Material Sample Collection with Tritium and Gamma Analyses at the University of Illinois's Nuclear Research Laboratory TRIGA Nuclear Research Reactor

G. Charters, Dr. S. Aggarwal
New Millennium Nuclear Technologies
575 Union Blvd, Suite 102, Lakewood, CO 80228
USA

ABSTRACT

The University of Illinois in Champaign-Urbana has an Advanced TRIGA reactor facility which was built in 1960 and operated until August 1998. The facility was shutdown for a variety of reasons, primarily due to a lack of usage by the host institution. In 1998 the reactor went into SAFSTOR and finally shipped its fuel in 2004. At the present time a site characterization and decommissioning plan are in process and hope to be submitted to the NRC in early 2006.

The facility had to be fully characterized and part of this characterization involved the collection and analysis of samples. This included various solid media such as, concrete, graphite, metals, and sub-slab surface soils for immediate analysis of Activation and Tritium contamination well below the easily measured surfaces. This detailed facility investigation provided a case to eliminate historical unknowns, increasing the confidence for the segregation and packaging of high specific activity Low Level Radwaste (LLRW), from which a strategy of "surgical-demolition" and segregation could be derived thus maximizing the volumes of "clean material."

Performing quantitative volumetric concrete or metal radioanalyses safer and faster (without lab intervention) was a key objective of this dynamic characterization approach. Currently, concrete core bores are shipped to certified laboratories where the concrete residue is run through a battery of tests to determine the contaminants. The existing core boring operation volatilizes or washes out some of the contaminants (like tritium) and oftentimes cross-contaminates the area around the core bore site. The volatilization of the contaminants can lead to airborne problems in the immediate vicinity of the core bore. Cross-contamination can increase the contamination area and thereby increase the amount of waste generated that needs to be treated and stabilized before disposal. The goal was to avoid those field activities that could cause this type of release.

Therefore, TRUPRO®, a sampling and profiling tool in conjunction with radiometric instrumentation was utilized to produce contamination profiles through the material being studied. All samples (except metals) on-site were analyzed within 10 minutes for tritium using a calibrated portable liquid scintillation counter (LSC) and analyzed for gamma activation products using a calibrated ISOCS. Improved sample collection with near real time analysis along with more historical hazard analysis enhanced significantly over the baseline coring approach the understanding of the depth distribution of contaminants. The water used in traditional coring can result in a radioactive liquid waste that needs to be dealt with. This would have been an issue at University of Illinois. Considerable time, risk reduction and money are saved using this profiling approach.

INTRODUCTION

The objective of this Facilities Project was to deploy a concrete sampling and profiling tool to a 1500 KW TRIGA Mark II reactor Primary Bioshield and associated Containment Unit. The facilities undergoing D&D are radiologically activated and contaminated. The concrete samples produced were of a high quality and ready for analysis in on-site instruments eliminating the usual time-consuming sample preparation. As the facility characterization objectives became more defined by the immediate results produced by the system and approach the drilling methods were flexible and adapted to the projects ongoing diverse characterization needs. This enhanced the ability to minimize, eliminate historical process uncertainties and risks for the development of a detailed site Decommissioning Plan. The Decommissioning Plan is used for future demolition work, waste size reduction and packaging into manageable pieces for safe radwaste handling and disposal so meeting Waste Acceptance Criteria.

There was a need to detect Fission Products, Tritium & Activation Products, Cs-137, H-3, Co-60, Eu-152, & Eu-154 at depth within the TRIGA reactor containment concrete matrix. This was to support the development of a Facility Decommissioning Plan and the evaluation of waste volumes, packaging and shipment goals and to allow increased worker safety awareness. The accepted method for detecting H-3 is Liquid Scintillation Counting (LSC) and Cs-137, Eu-152, Eu-154 and Co-60 is gamma spectrometry.

Profiling the concretes and associated reactor special materials of the TRIGA structure to depth included the following desired capabilities and design features of the equipment: (a) powered by a facility electric supply coupled to 200 foot power cables (b) removal of concrete, metal and soil samples, using a specialized tungsten carbide drilling technology (c) remove incremental depths of potentially contaminated concrete from predetermined sampling points sequentially to depth varying from one inch, six inch and one foot sample increments to 7 feet (1 – 84 inch) in any direction (d) collection of all concrete particulate from sampling operations into specialized filter units and (e) produce representative samples in a powdered form for optimal counting by liquid scintillation counting and gamma spectrometry.

The analytical results indicate that there was measurable gamma and tritium beta activity in most of the locations and depths sampled i.e. the Reactor Primary Bioshield, Containment Unit Floor, and the subterranean walls floors, ceiling and subsurface soils. The Bioshield Wall and floor had a historical process leak and surface surveys had revealed elevated activity levels in these areas. Sampling to depth incrementally into predetermined sampling points on walls, floors and trenches in these elevated locations showed volumetrically how considerable contamination had penetrated through the easily measured surface material. Considerably large volumes of concrete in the reactor Bioshield Wall annulus had substantially higher specific activation activities than the containment building concrete matrix values.

The sampling and radionuclide contamination data for concrete surfaces, walls and slab subsurface soil matrices, adequately characterized the extent and amount of the Containment Building's concrete structures levels of contamination that exist. Adequate samples and data were produced to determine the extent of contaminants existing in the concrete slab sub-surfaces and Primary Bioshield in quantities that might be dispersed to the air and working environment

on slab and concrete removal. Despite highly confined space for sample location access, sampling and profiling to depths below slab surfaces and deep into the Primary Bioshield to 87 inches to retrieve concrete and metal samples at depth at this site was relatively straightforward and was used in the acquisition of incremental samples to measure levels of contamination.

Technical Field

In preparation for this TRIGA reactor sampling and characterization work, historical information regarding the operation of the facility and detailed blueprints and diagrams of the facility and process components were evaluated. This information was used to optimize the efficacy of sampling and characterization efforts. Determining the appropriate axis of sampling into the bioshield beyond shadow shielding plates alongside and parallel to neutron beam ports was paramount to successfully sampling to the required depth and show the limits of activation through the concrete matrix.

By upfront strategizing of sampling approach and accessibility into unknown volumes of potentially contaminated material, numbers of samples, consumable sampling equipment, secondary wastes and sampling efforts were kept to a minimum but maximized the usefulness of the data set of sample activities produced to enhance the detail and certainty in developing a Decommissioning Plan. Rather than performing random concrete core samples, Health Physics performed field surveys in order to help identify any additional locations needing further radiological analyses. This sampling technology increased the probability of detecting possible remaining contamination at depth and the collected powder samples were representative for selected concrete volumes. Professional judgment based on technical knowledge and professional experiences of the site processes were used as part of the scientific investigation where knowledge was incomplete. Known locations of particular interest were the Reactor Annulus (Primary Bioshield Walls) and the Containment Unit floor working area due to the radiation dose readings of those wall and floor structures. From historical knowledge and field surveys, a set of candidate sampling locations was developed for the sampling process. The predetermined designated sampling location was marked off on the concrete surfaces. This helped enhance the sampling effort by better targeting the contamination of concern. Collected samples (powdered concrete and metal shavings) were representative of the radiological conditions of the Containment concrete matrices.

Over 88 separate samples were obtained. These included powdered concrete and metal fine shaving samples with 12 soil samples at depth from the TRIGA reactor containment facility subsurface soil matrix. Included in this sampling activity were associated site evaluations, and day to day sampling plans and documented reporting. Sampling was in both the horizontal direction (i.e., through walls) and in the vertical directions (downward into the floors and up into ceilings of the reactor lower bioshield concrete shielding). Some sampling was at angles from a perpendicular orientation to a wall or floor surface to obtain samples from inaccessible depths within structures. Drill guides and mounting equipment were used to achieve these sampling angles; equipment provided included a stable and versatile lightweight drill mount gantry platform to ensure safe and deep sampling operations throughout the containment structure. All of the retrieved powdered samples were initially assessed semi quantitatively by hand held probe and measured quantitatively for specific radionuclide activities of interest, specifically tritium,

fission products and activation products.

The portable sampling system was lifted by the facility gantry crane into difficult and inaccessible areas of the reactor containment building and was wheeled easily over the containment unit floor to access sample locations. A facility electrical supply, 100 feet of power cable, connecting plastic hoses, vacuum drill, and vacuum pump with a multiple filter manifold system were maneuvered around the facility despite the tight inaccessible conditions. To access the subterranean tunnel for sampling through the concrete floors and walls of the 48 inch wide and high tunnel and up into the concrete ceiling directly underneath the TRIGA nuclear reactor core, 35 feet of vacuum hose and electrical cable were laid out into these highly restrictive and confined areas with the vacuum pump and manifold remaining above ground at the access tunnel entrance. 9 locations and 12 samples were retrieved from these cramped and limited spaces.

Methodology

Incremental concrete and metal samples were obtained for information on the extent of the neutron activation in the reactor's bioshield wall and the distribution of tritium contamination throughout the containment concrete structure. A series of vertically aligned sampling holes were performed with the intent to ultimately develop an activation product distribution representation of the activated concrete and metal reactor tank liner. Sampling in close proximity to the reactor tank liner was thought to be to risky given the tank still contained water so sampling radially inward toward the reactor core was achieved twelve times into areas of highest potential neutron activation thus developing a worst case scenario for the model of the degree of material activation and tritium distribution. It is known that this activated and tritium contaminated material would be required to be handled and packaged separately from the bulk of the Containment Unit "clean" concrete due to the expected higher dose rates. The desire was to better determine activation and tritium migration profiles to maximize amounts of non- activated from contaminated concrete, to remove only what really needs to be taken out separately, as each additional amount of concrete removed from the annulus increases the packaging, shipping and disposal costs.

The slab surface holes and subsurface soil sampling holes were drilled to determine how deep the surface tritium contamination had migrated into the concrete matrices. Generally, these areas were sampled by drilling and collecting the material from three 1-inch increments to provide enough sample mass for gamma spectrometry and on through the bulk of the slab to retrieve the next 3 to 6 inches of subsurface soils. Thirteen concrete samples were taken from the floor of the reactor room. Ten of the samples were taken at points on the surface survey grid and three samples were taken from the thermal column shield door trench. One additional concrete sample was taken in the southwestern corner of the Mechanical Equipment Room.

The profiling equipment sequentially collected all material from the holes to minimize waste generation. The bulk material samples were retrieved by use of a specially designed, two-stage vacuum sample retrieval unit that prevents cross contamination of the clean retrieved samples. No circulation medium is required with this profiling process; therefore, the only by-product from drilling is the sample. Filter units were changed out every sample to prevent cross contamination of the samples and drill bits and connecting tubing were changed out where high

contamination levels were observed. The samples are acquired in a dry powder form, at accurately predetermined incremental depths to provide confidence in representativeness of sample point locations. The drill mounted gantry system and sampling retrieval equipment was used to sample at depth as realistically as possible given the sampling point suitability and data value. Deep holes into the concrete matrices had some special challenges. Sometimes the drill bits as they sampled through the concrete bioshield came in close proximity to the metal tank liner so a specialized electric drill cutoff mechanism was employed to ensure liner integrity and that no piercing of the liner occurred. Some locations were improperly marked out prior to drilling due to historically inaccurate facility blue prints causing the occasional striking of imbedded shadow beam port metal objects, and prevented drilling deeper. The samples were placed into individually labeled and pre weighed sample pots and the drill bit and connecting tubing cleaned by gently knocking whilst under vacuum to purge contamination and changed out to keep the system free of possible contamination.

The sampling technology collected samples of graphite blocks from the thermal column. The sampling approach prevented airborne contamination by the use of 0.01 micron filters but to minimize further the potential for airborne contaminants, the graphite samples were collected in a sample collection box that was connected to a high-efficiency particulate air (HEPA) filter vacuum. For the collection of metal samples an electric drill with a specialized hollow metal cutting bit was used under vacuum inside the sample collection box, the vacuum line was also used to collect the shavings onto the filter and then transferred to a sample container.

Drilling and sampling were performed without the use of water and without the use of fugitive dust minimization techniques. A half inch thick metal plate cradle allowed the positioning of the sampling system to allow safe access and anchorage of drilling mechanism using longer drill bit lengths. Samples were placed into 125cc clear glass containers with Teflon lid liners. Each layer of medium encountered was sampled and bulked individually.

All drilling observations were logged along with the sample locations, the sampling methodologies, the resulting sample mass data, tritium and speciated gamma data and description of sample depth and composition.

Bioshield Wall

Various sampling locations were selected from the Bioshield Wall based on the ability to maximize the value of data produced with minimum effort to determine the boundary of increased activity levels within larger volumes of concrete. The annulus was sampled to depth to observe any possible leaks or activation product activity levels in the structure. Concrete and metal samples from the beam port metal shadow shields and associated cavity metal liner were obtained to develop an activity profile of the concrete matrix. This enabled to segregate activated concrete from non activated material with certainty

To analyze the potential for neutron activation in the concrete directly below the TRIGA core location (#33) was drilled vertically upward with accesses by the subterranean service tunnel. The hole reached a sampling depth of 15", thus sampling through the reactor cavity subsurface large volumes of concrete in the worst activation scenario into very inaccessible sampling

locations.

Containment Unit Slab Floor

Various sampling locations were selected from the Containment Unit working floor area based on dose rate readings. The samples from 3 –16 and 36 & 37 (excluding 1 & 2 as background samples) were floor and subsurface soil locations. Leaks and historical flooding in North East, East and South East quadrants of the facility had contaminated the upper 1 inch of the concrete matrix and on into the subsurface soils.

Tunnel Sampling

Four samples of the concrete tunnel floor were taken and analyzed for tritium and activation products. Four concrete samples were also taken of the tunnel ceiling/reactor bottom from the tunnel at four different elevations to an elevation of 15 inches above the tunnel ceiling, about halfway to the reactor tank. Three soil samples were collected from under the reactor through the tunnel walls and four soil samples were collected from below the tunnel floor.

Sample Analysis

The accepted method for detecting H-3 is Liquid Scintillation Counting (LSC) and Cs-137, Eu-152, Eu-154 and Co-60 is gamma spectrometry. For concrete, graphite, metal, and some soil samples, gamma spectroscopy was used to identify and quantify the levels of gamma-emitting radionuclides in the samples using an on-site high-purity germanium (HPGe) detector. The concrete, graphite, and soil samples collected were also analyzed for tritium and total beta activity using a Beta Scout portable LSC.

RESULTS

Table 1 summarizes the samples collected and analyzed for the reactor annulus and containment floors and associated structures along with special metal materials retrieved from the reactor pool showing sampling point locations and depths respectively. The tritium and gamma results are also tabulated. The Minimum Detectable Activity (MDA) for each isotope is also shown in the table. The MDA for the gamma results varied for each sample due to the sample size and total activity of the sample.

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Table I. Tritium Activation Product Activities (pCi/g) With Depth (Inches) University of Illinois TRIGA Reactor

Sample No.	Location	Media	Depth (Inches)	H3 (pCi/g)	MDA H3 (pCi/g)	Gamma Isotope	Activity (pCi/g)	MDA
NMNT-01	Bkgd Bioshield	Concrete	3.0			K-40	<MDA	38.6
						Co-60	3.5	3.3
NMNT-02	Bkgd Floor	Concrete	3.0			K-40	<MDA	30.4
						Co-60	<MDA	2.3
NMNT-03	Floor Loc. 3	Concrete	1.0	20300.0	272.6	K-40	<MDA	38.0
						Co-60	<MDA	3.3
						Eu-152	<MDA	4.2
NMNT-04	Floor Loc. 3	Soil	15.0	2461.8	140.8	K-40	<MDA	20.9
NMNT-05	Floor Loc. 4	Concrete	1.0	17129.8	272.6	K-40	<MDA	34.9
						Co-60	<MDA	3.0
						Cs-137	<MDA	1.5
NMNT-06	Floor Loc. 5	Concrete	1.0	11348.3	141.2	K-40	<MDA	4.0
						Co-60	<MDA	3.2
						Eu-154	4.9	4.3
NMNT-07	Floor Loc. 6	Concrete	1.0	19540.7	272.6	K-40	<MDA	44.5
						Co-60	<MDA	4.0
NMNT-08	Floor Loc. 7	Concrete	1.0	13870.2	141.2	K-40	<MDA	37.9
NMNT-09	Floor Loc. 8	Concrete	1.0	14188.3	272.6	K-40	<MDA	44.6
NMNT-10	Floor Loc. 8	Soil	15.0	6406.7	140.8	K-40	<MDA	34.6
						Co-60	<MDA	2.8
NMNT-11	Floor Loc. 9	Concrete	1.0	30833.7	272.6	K-40	<MDA	34.3
						Co-60	<MDA	2.8

Sample No.	Location	Media	Depth (Inches)	H3 (pCi/g)	MDA H3 (pCi/g)	Gamma Isotope	Activity (pCi/g)	MDA
NMNT-17	Floor Loc. 14	Concrete		24108.6	272.6	K-40	<MDA	48.5
						Co-60	<MDA	4.0
						Cs-137	<MDA	1.8
NMNT-18	Floor Loc. 15	Concrete	1.0	11408.4	141.2	K-40	<MDA	43.0
NMNT-19	Floor Loc. 16	Concrete	1.0	2791.5	272.6	K-40	<MDA	58.3
						Co-60	<MDA	4.7
						Ra-226+D	3.3	3.1
NMNT-20	Bioshield Loc. 17	Concrete	9.0	59905.9	152.3	K-40	489.0	139.0
						Co-60	8390.0	60.7
						Eu-152	9010.0	87.9
						Eu-154	578.0	80.1
NMNT-21	Metal from tank wall	Metal (Al)	1.0	NA	NA	K-40	<MDA	1040.0
						Co-60	8990.0	169.0
						Eu-152	4730.0	274.0
NMNT-21A	Filter resin	Resin		< MDA	297.1	K-40	<MDA	3.4
						Co-60	48.0	0.5
NMNT-22	Bioshield Loc. 18-1	Concrete	39.0	< MDA	297.1	K-40	<MDA	34.6
						Co-60	30.7	3.4
						Eu-152	33.1	5.1
NMNT-23	Bioshield Loc. 18-2	Concrete	51.0	360.3	152.3	K-40	<MDA	31.7
NMNT-24	Bioshield Loc. 18-3	Concrete	63.0	< MDA	297.1	Co-60	<MDA	2.8
						Ra-226+D	<MDA	1.8

NMNT-12	Floor Loc. 10	Concrete	1.0	13936.2	141.2	K-40	<MDA	43.7
NMNT-13	Floor Loc. 11	Concrete	1.0	16610.7	272.6	K-40	<MDA	44.6
						Co-60	<MDA	3.7
NMNT-14	Floor Loc. 12	Concrete	1.0	21333.7	141.2	K-40	<MDA	46.7
NMNT-15	Floor Loc. 12	Soil	15.0	8305.4	297.7	K-40	<MDA	17.2
NMNT-16	Floor Loc. 13	Concrete	1.0	13690.0	141.2	K-40	<MDA	45.7
						Co-60	<MDA	3.7

NMNT-25	Bioshield Loc. 19-1	Concrete	63.0	< MDA	152.3	K-40	<MDA	13.5
						Co-60	<MDA	1.2
NMNT-26	Bioshield Loc. 19-2	Concrete	68.0	10039.4	141.2	K-40	<MDA	14.4
						Co-60	15.1	1.4
						Eu-152	17.9	2.2
NMNT-27	Bioshield Loc. 20	Concrete		46330.4	145.7	K-41	<MDA	36.3
				51169.7	131.5	Co-61	<MDA	33.3

Table II. Gamma Activities (pCi/g) With Depth (Inches) Old Cave Entombment Sumps & Trenches

Sample No.	Location	Media	Depth (Inches)	H3 (pCi/g)	MDA H3 (pCi/g)	Gamma Isotope	Activity (pCi/g)	MDA
NMNT-27	Bioshield Loc. 20-	Concrete				Cs-137	<MDA	0.8
NMNT-28	Bioshield Loc. 20-2	Concrete	15.0	<MDA	152.3	Co-60	8.1	2.7
						Eu-152	7.1	2.9
NMNT-29	Bioshield Loc. 20-3	Concrete	27.0	3345.2	297.1	K-40	<MDA	40.8
						Co-60	23.1	3.9
						Eu-152	31.3	5.8
NMNT-30	Bioshield Loc. 20-4	Concrete	33.0	7331.4	152.3	K-40	<MDA	38.8
						Co-60	66.7	4.4
						Eu-152	59.1	7.2
NMNT-31	Bioshield Loc. 20-5	Concrete	36.0	5202.4	297.1	Co-60	80.1	3.1
						Eu-152	79.0	6.7
NMNT-32	Bioshield Loc. 21-1	Concrete	3.0	5349.9	152.3	K-40	<MDA	21.7
						Co-60	12.6	2.0
						Eu-152	9.4	2.6
NMNT-33	Bioshield Loc. 21-2	Concrete	15.0	< MDA	297.1	K-40	<MDA	20.3
						Co-60	12.2	1.8

Sample No.	Location	Media	Depth (Inches)	H3 (pCi/g)	MDA H3 (pCi/g)	Gamma Isotope	Activity (pCi/g)	MDA
NMNT-37	BST Floor Loc. 22	Concrete	1.0			Cs-137	<MDA	4.3
						Eu-152	81.5	10.9
						Eu-154	10.6	7.2
NMNT-38	BST Floor Loc. 23	Concrete	1.0	8190.0	141.2	K-40	<MDA	48.5
						Co-60	83.3	5.5
						Cs-137	<MDA	2.7
						Eu-152	8.9	6.1
NMNT-39	BST Floor Loc. 24	Concrete	1.0	18144.9	272.6	K-40	<MDA	49.6
						Co-60	<MDA	4.1
NMNT-40	Tunnel Loc. 25	Concrete	1.0	8057.9	141.2	K-40	<MDA	48.5
						Co-60	23.6	4.7
						Eu-152	18.7	5.6
NMNT-41	Tunnel Loc. 25	Soil	15.0	9174.9	297.7	Co-60	4.9	0.0
NMNT-42	Tunnel Loc. 26	Concrete	1.0	8784.4	141.2	K-40	<MDA	44.8
						Co-60	<MDA	3.4
NMNT-43	Tunnel Loc. 27	Soil	15.0	4187.3	297.7	Co-60	13.9	0.0

						Cs-137	<MDA	1.1
						Eu-152	10.2	2.5
NMNT-34	Bioshield Loc. 21-3	Concrete	27.0	1321.0	152.3	K-40	<MDA	33.7
						Co-60	58.9	3.6
						Eu-152	52.6	6.5
NMNT-35	Bioshield Loc. 21-4	Concrete	33.0	2664.6	297.1	Co-60	291.0	6.4
						Eu-152	262.0	12.3
NMNT-36	Bioshield Loc. 21-5	Concrete	36.0	1783.3	152.3	Co-60	533.0	6.3
						Eu-152	534.0	15.6
						Eu-154	32.6	13.5
NMNT-37	BST Floor Loc. 22	Concrete	1.0	11927.4	272.6	K-40	<MDA	43.4
						Co-60	254.0	5.0

NMNT-44	Tunnel Loc. 28	Concrete	1.0	6208.6	141.2	K-40	<MDA	36.1
NMNT-45	Tunnel Loc. 29	Soil	15.0	< MDA	297.7	K-40	<MDA	65.1
						Co-60	<MDA	5.1
NMNT-46	Tunnel Loc. 30	Soil	15.0	7105.7	297.7	K-40	<MDA	44.7
NMNT-47	Tunnel Loc. 31	Concrete	1.0	6869.0	141.2	K-40	<MDA	31.1
NMNT-48	Tunnel Loc. 31	Soil	15.0	1649.5	297.7	Co-60	<MDA	0.0
NMNT-49	Tunnel Loc. 32	Soil	15.0	<MDA	297.7	K-40	<MDA	59.8
						Co-60	<MDA	5.2
NMNT-50	Tunnel Ceiling 33-1	Concrete	4.0	10531.7	297.1	K-40	<MDA	30.4
						Co-60	<MDA	2.5
NMNT-51	Tunnel Ceiling 33-2	Concrete	8.0	53763.4	152.3	Co-60	<MDA	2.6
NMNT-52	Tunnel Ceiling 33-3	Concrete	12.0	27107.8	297.1	K-40	<MDA	27.7

Table II. Gamma Activities (pCi/g) With Depth (Inches) Old Cave Entombment Sumps & Trenches

Sample No.	Location	Media	Depth (Inches)	H3 (pCi/g)	MDA H3 (pCi/g)	Gamma Isotope	Activity (pCi/g)	MDA
NMNT-53	Tunnel Ceiling 33-4	Concrete	15.0	15851.6	152.3	K-40	<MDA	29.8
NMNT-54	Bioshield Loc. 34-1	Concrete	15.0	< MDA	297.1	K-40	<MDA	22.4
						Co-60	<MDA	2.0
NMNT-55	Bioshield Loc. 34-2	Concrete	27.0	< MDA	152.3	K-40	<MDA	23.2
NMNT-56	Bioshield Loc. 34-3	Concrete	9.0	< MDA	297.1	Co-60	<MDA	2.8
NMNT-57	Bioshield Loc. 34-4	Concrete	51.0	< MDA	152.3	K-40	<MDA	30.0
NMNT-58	Bioshield Loc. 34-5	Concrete	63.0	< MDA	297.1	K-40	<MDA	26.8
						Co-60	<MDA	2.2

Sample No.	Location	Media	Depth (Inches)	H3 (pCi/g)	MDA H3 (pCi/g)	Gamma Isotope	Activity (pCi/g)	MDA
NMNT-66	Bioshield Loc. 35-6	Concrete	69.0			Eu-152	206.0	9.7
						Eu-154	12.9	7.4
NMNT-67	Bioshield Loc. 35-7	Concrete	81.0	< MDA	152.3	Co-60	3550.0	28.9
						Eu-152	3890.0	50.3
						Eu-154	229.0	41.8
NMNT-68	Floor Loc. 36 (Bay)	Concrete	1.0	16749.1	272.2	K-40	<MDA	34.1
						Co-60	82.6	3.3
						Eu-152	88.2	6.9

NMNT-59	Bioshield Loc. 34-6	Concrete	75.0	< MDA	297.1	K-40	<MDA	32.4
						Co-60	6.4	3.0
						Eu-152	4.4	3.7
NMNT-60	Bioshield Loc. 34-7	Concrete	87.0	< MDA	152.3	K-40	<MDA	45.1
						Co-60	388.0	6.6
						Eu-152	476.0	15.5
						Eu-154	26.9	10.2
NMNT-61	Bioshield Loc. 35-1	Concrete	9.0	< MDA	297.1	K-40	<MDA	21.4
						Co-60	<MDA	1.7
NMNT-62	Bioshield Loc. 35-2	Concrete	21.0	< MDA	152.3	K-40	<MDA	41.9
						Co-60	55.5	4.7
						Eu-152	60.2	7.3
NMNT-63	Bioshield Loc. 35-3	Concrete	33.0	< MDA	297.1	K-40	<MDA	19.8
						Co-60	<MDA	1.7
NMNT-64	Bioshield Loc. 35-4	Concrete	45.0	< MDA	297.1	K-40	<MDA	35.6
						Co-60	<MDA	2.8
NMNT-65	Bioshield Loc. 35-5	Concrete	57.0	< MDA	152.3	K-40	<MDA	45.7
						Co-60	17.5	3.9
						Eu-152	10.9	4.9
NMNT-66	Bioshield Loc. 35-6	Concrete	69.0	< MDA	297.1	Co-60	232.0	4.5

NMNT-69	Floor Loc. 37 (Bay)	Concrete	1.0	9577.0	141.2	K-40	<MDA	38.2
						Co-60	<MDA	3.4
NMNT-70	Therm. Col. Loc. 38	Concrete	3.0	< MDA	297.1	K-40	<MDA	32.0
						Co-60	203.0	4.6
						Eu-152	241.0	9.4
						Eu-154	23.7	6.8
NMNT-71	Bioshield Loc. 39	Concrete	30.0	< MDA	297.1	K-40	<MDA	47.5
	36 Inches Above Reactor Center Line					Co-60	265.0	6.4
						Eu-152	181.0	11.3
						Eu-154	14.8	10.0
NMNT-72	Bioshield Loc. 40	Concrete	30.0	< MDA	153.0	K-40	<MDA	31.0
	30 Inches Above Reactor Center Line					Co-60	26.8	3.0
						Eu-152	15.9	4.4
NMNT-73	Bioshield Loc. 41	Concrete	30.0	< MDA	297.1	K-40	<MDA	18.0
	24 Inches Above Reactor Center Line					Co-60	47.4	1.9
						Eu-152	38.7	4.1
NMNT-74	Bioshield Loc. 42	Concrete	30.0	6406.7	153.0	K-40	<MDA	29.8
	18 Inches Above Reactor Center Line					Co-60	6.0	2.5
						Eu-152	5.2	3.0
NMNT-75	Graphite 1	Graphite		< MDA	297.1	K-40	<MDA	47.4

Table II. Gamma Activities (pCi/g) With Depth (Inches) Old Cave Entombment Sumps & Trenches

Sample No.	Location	Media	Depth (Inches)	H3 (pCi/g)	MDA H3 (pCi/g)	Gamma Isotope	Activity (pCi/g)	MDA
NMNT-75	Graphite 1	Graphite				Co-60	119.0	5.6
						Eu-152	109.0	7.8
NMNT-76	Graphite 2	Graphite		< MDA	153.0	K-40	<MDA	33.5
						Co-60	4.9	2.8
						Eu-152	14.6	3.8
NMNT-77	Graphite 3	Graphite		8882.1	297.1	Co-60	35.6	8.0
						Eu-152	5430.0	49.7
						Eu-154	335.0	35.9
NMNT-78	Graphite 4	Graphite		23249.1	153.0	Co-60	105.0	17.7
						Eu-152	17200.0	111.0
						Eu-154	1200.0	81.6
NMNT-79	Grid Plate Center	Metal (Al)		NA	NA	Co-60	110000.0	628.0
						Eu-152	6350.0	387.0
NMNT-80	Grid Plate Edge	Metal (Al)		NA	NA	Co-60	121000.0	703.0
						Cs-137	<MDA	213.0
NMNT-81	Grid Plate Bolt	Metal (SS)		NA	NA	Co-60	6.0E+07	3.1E+05
NMNT-82	Pipe @ 16"	Metal (Al)		NA	NA	K-40	<MDA	681.0
						Co-60	2640.0	73.0
NMNT-83	Pipe @ 24"	Metal (Al)		NA	NA	Co-60	65.3	16.2
						Eu-152	96.2	22.3
NMNT-84	Pipe @ 32"	Metal (Al)		NA	NA	K-40	<MDA	329.0
						Co-60	120.0	30.6
						Eu-152	65.1	34.4
NMNT-	Pipe @	Metal		NA	NA	Co-60	99.1	31.0

Sample No.	Location	Media	Depth (Inches)	H3 (pCi/g)	MDA H3 (pCi/g)	Gamma Isotope	Activity (pCi/g)	MDA
NMNT-87	Trench loc. 43-1	Soil				Co-60	9.6	6.6
NMNT-88	Trench loc. 43-2	Soil		< MDA	297.7	K-40	<MDA	11.6
						Co-60	1.1	1.0
Boral	Boral Curtain	Metal (B)				K-40	<MDA	535.0
						Co-60	14800.0	152.0
						Eu-152	514.0	191.0
Bkg Soil	Background soil	Soil				K-40	<MDA	17.1
						Co-60	1.6	1.5
Unknown 1	Off-site sand	Soil				K-40	2.5	2.4
						Co-60	0.3	0.3
Unknown 2	Subsurface soil	Soil				K-40	<MDA	5.9
						Co-60	0.8	0.5

85	8"	(Al)						
						Eu-152	6850.0	95.9
						Eu-154	444.0	85.3
NMNT-86	Floor Loc. 5	Soil		4821.7	297.7	K-40	<MDA	16.9
NMNT-87	Trench loc. 43-1	Soil		8247.7	297.7	K-40	<MDA	78.2

Based on the samples evaluated, Fig. 1. to Fig. 4. show the following:

1. There was high gamma activation within the adjacent 12 inches of reactor concrete annulus at reactor center line and was also present 48 inches above the reactor center line.
2. The tritium and total beta count rate from concrete floor and trench samples demonstrate that the concrete is contaminated with tritium in varying concentrations. All fourteen samples had tritium levels greater than the MDA. While tritium levels in the reactor water were always within discharge limits during reactor operation, it is suspected that the tritium contamination resulted from years of buildup from small leaks in the reactor and Bulk Shielding Tank.
3. The concrete samples from the tunnel floor showed no tritium above the MDA and total beta activity at two to three times the background count rate. The samples from the tunnel ceiling/reactor bottom (NMNT-50 through NMNT-53), however, had much higher levels of tritium and total beta activity than the floor. The estimated total beta activity for the four ceiling samples ranged from 0.54 to 2.9 uCi/g. All four samples also had tritium concentrations above the MDA. The maximum tritium concentration is an estimated 0.056 uCi/g.
4. The results indicate the concrete is contaminated to a depth of about 36 inches (91.44 cm) from the reactor tank at the reactor centerline. However, the "radius of activation" is not expected to be symmetrical due to the various internal components in the bioshield such as the beam tubes, thermal columns, and shadow shields.
5. From the derived vertical profile obtained by taking concrete samples near the reactor tank wall at four different elevations separated by 6 inches the first 1.5 feet up from the core centerline on the tank centerline from the Bulk Shielding Tank wall the concrete is proven to be activated to a depth of 6 inches at an elevation of at least 3 feet from the core centerline.

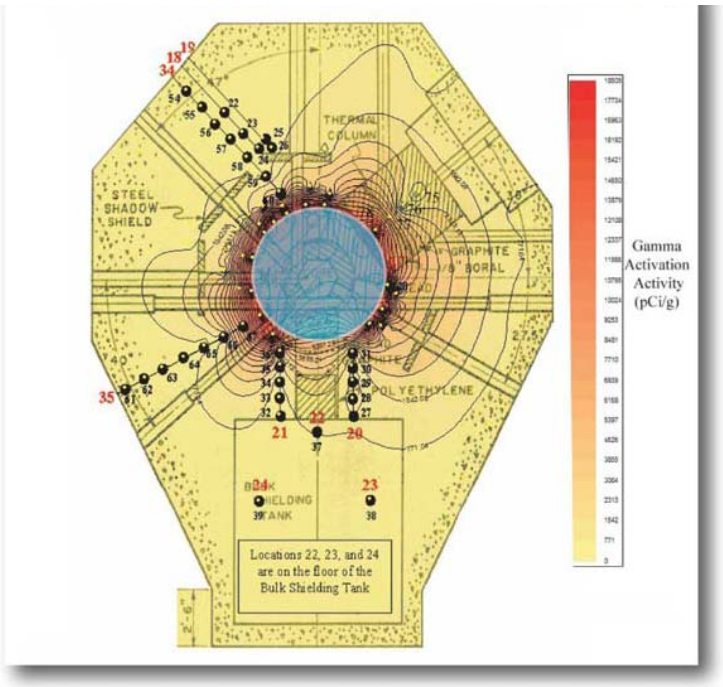

Fig. 1. TRIGA Annulus total gamma activation activity distribution (pCi/g) cross section with reactor centre line

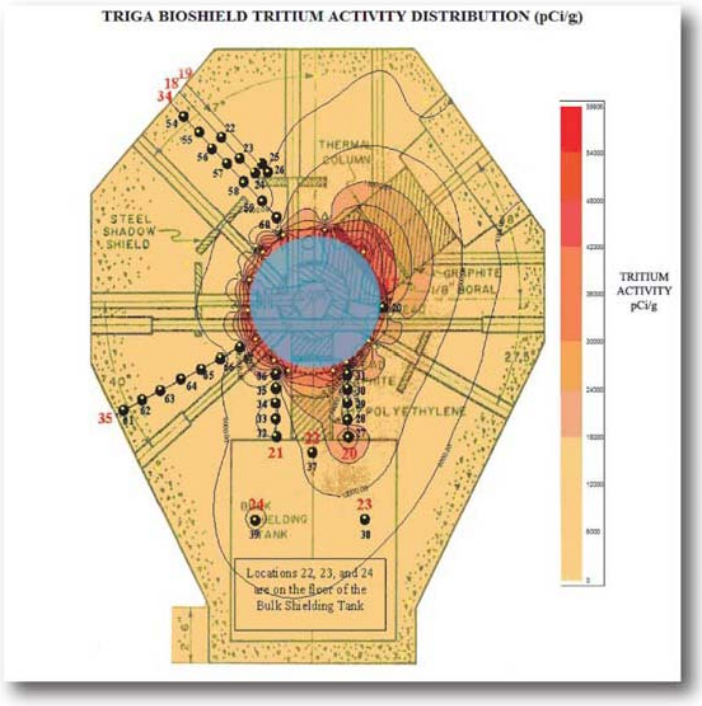

Fig. 2. TRIGA Annulus total Tritium activity distribution (pCi/g) cross section with reactor centre line

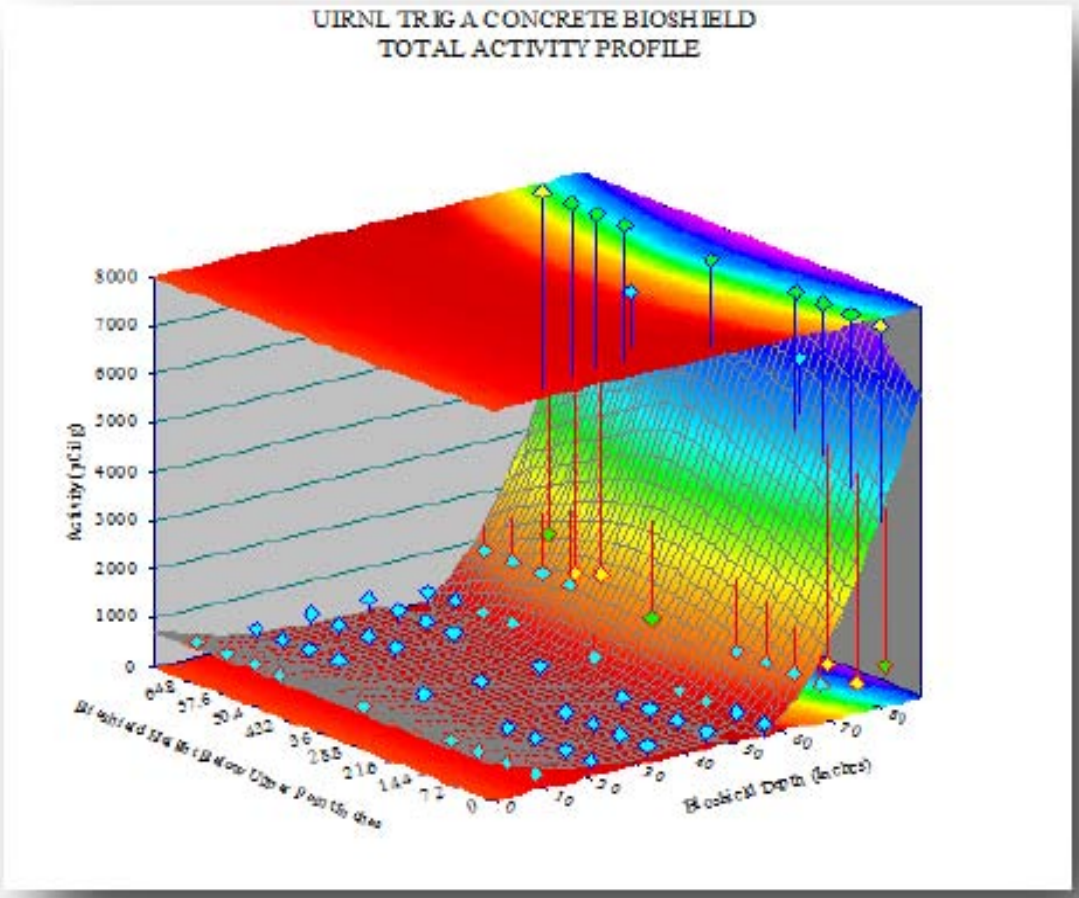

Fig. 3. Cross section activity profile of UIRNL TRIGA Bioshield to 82-inches depth

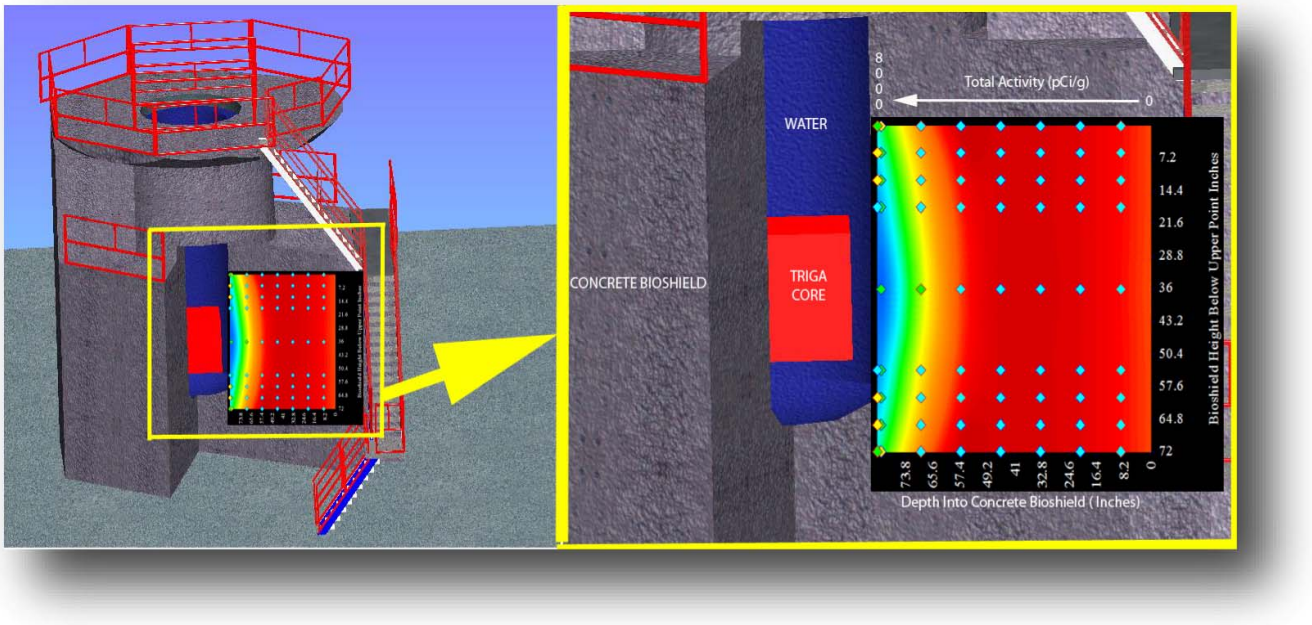

Fig. 4. TRIGA Reactor Bioshield activity profile distribution (pCi/g)

CONCLUSIONS AND RECOMMENDATIONS

A summation of radionuclide activation activity was performed by extrapolating concentrations over all the TRIGA reactor structural materials. This resulted in greater resolution of bounding the maximum degree of tritium migration and the activation activity profile so maximizing confidence and estimations of volumes of clean material with a significant reduction in project time and secondary wastes produced. Rather than sampling by coring and subsampling the cores this approach tightened error margins of waste volumes and radionuclide inventories for the development of a Decommissioning Plan and thus safer more effective operational demolition of the TRIGA reactor facility will be achieved under less tightly controlled, contained, and monitored conditions, much less restrictive demolition becomes feasible.

- Bulk samples were taken of the concrete from 1 inch to 2 feet at three locations. Due to the mobility of tritium in porous media such as concrete, it is expected that the concrete floor will contain extensive volumetric contamination.
- The gamma spectroscopy results indicate that the floor is not contaminated with activation products. In all floor samples, gamma-emitting nuclides of concern were less than their respective MDAs.
- The concrete sampling and profiling technology produced high quality, representative samples, so that the radioanalysis was reliable.
- Confidence is greatly increased in radiological safety and in D&D operations by eliminating previously undeterminable historical unknowns.

Core bore sampling would have meant not accessing the contaminated areas of the structure,

inherently causing inaccuracies in building material activity quantification and waste volume removal estimates. Anomalies of activity levels at the various critical locations of the documented historical leaks throughout the structure were unearthed so decreasing project turn-around time, and reducing radiological risk. The downtime associated with these delays reduces productivity, increases costs, and most importantly, delays worker awareness of the area during demolition. Historical and physical radiological unknowns are eliminated using the sampling and characterization approach so increasing confidence in decision making abilities.

Most importantly the concrete sampling and profiling technology in conjunction with the infield portable quantitative radiometric instrumentation, significantly improved technical awareness of the activation and contamination of subsurface materials. This incuded underlying concrete structures and soils up to 1 foot under the lower concrete slab foundation indicating an ingress of tritium activity from historical reactor leaks.