Therefore, the reactor tank has a total weight of 15,350 lbs and a total waste volume of 92.63 ft³. Based upon the measured axial flux data (Ref. 8), the activated region of the reactor tank has a height of 7-feet high, +/- 3-½ feet from the core mid-plane. The total weight of steel and concrete in the activated region is 10,174 lbs with displacement volume of 56.2 ft³. The component activities and scaling factors are summarized in Table 3-3 below.

Table 3-3
Reactor Tank
Characterization Results

Component: Reactor Tank				
Waste Weight/Volume:	10,174 lbs			56.23 ft3
	4.61E+06 g			1.59E+06 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	4.82E-04	3.03E-04	1.09E+01		7.57E-06
C-14	4.48E-07	2.81E-07	1.02E-02	3.52E-07	
Ca-45	1.26E-04	7.90E-05	2.85E+00		1.13E-07
Cr-51	1.18E-06	7.41E-07	2.68E-02		1.06E-09
Mn-54	1.03E-05	6.48E-06	2.34E-01		9.26E-09
Fe-55	6.47E-04	4.06E-04	1.47E+01		5.80E-07
Fe-59	9.77E-06	6.14E-06	2.22E-01		8.77E-09
Co-58	9.66E-07	6.06E-07	2.19E-02		8.66E-10
Co-60	4.41E-05	2.77E-05	1.00E+00		3.96E-08
Ni-59	1.27E-07	8.01E-08	2.89E-03	3.64E-09	
Ni-63	1.49E-05	9.33E-06	3.37E-01		2.66E-06
Zn-65	7.31E-07	4.59E-07	1.66E-02		6.56E-10
Nb-94	1.00E-09	6.29E-10	2.27E-05	3.14E-08	
Tc-99	4.25E-12	2.67E-12	9.64E-08	8.90E-12	
Eu-152	2.65E-05	1.66E-05	6.01E-01		
Total	1.36E-03			3.87E-07	1.10E-05

Reflector Assembly

The reactor core is located inside a cylindrical graphite reflector dimensionally 1-foot thick and 22-inches deep. The graphite reflector is completely canned in Aluminum (6061-T6). The reflector is designed with a well channel to allow for rotation of the specimen rack. The core bottom plate rests on a flange in the lower reflector and the core top plate rest on the upper section of the reflector. Two Aluminum (6061-T6) channels are assumed to be welded to the underside of the aluminum reflector enclosure to receive the four stainless steel leveling bolts which rest on the core support platform structure. The reflector assembly represents approximately 3000 lbs. of activated graphite and aluminum with waste volume of 17.8 ft^3. The component activities and scaling factors are summarized in Table 3-4 below.

Table 3-4
Reflector Assembly
Characterization Results

Component: **Reflector Assembly**					
Waste Weight/Volume:	3,007 lbs				17.84 ft3
	1.36E+06 g				5.05E+05 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	2.15E-06	4.25E-06	7.29E-07		1.06E-07
C-14	2.38E-03	4.71E-03	8.08E-04	5.88E-03	
Cr-51	1.83E-02	3.63E-02	6.23E-03		5.18E-05
Mn-54	5.36E-03	1.06E-02	1.82E-03		1.51E-05
Fe-55	9.22E-01	1.82E+00	3.13E-01		2.61E-03
Fe-59	1.37E-02	2.72E-02	4.67E-03		3.88E-05
Co-58	3.37E-04	6.66E-04	1.14E-04		9.52E-07
Co-60	2.94E+00	5.82E+00	1.00E+00		8.32E-03
Ni-59	1.42E-04	2.81E-04	4.83E-05	1.28E-05	
Ni-63	1.66E-02	3.29E-02	5.65E-03		9.40E-03
Zn-65	7.53E-02	1.49E-01	2.56E-02		2.13E-04
Nb-94	1.27E-05	2.52E-05	4.33E-06	1.26E-03	
Tc-99	6.51E-10	1.29E-09	2.21E-10	4.30E-09	
Total	4.00E+00			7.16E-03	2.06E-02

Thimble Tubes

There are four (4) aluminum thimble tubes. There two types of thimbles tubes, which are identical expect for the number of holes per tube. The thimble tubes have an outside diameter of 2-inches, a wall thickness of 0.083-inches and a length of 16-5/16 inches. The difference between the Type 1 and Type 2 thimbles is the number of holes in each tube. Each thimble is attached to the bottom of the lower grid plate with two (2) stainless steel socket head cap screws. The thimble tubes have a total weight approximately 4.3 lbs with a waste volume of 0.025 ft³. The component activities and scaling factors are summarized in Table 3-5 below. The table below includes results for all four aluminum thimble tubes and associated stainless steel screws.

Table 3-5
Thimble Tubes
Characterization Results

Component: Thimble Tubes					
Waste Weight/Volume:	4.3 lbs				0.025 ft3
	1.96E+03 g				6.99E+02 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	1.26E-06	1.80E-03	7.83E-04		4.50E-05
C-14	2.88E-06	4.12E-03	1.79E-03	5.15E-04	
Cr-51	1.90E-04	2.71E-01	1.18E-01		3.87E-04
Mn-54	3.04E-06	4.36E-03	1.89E-03		6.22E-06
Fe-55	7.80E-04	1.12E+00	4.85E-01		1.60E-03
Fe-59	1.16E-05	1.66E-02	7.20E-03		2.37E-05
Co-58	6.14E-06	8.79E-03	3.82E-03		1.26E-05
Co-60	1.61E-03	2.30E+00	1.00E+00		3.29E-03
Ni-59	3.44E-06	4.93E-03	2.14E-03	2.24E-04	
Ni-63	3.96E-04	5.67E-01	2.46E-01		1.62E-02
Zn-65	1.14E-04	1.64E-01	7.11E-02		2.34E-04
Nb-94	2.14E-08	3.07E-05	1.33E-05	1.53E-03	
Tc-99	1.57E-10	2.25E-07	9.77E-08	7.49E-07	
Total	3.12E-03	4.46E+00		2.27E-03	2.18E-02

Core Support Assembly

The core support platform assembly is an approximately 30.25 inch square frame, with upper and lower bearing plates and stiffener gussets assumed to be Aluminum (6061-T6). This assembly incorporates the pad reflector with concave depressions for positioning of the stainless steel reflector assembly leveling bolts. The core support assembly represents 39.2 lbs. of activated aluminum with a waste volume of 0.233 ft^3. The component activities and scaling factors are summarized in Table 3-6 below.

Table 3-6
Core Support Assembly
Characterization Results

Component: Core Support Assembly					
Waste Weight/Volume:	39.21 lbs				0.233 ft3
	1.78E+04 g				6.59E+03 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	4.41E-18	6.70E-16	8.93E-15		1.68E-17
C-14	1.12E-06	1.70E-04	2.27E-03	2.13E-05	
Cr-51	1.29E-05	1.95E-03	2.60E-02		2.79E-06
Mn-54	1.64E-07	2.48E-05	3.31E-04		3.55E-08
Fe-55	5.00E-05	7.59E-03	1.01E-01		1.08E-05
Fe-59	7.41E-07	1.13E-04	1.50E-03		1.61E-07
Co-58	1.46E-07	2.22E-05	2.96E-04		3.17E-08
Co-60	4.95E-04	7.51E-02	1.00E+00		1.07E-04
Ni-59	9.79E-08	1.49E-05	1.98E-04	6.75E-07	
Ni-63	1.13E-05	1.72E-03	2.29E-02		4.91E-05
Zn-65	5.31E-05	8.06E-03	1.07E-01		1.15E-05
Nb-94	8.48E-09	1.29E-06	1.72E-05	6.44E-05	
Tc-99	2.18E-13	3.31E-11	4.41E-10	1.10E-10	
Total	6.24E-04			8.63E-05	1.82E-04

3.2.2 Reactor Core Components

The fuel elements are supported and spaced by means of top and bottom grid plates of Aluminum 6061. The elements are spaced so that about 33% of the core volume is occupied by water. The reactor core components consist of the Bottom Grid Plate, Top Grid Plate and Transient Rod Guide Tube.

Bottom Grid Plate

The bottom grid plate is ¾-inch thick, with holes to receive the lower end-fixtures of the fuel elements. These lower end fixtures are ¼-inch diameter cylindrical projections on the bottoms of the fuel elements. A 5/8-inch shoulder is provided on the end-fixture, and the hole in the bottom grid plate is countersunk by a corresponding amount. The weight of the fuel element rests on this shoulder and not on the bottom of the end fixture, which is used only to position the fuel element as it is being put into place. The bottom grid plate is secured to an inner lip on the bottom of the graphite reflector assembly with four (4) stainless steel socket head cap screws. There are four (4) holes in the bottom grid plate to accommodate control rods. The bottom grid plate, with stainless steel screws, weighs approximately 14 pounds with a waste volume of 0.083 ft³. The component activities and scaling factors are summarized in Table 3-7 below. The table below includes results for the aluminum grid plate and associated stainless steel screws.

Table 3-7
Bottom Grid Plate
Characterization Results

Component: **Bottom Grid Plate**				
Waste Weight/Volume:	14.17 lbs			0.0830 ft3
	6.43E+03 g			2.35E+03 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	2.67E-06	1.14E-03	2.99E-04		2.84E-05
C-14	1.74E-05	7.41E-03	1.95E-03	9.26E-04	
Cr-51	5.32E-04	2.26E-01	5.95E-02		3.23E-04
Mn-54	1.74E-05	7.38E-03	1.94E-03		1.05E-05
Fe-55	2.16E-03	9.20E-01	2.42E-01		1.31E-03
Fe-59	3.23E-05	1.37E-02	3.62E-03		1.96E-05
Co-58	3.05E-05	1.30E-02	3.42E-03		1.85E-05
Co-60	8.93E-03	3.80E+00	1.00E+00		5.43E-03
Ni-59	8.27E-06	3.52E-03	9.26E-04	1.60E-04	
Ni-63	9.58E-04	4.08E-01	1.07E-01		1.16E-02
Zn-65	7.89E-04	3.35E-01	8.83E-02		4.79E-04
Nb-94	1.50E-07	6.40E-05	1.68E-05	3.20E-03	
Tc-99	6.37E-10	2.71E-07	7.13E-08	9.04E-07	
Total	1.35E-02			4.29E-03	1.93E-02

Top Grid Plate

The top grid plate is also ¾-inch thick, with 1½-inch holes for the fuel elements and control rods. The top grid plate does not support any portion of the weight of the fuel elements but does; however, support the control rod guide tubes. The holes serve only to determine the lateral position of the fuel elements and to permit withdrawal of the fuel elements from the core. Space for passage of cooling water through the top grid plate is provided by grooves cut in the top portions of the fuel element top end-fixtures. The top grid plate is secured to the top of the graphite reflector assembly with four (4) stainless steel socket head cap screws.

The top grid plate, with stainless steel screws, weighs approximately 12 pounds with a waste volume of 0.07 ft³. The component activities and scaling factors are summarized in Table 3-8 below. The table below includes results for the aluminum grid plate and associated stainless steel screws.

Table 3-8
Top Grid Plate
Characterization Results

Component: **Top Grid Plate**				
Waste Weight/Volume:	11.70 lbs			0.0683 ft3
	5.31E+03 g			1.93E+03 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	4.55E-07	2.35E-04	1.64E-04		5.87E-06
C-14	2.79E-06	1.44E-03	1.01E-03	1.80E-04	
Cr-51	8.93E-05	4.62E-02	3.23E-02		6.60E-05
Mn-54	2.98E-05	1.54E-02	1.08E-02		2.20E-05
Fe-55	3.71E-04	1.92E-01	1.34E-01		2.74E-04
Fe-59	6.20E-06	3.20E-03	2.24E-03		4.58E-06
Co-58	5.41E-05	2.79E-02	1.95E-02		3.99E-05
Co-60	2.77E-03	1.43E+00	1.00E+00		2.04E-03
Ni-59	1.33E-06	6.86E-04	4.79E-04	3.12E-05	
Ni-63	1.70E-04	8.80E-02	6.15E-02		2.51E-03
Zn-65	1.51E-04	7.81E-02	5.46E-02		1.12E-04
Nb-94	7.10E-08	3.67E-05	2.56E-05	1.83E-03	
Tc-99	1.07E-09	5.55E-07	3.88E-07	1.85E-06	
Total	3.65E-03			2.05E-03	5.08E-03

Transient Rod Guide Tube

The Aluminum (6061-T6) Transient Rod Guide Tube with stainless steel fasteners weighs approximately 1.7 lbs. with a waste volume of 0.01 ft^3. The guide tube has an outside diameter of 1½-inch thick tubes, a wall thickness of 0.065-inches and a length of 60-inches. The guide tube also includes a small amount of stainless steel in the form of a threaded rod and dowel pin. The component activities and scaling factors are summarized in Table 3-9 below. The table below includes results for the aluminum tube and associated stainless steel parts.

Table 3-9
Transient Rod Guide Tube
Characterization Results

Component: Transient Rod Guide Tube					
Waste Weight/Volume:	1.681 lbs				0.0096 ft3
	7.63E+02 g				2.73E+02 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	1.87E-09	6.86E-06	7.90E-07		1.72E-07
C-14	3.00E-06	1.10E-02	1.27E-03	1.38E-03	
Cr-51	3.27E-05	1.20E-01	1.38E-02		1.71E-04
Mn-54	7.91E-06	2.90E-02	3.34E-03		4.14E-05
Fe-55	1.29E-04	4.72E-01	5.43E-02		6.74E-04
Fe-59	2.13E-06	7.80E-03	8.98E-04		1.11E-05
Co-58	7.08E-06	2.60E-02	2.99E-03		3.71E-05
Co-60	2.37E-03	8.69E+00	1.00E+00		1.24E-02
Ni-59	2.37E-07	8.68E-04	9.98E-05	3.94E-05	
Ni-63	3.31E-05	1.21E-01	1.39E-02		3.46E-03
Zn-65	1.67E-04	6.12E-01	7.04E-02		8.74E-04
Nb-94	6.48E-08	2.38E-04	2.73E-05	1.19E-02	
Tc-99	8.93E-12	3.27E-08	3.77E-09	1.09E-07	
Total	2.75E-03			1.33E-02	1.77E-02

3.2.3 Experimental Facilities

Special neutron irradiation facilities are provided for the production of radioisotopes. These include the Thermal Neutron Irradiation Facility, the Isotope Production (aka Lazy Susan), the Neutron Radiography Unit, the Fast Flux Irradiation Facility (FIR), the Center Beam Tube Assembly and the Rabbit Tube Assembly. The results for each experimental facility are discussed in detail below.

Thermal Neutron Irradiation Facility

The Thermal Neutron Irradiation Facility consists of a completely enclosed Aluminum (6061-T6) and Graphite filled chamber dimensionally 20 inches wide, 15.4 inches deep and 23 inches high, contoured to the approximate radius of the reactor tank. The seal welded enclosure contains three 1½-inch diameter wells, centered in the width of the chamber and evenly spaced at 3-5/8-inch intervals to a depth of 17½-inches from the top plate, for introduction of specimens. The aluminum enclosure incorporates an aluminum platform which elevates the chamber 22 inches off the reactor pit floor. The entire assembly has a weight of 785 lbs and a volume of 6.30 ft^3. The component activities and scaling factors are summarized in Table 3-10 below.

Table 3-10
Thermal Neutron Irradiation Facility
Characterization Results

Component: Thermal Neutron Irradiation Facility					
Waste Weight/Volume:	785.0 lbs			6.300 ft3	
	3.56E+05 g			1.78E+05 cc	
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	3.45E-09	1.93E-08	1.27E-07		4.84E-10
C-14	2.23E-05	1.25E-04	8.18E-04	1.56E-04	
Cr-51	1.63E-04	9.16E-04	6.00E-03		1.31E-06
Mn-54	9.23E-06	5.18E-05	3.39E-04		7.39E-08
Fe-55	9.33E-03	5.23E-02	3.43E-01		7.47E-05
Fe-59	1.38E-04	7.72E-04	5.06E-03		1.10E-06
Co-58	5.62E-07	3.15E-06	2.06E-05		4.50E-09
Co-60	2.72E-02	1.53E-01	1.00E+00		2.18E-04
Ni-59	1.25E-06	6.99E-06	4.58E-05	3.18E-07	
Ni-63	1.43E-04	8.03E-04	5.27E-03		2.30E-04
Zn-65	6.68E-04	3.75E-03	2.46E-02		5.35E-06
Nb-94	9.94E-08	5.57E-07	3.65E-06	2.79E-05	
Tc-99	1.19E-12	6.68E-12	4.38E-11	2.23E-11	
Total	3.77E-02			1.84E-04	5.30E-04

Isotope Production Facility (aka 'Lazy Susan")

The Isotope Production Facility (aka "Lazy Susan") is a rotary specimen rack located in the well in the reflector can. The rotary specimen rack consists of an aluminum ring which can be rotated around the core and includes stellite bearings. Forty aluminum cups, evenly spaced, are hung from the ring and serve as irradiation specimen holders. The ring is driven by an electric motor or can be manually rotated from the top of the reactor pit, so that any one of the cups can be aligned with the single isotope-removal pipe which runs to the top of the reactor pit for installation and removal of specimens. The rotary specimen rack is completely enclosed in an aluminum welded box. The aluminum ring is located at approximately the level of the top grid plate. The specimen cups extend from the ring down to about 4-inches below the top of the active fuel region.

The Isotope Production Facility has a weight of about 140 lbs and a waste volume of 0.83 ft^3. The component activities and scaling factors are summarized in Table 3-11 below.

Table 3-11
Isotope Production Facility
Characterization Results

Component: Isotopic Production Facility				
Waste Weight/Volume:	139.7 lbs			0.826 ft3
	6.34E+04 g			2.34E+04 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
---	---	---	---	---	---
H-3	8.76E-15	3.74E-13	1.04E-14		9.36E-15
C-14	1.17E-03	4.99E-02	1.39E-03	6.24E-03	
Cr-51	1.36E-02	5.80E-01	1.62E-02		8.29E-04
Mn-54	3.26E-04	1.40E-02	3.89E-04		1.99E-05
Fe-55	5.19E-02	2.22E+00	6.19E-02		3.17E-03
Fe-59	7.74E-04	3.31E-02	9.23E-04		4.73E-05
Co-58	2.90E-04	1.24E-02	3.46E-04		1.77E-05
Co-60	8.39E-01	3.59E+01	1.00E+00		5.12E-02
Ni-59	1.02E-04	4.36E-03	1.22E-04	1.98E-04	
Ni-63	1.19E-02	5.10E-01	1.42E-02		1.46E-02
Zn-65	5.57E-02	2.38E+00	6.64E-02		3.40E-03
Nb-94	9.75E-06	4.17E-04	1.16E-05	2.08E-02	
Tc-99	3.95E-10	1.69E-08	4.70E-10	5.62E-08	
Total	9.75E-01			2.73E-02	7.33E-02

Neutron Radiography Unit

The Neutron Radiography Unit is a 12-inch diameter Aluminum (6061-T6) beam tube which permits the streaming of a near-parallel beam of thermal neutrons from the reflector area of the reactor. The unit also includes lead and high boron glass component. The unit weighs 350 lbs and has a waste volume of approximately 2.0 ft^3. The component activities and scaling factors are summarized in Table 3-12 below.

Table 3-12
Neutron Radiography Unit
Characterization Results

Component: Neutron Radiography Unit				
Waste Weight/Volume:	350.0 lbs			2.076 ft3
	1.59E+05 g			5.88E+04 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	2.50E-16	4.25E-15	2.79E-15		1.06E-16
C-14	2.10E-04	3.57E-03	2.34E-03	4.46E-04	
Cr-51	2.41E-03	4.10E-02	2.69E-02		5.86E-05
Mn-54	9.27E-06	1.58E-04	1.04E-04		2.25E-07
Fe-55	9.37E-03	1.59E-01	1.05E-01		2.28E-04
Fe-59	1.38E-04	2.35E-03	1.55E-03		3.36E-06
Co-58	8.29E-06	1.41E-04	9.26E-05		2.01E-07
Co-60	8.95E-02	1.52E+00	1.00E+00		2.18E-03
Ni-59	1.84E-05	3.13E-04	2.06E-04	1.42E-05	
Ni-63	2.11E-03	3.59E-02	2.36E-02		1.03E-03
Zn-65	9.86E-03	1.68E-01	1.10E-01		2.40E-04
Nb-94	1.47E-06	2.49E-05	1.64E-05	1.25E-03	
Tc-99	1.76E-11	2.99E-10	1.96E-10	9.96E-10	
Total	1.14E-01			1.71E-03	3.73E-03

Fast Flux Irradiation Facility

The "FIR" is a Fast Neutron Irradiation Facility which is inserted in the "F-ring" of the reactor in place of a previously installed graphite "dummy" element. It is lined with boron, cadmium and gold, all of which serve as filters for thermal neutrons while permitting the fast neutrons to pass into the sample essentially unimpeded. Samples are inserted in a thin 8-inch long, 1-inch ID tube and lowered into the core. This chamber internally contains concentric cylinders of Sodium Borate, Cadmium and Gold to a height of 10.5 inches. The assembly consists of a 1.45-inch O.D. by 0.030 inch wall cylinder of Aluminum (6061-T6). The chamber incorporates a seal welded aluminum cap on the bottom end and a flanged top end to mate with existing experiment tubes. The chamber weighs approximately 9.1 lbs with a waste volume of 0.04 ft^3.

The component activities and scaling factors are summarized in Table 3-13 below.

Table 3-13
Fast Flux Irradiation Facility
Characterization Results

Component: Fast Flux Irradiation Facility				
Waste Weight/Volume:	9.12 lbs			0.043 ft3
	4.14E+03 g			1.22E+03 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 2 Fraction
H-3	1.15E-16	9.48E-14	1.02E-13		2.37E-15
C-14	1.26E-06	1.04E-03	1.12E-03	1.30E-04	
Cr-51	1.34E-05	1.10E-02	1.19E-02		1.57E-05
Mn-54	4.28E-06	3.51E-03	3.78E-03		5.02E-06
Fe-55	5.30E-05	4.35E-02	4.69E-02		6.22E-05
Fe-59	9.07E-07	7.45E-04	8.02E-04		1.06E-06
Co-58	3.82E-06	3.14E-03	3.38E-03		4.49E-06
Co-60	1.13E-03	9.29E-01	1.00E+00		1.33E-03
Ni-59	9.39E-08	7.71E-05	8.30E-05	3.51E-06	
Ni-63	1.40E-05	1.15E-02	1.24E-02		3.28E-04
Zn-65	7.35E-05	6.03E-02	6.50E-02		8.62E-05
Nb-94	3.26E-08	2.68E-05	2.89E-05	1.34E-03	
Tc-99	4.70E-12	3.86E-09	4.15E-09	1.29E-08	
Total	1.30E-03			1.47E-03	1.83E-03

Rabbit Tube Assembly

The pneumatically operated "rabbit" tube assembly is used for the production of short half-lived isotopes. It consists of two Aluminum (6061-T6) pipes leading down through the water tank to a position at the outer edge of the core; the two tubes are joined at this point. The specimen is fed in and out of through one of the pipes and a vacuum applied to the other pipe provides the pressure difference required to inject or eject the specimen. With a total length of approximately 230 inches, the rabbit tube assembly weighs 16.5 lbs with a waste volume of 0.098 ft^3.

The component activities and scaling factors are summarized in Table 3-14 below. The table below includes results for both rabbit tubes.

Table 3-14
Rabbit Tube Assembly
Characterization Results

Component: Rabbit Tubes Assembly				
Waste Weight/Volume:	16.51 lbs			0.0979 ft3
	7.49E+03 g			2.77E+03 cc
	Total Activity (Ci)	Total Specific Activity (uCi/cc)	Co-60 Scaling Factors	Class A Table 1 Fraction	Class A Table 2 Fraction
H-3	2.91E-16	1.05E-13	1.02E-13		2.63E-15
C-14	3.19E-06	1.15E-03	1.12E-03	1.44E-04	
Cr-51	3.39E-05	1.22E-02	1.19E-02		1.75E-05
Mn-54	1.08E-05	3.89E-03	3.78E-03		5.56E-06
Fe-55	1.34E-04	4.82E-02	4.69E-02		6.89E-05
Fe-59	2.29E-06	8.25E-04	8.02E-04		1.18E-06
Co-58	9.65E-06	3.48E-03	3.38E-03		4.97E-06
Co-60	2.85E-03	1.03E+00	1.00E+00		1.47E-03
Ni-59	2.37E-07	8.55E-05	8.30E-05	3.88E-06	
Ni-63	3.53E-05	1.27E-02	1.24E-02		3.63E-04
Zn-65	1.85E-04	6.69E-02	6.50E-02		9.55E-05
Nb-94	8.24E-08	2.97E-05	2.89E-05	1.49E-03	
Tc-99	1.19E-11	4.27E-09	4.15E-09	1.42E-08	
Total	3.27E-03			1.63E-03	2.03E-03

4.0   <u>COMPONENT PACKAGING REQUIREMENTS</u>

All reactor components evaluated in this analysis are suitable for disposal as NRC Class A waste. As stated in the previous section, detailed radiation surveys must be performed to finalize the preliminary results detailed in this analysis. In addition, the radiation surveys identify any "hot spots" on the reactor components which need to be taken into consideration during packaging of the components for transport and disposal.

The reactor assembly, consisting of the grid plates, reflector assembly, core support, thimble tubes, and other tubing sections between the grid plates, can be packaged intact in a single, large capacity disposal liner such as the 8-120 and transported for disposal in a single Type A cask shipment. The cask is required due to the anticipated dose rates on the individual reactor component. In addition, the large payload capacity minimizes handling of components, reducing the time and dose necessary to package the components for disposal. Tubing sections extending above the top grid plate need to be segmented to accommodate the internal height of the disposal liner and furthermore, to increase packaging efficiency of the activated components. A packaging efficiency of up to 15% (waste volume / liner internal volume) can be achieved with segmentation of long components and careful packaging. A typical packaging efficiency of 20% is reasonable for the reactor tank steel wall and concrete.

The remainder of the components within the reactor tank can be packaged for disposal as Low Specific Activity (LSA) material. This is also true of the reactor tank, inclusive of the gunite, steel and concrete. Disposal liner and transportation package selection will be based upon the detailed radiation surveys. The total displacement volume of reactor concrete pit, tank assembly, experimental facilities and other low level activity items could be as much as 600 ft^3, with a total waste weight of up to 88,500 lbs. These values are 125% of the weight and volume of entire reactor tank and concrete pit, which is bounding. The bounding estimate should include any activated soil outside the poured concrete pit. The Co-60 activation in the soil is expected to be no more than 9.0 pCi/g at the axial centerline, below the reactor core, and 1.5 pCi/g at the reactor core mid-plane elevation. The anticipated activation levels in the soil are estimated using the boundary fluxes from the ANISN models and a standard concrete material composition.

Based upon the axial flux profile information in the flux test report, the maximum activated portion of the reactor tank steel liner, gunite and concrete pit that must be removed from the facility during decommissioning operations is approximately 7-feet, +/- 3½ feet from the core mid-plane, and the entire tank bottom. It is anticipated that following removal of these activated materials, residual concrete and steel should not possess activity levels of Co-60 in excess of the default DCGL of 3.8 pCi/g. It is highly recommended to remove the entire gunite surface, since it is contaminated due to contact with the tank water. The activation of the reactor tank steel liner, gunite and concrete may have localized hotspots in areas behind the thermal irradiation facility and the radiography beam tube. However, the activation levels in these components are low enough such that localized hotspots will not change the overall approach to removal and disposal of these components, nor approach any regulatory limits. The activated and contaminated reactor components requiring disposal as LLRW have a total displacement volume of 280 ft^3 and a total weight of 42,630 lbs. This includes the activated portion of the reactor tank and concrete and all reactor components.

5.0 REFERENCES

1. ORNL CCC-254, ANISN-ORNL One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering, Oak Ridge National Laboratory, April 1991.

2. DLC-185, BUGLE-96, Coupled, 47-Neutron, 20 Gamma-Ray Cross Section Library derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications, Oak Ridge National Laboratory Radiation Shielding Information Center, July 1999.

3. General Atomics Report "Calculated Fluxes and Cross Sections for TRIGA Reactors", GA-4361, August 1963.

4. TRIGA Reactor Description Report, University of Arizona TRIGA Mark I Research Reactor, revised January 1977.

5. ORNL/CCC-371, ORIGEN 2.2, Isotope Generation and Depletion Code, Matrix Exponential Method, Oak Ridge National laboratory, May 2002.

6. Evans, J.C., et al., NUREG 3474, Long-Lived Activation Products in Reactor Materials, Pacific Northwest Laboratories, August 1984.

7. NUREG/BR-0204, Instructions for Completing NRC's Uniform Low-Level Radioactive Waste Manifest, July 1998.

8. University of Arizona Flux Test Report, "Measurement of Absolute Thermal Neutron Flux at the Pool Perimeter of the University of Arizona Research Reactor," dated Dec 1, 2008.

APPENDIX A
ANISN Number Densities

ANISN Atom Densities (a/b-cm)			
Nuclide / Element	Core Region #1	Core Region #2	Avg (Axial Model)
U-235	1.4592E-04	1.4835E-04	1.4672E-04
U-236	1.8913E-07	1.9228E-07	1.9017E-07
U-238	5.8011E-04	5.8978E-04	5.8330E-04
Np-237	4.1548E-10	4.2240E-10	4.1776E-10
Pu-239	5.7657E-08	5.8618E-08	5.7974E-08
Pu-240	2.5812E-10	2.6242E-10	2.5953E-10
Pu-241	2.1666E-12	2.2027E-12	2.1785E-12
H	5.6563E-02	5.9616E-02	5.7570E-02
Boron	0.0000E+00	---	0.0000E+00
B-10	0.0000E+00	---	0.0000E+00
B-11	0.0000E+00	---	0.0000E+00
C (Graphite)	0.0000E+00	0.0000E+00	0.0000E+00
O-16	1.1254E-02	1.2494E-02	1.1663E-02
Al-27	5.7217E-04	---	3.8354E-04
Chromium	8.8513E-04	6.1560E-04	7.9627E-04
Cr-50	3.8503E-05	2.6779E-05	3.4638E-05
Cr-52	7.4165E-04	5.1581E-04	6.6720E-04
Cr-53	8.4088E-05	5.8482E-05	7.5646E-05
Cr-54	2.0889E-05	1.4528E-05	1.8792E-05
Mn-55	7.0049E-05	4.8447E-05	6.2927E-05
Iron	3.1383E-03	2.1827E-03	2.8233E-03
Fe-54	1.8359E-04	1.2769E-04	1.6516E-04
Fe-56	2.8794E-03	2.0026E-03	2.5903E-03
Fe-57	6.6532E-05	4.6274E-05	5.9853E-05
Fe-58	8.7872E-06	6.1116E-06	7.9051E-06
Nickel	4.2587E-04	2.9639E-04	3.8318E-04
Ni-58	2.8993E-04	2.0178E-04	2.6087E-04
Ni-60	1.1166E-04	7.7713E-05	1.0047E-04
Ni-61	4.8549E-06	3.3788E-06	4.3683E-06
Ni-62	1.5459E-05	1.0759E-05	1.3910E-05
Ni-64	3.9606E-06	2.7564E-06	3.5636E-06
Zr	2.0631E-02	2.0963E-02	2.0741E-02

ANISN Atom Densities (a/b-cm)					
Carbon Steel Regions (a/b-cm)		**Aluminum 6061 Regions (a/b-cm)**		**304 Stainless Steel Regions (a/b-cm)**	
Chromium	1.5495E-04	Mg	6.6899E-04	Chromium	1.7112E-02
Cr-50	6.7405E-06	Al	5.8168E-02	Cr-50	7.4439E-04
Cr-52	1.2984E-04	Si	3.4736E-04	Cr-52	1.4339E-02
Cr-53	1.4721E-05	Chromium	6.0979E-05	Cr-53	1.6257E-03
Cr-54	3.6569E-06	Cr-50	2.6526E-06	Cr-54	4.0385E-04
Mn-55	8.7994E-04	Cr-52	5.1094E-05	Mn-55	1.3467E-03
Iron	8.3170E-02	Cr-53	5.7930E-06	Iron	6.1135E-02
Fe-54	4.8654E-03	Cr-54	1.4391E-06	Fe-54	3.5764E-03
Fe-56	7.6309E-02	Mn-55	4.4395E-05	Fe-56	5.6091E-02
Fe-57	1.7632E-03	Iron	2.0381E-04	Fe-57	1.2961E-03
Fe-58	2.3288E-04	Fe-54	1.1923E-05	Fe-58	1.7118E-04
Nickel	5.3294E-04	Fe-56	1.8700E-04	Nickel	8.2390E-03
Ni-58	3.6283E-04	Fe-57	4.3208E-06	Ni-58	5.6091E-03
Ni-60	1.3974E-04	Fe-58	5.7067E-07	Ni-60	2.1603E-03
Ni-62	1.9346E-05	**Standard Concrete Regions (a/b-cm)**		Ni-62	2.9908E-04
Boron Carbide Regions (a/b-cm)		H	8.5647E-03	**Graphite Regions (a/b-cm)**	
Boron	1.0696E-01	Al	1.6260E-03	C	8.5386E-02
B-10	2.1285E-02	Si	8.4654E-03	**Water Regions (a/b-cm)**	
B-11	8.5675E-02	Ca	6.4619E-03	H	6.6693E-02
C	2.8191E-02	Iron	9.8832E-04	O	3.3346E-02
O-16	1.6196E-04	Fe-54	5.7817E-05	**Aluminum 1100 Regions (a/b-cm)**	
Iron	2.1654E-04	Fe-56	9.0679E-04	Al	6.0214E-02
Fe-54	1.2667E-05	Fe-57	2.0952E-05		
Fe-56	1.9867E-04	Fe-58	2.7673E-06		
Fe-57	4.5906E-06				
Fe-58	6.0630E-07				

APPENDIX B
ORIGEN Material Compositions

304 Stainless Steel (NUREG 3474)

- Density = 8.03 $\frac{g}{cm^3}$, 501 $\frac{lbs}{ft^3}$

Average NUREG 3474 304 Stainless Steel Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
Li	30000	1.30E-07	Sr	380000	2.00E-07
N	70000	4.52E-04	Y	390000	5.00E-06
Na	110000	9.70E-06	Zr	400000	1.00E-05
Al	130000	1.00E-04	Nb	410000	8.90E-05
Cl	170000	7.00E-05	Mo	420000	2.60E-03
K	190000	3.00E-06	Ag	470000	2.00E-06
Ca	200000	1.90E-05	Sb	510000	1.23E-05
Sc	210000	3.00E-08	Cs	550000	3.00E-07
Ti	220000	6.00E-04	Ba	560000	5.00E-04
V	230000	4.56E-04	La	570000	2.00E-07
Cr	240000	1.84E-01	Ce	580000	3.71E-04
Mn	250000	1.53E-02	Sm	620000	1.00E-07
Fe	260000	7.06E-01	Eu	630000	2.00E-08
Co	270000	1.41E-03	Tb	650000	4.70E-07
Ni	280000	1.00E-01	Dy	660000	1.00E-06
Cu	290000	3.08E-03	Ho	670000	1.00E-06
Zn	300000	4.57E-04	Yb	700000	2.00E-06
Ga	310000	1.29E-04	Lu	710000	8.00E-07
As	330000	1.94E-04	Hf	720000	2.00E-06
Se	340000	3.50E-05	W	740000	1.86E-04
Br	350000	2.00E-06	Pb	820000	6.70E-05
Rb	370000	1.00E-05	Th	900000	1.00E-06
			U	920000	2.00E-06

Carbon Steel (NUREG 3474)

- Density = $7.87 \frac{g}{cm^3}$, $491 \frac{lbs}{ft^3}$

Average NUREG 3474 Carbon Steel Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
Li	30000	3.00E-07	Sr	380000	1.50E-07
N	70000	8.40E-05	Y	390000	2.00E-05
C*	80000	2.90E-03	Zr	400000	1.00E-05
Na	110000	2.30E-05	Nb	410000	1.88E-05
Al	130000	3.30E-04	Mo	420000	5.60E-07
Cl	170000	4.00E-05	Ag	470000	2.00E-06
K	190000	1.20E-05	Sb	510000	1.10E-05
Ca	200000	1.40E-05	Cs	550000	2.00E-07
Sc	210000	2.60E-07	Ba	560000	2.73E-04
Ti	220000	2.00E-06	La	570000	1.00E-07
V	230000	8.00E-05	Ce	580000	1.00E-06
Cr	240000	1.70E-03	Sm	620000	1.70E-08
Mn	250000	1.02E-02	Eu	630000	3.10E-08
Fe	260000	9.80E-01	Tb	650000	4.50E-07
Co	270000	1.22E-04	Ho	670000	8.00E-07
Ni	280000	6.60E-03	Yb	700000	1.00E-06
Cu	290000	1.27E-03	Lu	710000	2.00E-07
Zn	300000	1.00E-04	Hf	720000	2.10E-07
Ga	310000	8.00E-05	Ta	730000	1.30E-07
As	330000	5.32E-04	W	740000	5.50E-06
Se	340000	7.00E-07	Pb	820000	8.20E-04
Br	350000	8.50E-07	Th	900000	1.80E-07
Rb	370000	4.80E-05	U	920000	2.00E-07

* The carbon weight percent is taken from Page 2, Grade 65 [Grade 450], for typical pressure vessel steel because there is no carbon listed in NUREG 3474.

Bioshield Concrete (NUREG 3474)

- Density = $2.30 \dfrac{g}{cm^3}$, $144 \dfrac{lbs}{ft^3}$

Average NUREG 3474 Bioshield Concrete Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
H	10000	6.10E-3	Rb	370000	3.50E-5
Li	30000	2.00E-5	Sr	380000	4.38E-4
B	50000	2.00E-5	Y	390000	1.82E-5
N	70000	1.20E-4	Zr	400000	7.10E-5
O*	80000	5.02E-1	Nb	410000	4.30E-6
Na	110000	7.39E-3	Mo	420000	1.03E-5
Mg*	120000	2.46E-3	Pd	460000	3.00E-6
Al	130000	3.10E-2	Ag	470000	2.00E-7
Si	140000	1.68E-1	Cd	480000	3.00E-7
P	150000	5.00E-3	Sn	500000	7.00E-6
S	16000	3.10E-3	Sb	510000	1.80E-6
Cl	170000	4.50E-5	Cs	550000	1.30E-6
K	190000	7.50E-3	Ba	560000	9.50E-4
Ca	200000	1.83E-1	La	570000	1.30E-5
Sc	210000	6.50E-6	Ce	580000	2.43E-5
Ti	220000	2.12E-3	Sm	620000	2.00E-6
V	230000	1.03E-4	Eu	630000	5.55E-7
Cr	240000	1.09E-4	Tb	650000	4.10E-7
Mn	250000	3.77E-4	Dy	660000	2.30E-6
Fe	260000	3.90E-2	Ho	670000	9.00E-7
Co	270000	9.80E-6	Yb	700000	1.40E-6
Ni	280000	3.80E-5	Lu	710000	2.70E-7
Cu	290000	2.50E-5	Hf	720000	2.20E-6
Zn	300000	7.50E-5	Ta	730000	4.40E-7
Ga	310000	8.80E-6	W	740000	1.40E-6
As	330000	7.90E-6	Pb	820000	6.10E-5
Se	340000	9.20E-7	Th	900000	3.50E-6
Br	350000	2.40E-6	U	920000	2.70E-5

* The oxygen and magnesium weight percents are calculated since these nuclides are excluded from NUREG 3474.

Aluminum 6061

- Density = $2.7 \dfrac{g}{cm^3}$, $169 \dfrac{lbs}{ft^3}$

Average Aluminum 6061 Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
B	50000	8.43E-7	Fe	260000	7.00E-3
N	70000	1.37E-4	Co	270000	1.72E-4
Mg	120000	1.00E-2	Ni	280000	4.04E-4
Al	130000	9.66E-1	Cu	290000	2.75E-3
Si	140000	6.00E-3	Zn	300000	2.50E-3
Ti	220000	1.50E-3	Nb	410000	3.40E-5
Cr	240000	1.95E-3	Mo	420000	5.60E-7
Mn	250000	1.50E-3			

Graphite

- Density = $1.7 \dfrac{g}{cm^3}$, $106 \dfrac{lbs}{ft^3}$

Average Graphite Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
H	10000	3.89E-5	Mg	120000	1.00E-4
B	50000	5.00E-7	Cl	170000	2.50E-5
C	60000	9.89E-1	Ti	220000	1.00E-3
O	80000	1.29E-3	Fe	260000	8.00E-3
F	90000	2.50E-5	Co	270000	5.00E-5
Na	110000	1.00E-4	Cu	290000	1.00E-4

Stellite

- Density = $8.06 \dfrac{g}{cm^3}$, $503 \dfrac{lbs}{ft^3}$

Average Stellite Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
Cr	240000	3.30E-1	Co	270000	4.23E-1
Mn	250000	1.00E-2	Ni	280000	3.00E-2
Fe	260000	3.00E-2	Nb	410000	5.00E-4