

## **A Multi-Phased Sampling Effort to Characterize a University TRIGA Research Reactor**

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### **ABSTRACT**

A radiological characterization project was conducted at the University of Illinois (University) TRIGA research nuclear reactor in July 2005 as part of the long-term facility decommissioning project. The characterization effort included multiple survey and sampling techniques designed to assess both contamination of the reactor building and equipment and activation of reactor components and the reactor bioshield. Radiation measurements included alpha and beta surface contamination measurements, gamma dose rate measurements, and gross gamma radiation measurements. Modeling was conducted based on the field measurements to predict concentrations of activation products in reactor components that were not directly sampled. The sampling effort included collecting removable contamination swipes, concrete samples from the reactor room floor and bioshield, soil samples from below and around the perimeter of the reactor building, graphite samples from graphite moderator, and metal samples from reactor components. Concrete samples were obtained using an innovative technology that allowed for quick sample collection and analysis. Concrete, soil, graphite, and metal samples were analyzed on-site using liquid scintillation counters and gamma spectroscopy. Additional samples were sent off-site for analysis.

### **INTRODUCTION**

Construction began on the Nuclear Research Laboratory (NRL) at the University's Urbana-Champaign campus in the summer of 1959. A TRIGA Mark II training and research nuclear reactor manufactured by the General Atomic Division of General Dynamics Corporation was installed in the NRL. The installation was completed in the summer of 1960 and the reactor first went critical on August 16, 1960.

In the early years, the reactor operated at with a maximum power rating of 100 kilowatts (kW) using fuel elements with a zirconium hydride moderator homogeneously combined with 20% enriched uranium. The fuel was arranged in a circular lattice in the core that was positioned at the bottom of the reactor tank under approximately 16 feet of water. A 1-foot-thick radial graphite reflector surrounded the core. By 1967, upgrades and license amendments allowed for the operating limit to be increased to 250 kW.

In 1967, the University decided to upgrade the reactor to utilize the most recent design characteristics of the TRIGA fuel and to install a new forced circulation cooling system. The original core was also replaced with a new core that was also light-water-cooled, graphite reflected, and contained uranium-zirconium hydride fuel-moderator elements with stainless steel cladding. The fuel elements in the new core, however, were positioned in a hexagonal lattice. The new cooling system enabled a steady-state thermal power level of up to 3,000 kW (3.0 megawatts); however, the NRC license limited the power to 1,500 kW (thermal). Fig. 1 and Fig. 2 provide sketches of the reactor configuration following the 1967 upgrades.

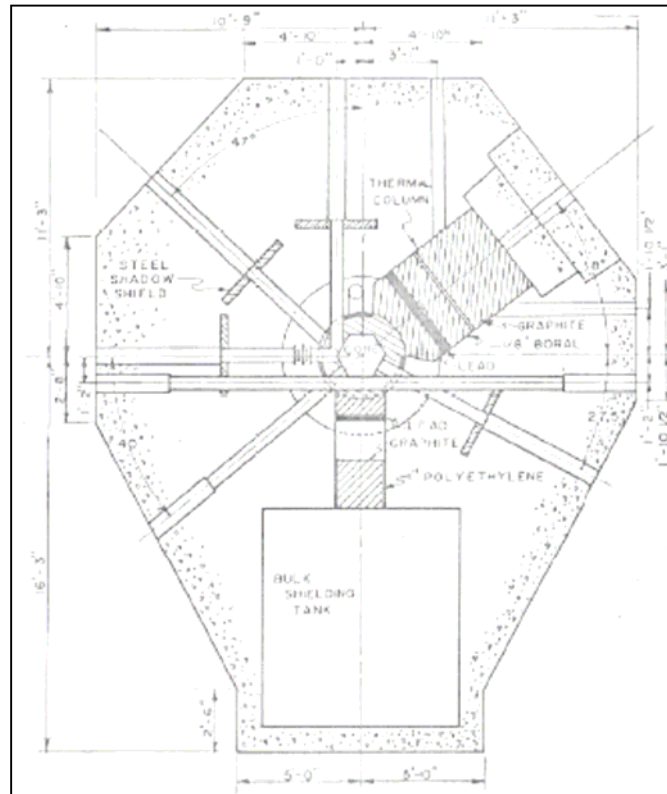


Fig. 1. Plan view of the reactor (Ref. 1)

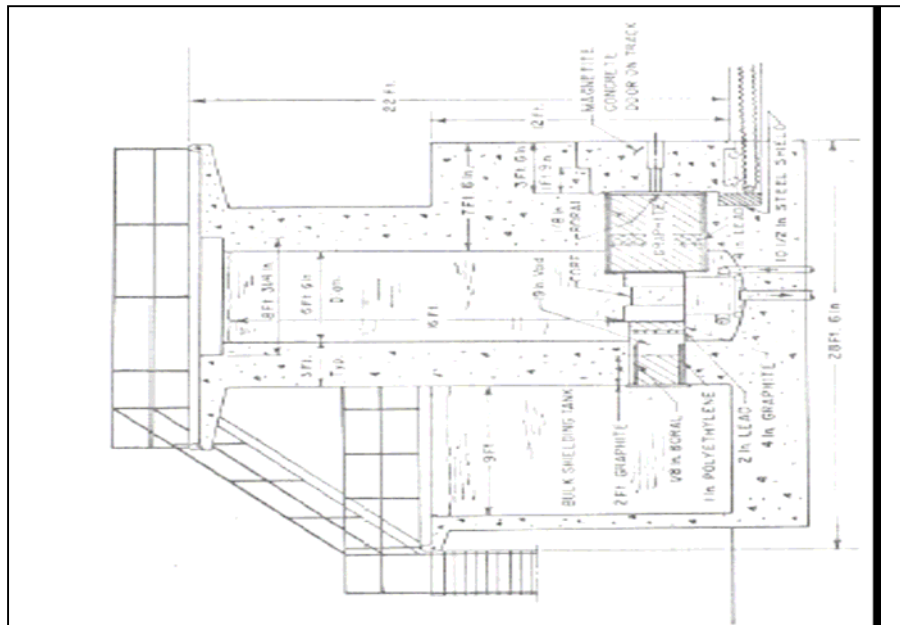


Fig. 2. Sectional view of the reactor (Ref. 1)

The primary use of the reactor was to support experiments conducted by University staff or outside organizations. Many of these experiments involved the irradiation of materials inside the reactor core with the use of a pneumatic transfer system for fast insertion and removal of specimens, a rotary specimen assembly (lazy Susan), the central thimble, or fuel positions. Other experiments utilized a neutron beam produced from the core through either one of nine beam ports or the thermal column, which utilizes graphite as a neutron moderating material.

On August 6, 1998, nearly 30 years since its initial start-up and more than 11,500 mega-watt-hours (MWhrs) of operation, the NRL TRIGA reactor was shut down permanently. In 1999, the reactor was officially placed in a SAFSTOR condition while waiting for arrangements to be made to remove and ship the reactor fuel. The Bulk Shielding Tank was used for wet storage of the fuel following shutdown. On August 18, 2004, the reactor fuel was removed and shipped to the U.S. Department of Energy's Idaho National Laboratory and the decommissioning process began in the summer of 2005 with the site characterization described in the following sections of this paper.

## CONTAMINATION POTENTIAL

The Historical Site Assessment (HSA) identified various radioactive isotopes at the NRL that may have impacted the facility (Ref. 2). These isotopes are primarily activation products and tritium. Activation products will appear in many of the materials used in and around the NRL reactor. Potential activation products include C-14 in graphite; Fe-55, Co-60, and Ni-63 in aluminum and stainless steel; and Fe-55, Co-60, Cs-134, and Eu-152 in the heavy concrete.

Tritium is known to have been present in waste water collected and discharged to the sanitary sewer system. While discharges met all appropriate concentration limits, the accumulation of tritium in concrete may present a contamination problem. The HSA identified primary coolant water leaks that occurred in 1971, 1974, 1976, and 1978, from either the reactor tank or the

primary coolant system (Ref. 2). Impacted areas include the reactor room floor, including the thermal column door trench, and the decay tank vault and sump. There were also leaks reported from the Bulk Shielding Tank

Principal long-lived (half-life greater than 1 year) fission product isotopes that are present in a reactor fuel following shut down include Sr-90, Cs-134, and Cs-137. While there was only one recorded fuel leak of reactor fuel in 1966, there were fuel leakages from the Low-Power Reactor Assembly that was formerly located in the Bulk Shielding Tank. As such, these fission products may be present as contaminants on the interior surface of the Bulk Shielding Tank and other surfaces that were in contact with reactor coolant water (reactor tank, N-16 decay tanks, heat exchanger, and concrete areas impacted by primary coolant water leaks).

Fission products were also released during a laser experiment that utilized uranium-235 coated tubes (Ref. 2). During the experiment, fission products were released into the reactor room. While gaseous releases through the NRL stack were within regulatory limits, there is a possibility that heavier isotopes were deposited on surfaces within the NRL facility, including xenon-137 which decays to cesium-137.

The HSA identified that there was also a potential for tritium to have impacted the soils under the NRL facility (Ref. 2). While no contamination was confirmed prior to the site characterization, the potential exists because there were known leaks of the primary coolant water onto concrete surface.

## **ACTIVATION POTENTIAL**

Many of the primary reactor components were constructed of an aluminum alloy. Neutron activation of aluminum may result in formation of several radioactive isotopes including Al-28, Na-24, and Mg-27 through a number of neutron capture processes. Fortunately, each of these isotopes has a very short half-life of less than 15 hours and they are of no concern. However, the aluminum alloy contains impurities in the form of other elements that can become activated by the reactor neutron flux and form radioactive isotopes. Co-60 (half-life = 5.3 years) and Fe-55 (half-life = 2.7 years) are the primary isotopes of concern in activated aluminum. Other activation products may include Ni-63 (half-life = 100 years), zinc-65 (half-life = 244 days), Mn-54 (half-life = 313 days), and Eu-152 (half-life = 13.6 years).

The heavy concrete of the bioshield is also a concern for activation, especially since it contains magnetite (iron) ores. The primary activation products are Co-60 and Fe-55 from the magnetite ore and Eu-152 from the concrete. Cs-134 (half-life = 2.1 years) and Eu-154 (half-life = 8.8 years) may also be present at lower levels.

Stainless steel and carbon steel are also activated when exposed to neutrons. The primary activation products in reactor components constructed of these materials will be Co-60, Fe-55, Ni-63, and Mn-54 (Ref. 3).

Graphite will also become activated and contaminated from surrounding activated materials. Potential isotopes present in activated graphite include C-14, Co-60, Eu-152, and Eu-154 (Ref. 3).

Neutron interactions with lead do not present a significant source of long-lived activation products. The lead bricks used in the thermal column are reportedly 99.85% pure lead (Ref. 2).

## CHARATERIZATION TOOLBOX

The overall objective of the characterization project was to identify sources of known and unknown contamination and activation such that demolition waste volumes and waste classifications could be estimated to support the development of the Decommissioning Plan. To accomplish this goal, however, it was not necessary to quantify all potential radionuclides of concern in all potentially contaminated or activated materials. In many instances, all that was necessary to provide sufficient information to support the decommissioning planning was to identify materials as contaminated or activated using total activity values (such as total gamma count rates or total beta activity) without fully characterizing the material.

To meet the project objectives, the characterization effort included multiple survey and sampling techniques. Radiation measurements included alpha and beta surface contamination measurements, gamma dose rate measurements, and gross gamma radiation measurements. Modeling was conducted based on the field measurements to predict concentrations of activation products in reactor components that were not directly sampled. The sampling effort included collecting removable contamination swipes, concrete samples from the reactor room floor and bioshield, soil samples from below and around the perimeter of the reactor building, graphite samples from graphite moderator, and metal samples from reactor components. Concrete samples were obtained using an innovative technology that allowed for quick and efficient sample collection and analysis. Concrete, soil, graphite, and metal samples were analyzed on-site using liquid scintillation counters (LSC) and gamma spectroscopy. Additional samples were sent off-site for analysis. Table I summarizes the characterization toolbox.

Table I. Characterization Toolbox

<b>Characterization Activity</b>	<b>Characterization Tools</b>
Floor, wall, and equipment contamination surveys	Direct measurements and scans Smear counting – liquid scintillation counter Smear counting – proportional counter
Concrete sampling for activation and contamination	Innovative sample collection tool Portable liquid scintillation counter Gamma spectroscopy Off-site radiochemistry
Graphite and reactor component sampling and surveys for activation	Innovative sample collection tool Portable liquid scintillation counter Gamma spectroscopy Dose rate meter (ion chamber) Underwater detector housing Dose modeling software
Soil sampling	Multiple sample collection tools Portable liquid scintillation counter Gamma spectroscopy Off-site radiochemistry
Inaccessible equipment and tank contamination surveys	Gamma radiation measurements Gamma spectroscopy
Water and mineral deposit sampling	Liquid scintillation counting

## **Surface Contamination Measurements**

To assess the general radiological condition of the NRL facility, surface contamination measurements were made throughout the facility using properly calibrated instruments capable of detecting both total alpha and total beta surface activity. To assess removable contamination, smear samples were taken over a 100cm<sup>2</sup> surface area and analyzed for tritium, C-14, and total beta activity or for total alpha and total beta activity.

Direct total alpha and total beta measurements were made at line intersections of a systematic survey grid. Ludlum 4360 alpha/beta phoswich detectors were used for these measurements. The minimum detectable activity for the direct instruments varied slightly with background, but they were generally about 430 dpm/100cm<sup>2</sup> beta and 35 dpm/100cm<sup>2</sup> alpha. Several beta activity data points were rejected because of the high background radiation level at the survey location. Easily accessible horizontal surfaces such as the tops of cabinets and pipes were surveyed and were considered representative of harder to reach areas such as the roof support beams.

Scanning measurements were performed on floors and walls of the NRL facility where radiation levels permitted. 100% of the accessible floor areas were scanned using a Ludlum floor monitor equipped with a gas flow proportional detector. Parts of the reactor level floor could not be scanned due to the high background radiation level caused by the materials in the Radioactive Materials Storage Cage. Wall scans were performed with the Ludlum 4360.

## **Dose Rate Measurements and Modeling Software and Gross Gamma Measurements**

For the purpose of measuring high-dose rate fields, an Eberline RO-7 ion chamber was used with multiple probes. An underwater housing was also used to measure dose rates inside the reactor tank. In particular, the underwater detector provided dose rates at various elevations in the center of the reactor. These measured dose rates were used in combination with the MicroShield® Version 6.01 dose modeling software was used to estimate the activity of the reactor's rotary specimen rack which is located around the reactor core. The rotary specimen rack is believed to be the reactor component containing the highest concentration of Co-60 because the drive chain contains Stellite bearings (unactivated Stellite contains as much as 60% stable Co-59). The model predicts that the rotary specimen rack may contain as much as 0.15 TBq (4 Ci) of Co-60.

For surveys of equipment with inaccessible surfaces that were potentially contaminated or activated, 2-inch by 2-inch sodium iodide (2x2 NaI) gamma scintillation detectors were used to measure the total gamma activity. Equipment surveyed included coolant pumps, filters, storage tanks, heat exchanger, and piping. The surveys indicated that there is contamination in bottom of the N-16 decay tanks and possibly in the primary coolant pipes and pump. NaI gamma spectroscopy revealed that Co-60 was the primary gamma-emitting contaminant in the N-16 decay tanks. Surveys on the heat exchanger were inconclusive.

## Sample Collection

Fifty-two concrete samples, twelve soil samples, and four graphite samples were collected at the NRL facility using the TruPro® sampling technology. The TruPro® sampling technology was deployed by New Millennium Nuclear Technologies (NMNT) of Lakewood, Colorado. This technology, which was used in place of traditional core boring, uses a hammer drill with hollow 12-millimeter drill bits to collect concrete powder samples. Drill bits ranged in length from about 30 cm to 2.4 m. While drilling, the drill bit is connected to a vacuum pump that removes all of the sample material from the hole as the drill is advanced. The sample material is collected in a filter and then transferred directly to a plastic sample container. Because the hollow drill bits are connected to a vacuum pump, the drilling process does not generate airborne dust.

For samples of the concrete floor, three 1-inch holes were drilled at each of the sample locations to provide enough sample volume for gamma spectroscopy analysis. For samples collected in the bioshield, the drill was first advanced to the sample point, for example, 12 inches from the outside surface of the bioshield. Next, the vacuum was directed to a separate sample filter and the drill was advanced three inches. The material from these three inches made up the sample material for that sample point. The sample points in the reactor bioshield are shown in Fig. 3.

The TruPro® technology was also used to collect soil samples from just below the concrete floor of the reactor room and concrete and soil samples under the reactor through the primary coolant water pipe tunnel. Soil samples were collected under the reactor by first drilling a hole through the concrete block walls and then advancing the hollow drill bit through the soil. The technology worked well for collecting soil samples just below the floor surface because the soil was very dry. However, for wetter soils, the drill bit tended to clog frequently.

The TruPro® technology was also used to collect samples of graphite blocks from the thermal column. While the TruPro® technology inherently limits airborne contamination, to further minimize 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.

To sample aluminum and stainless steel components, a standard electric drill with a metal cutting bit was used to generate metal shavings inside the sample collection box. A vacuum line was then used to collect the shavings into a sample filter and then the shavings were transferred to a sample container.

Two different GeoProbe® sampling units were used by Direct Push Analytical Corp. of Batavia, Illinois to collect 60 subsurface soil samples from below and around the NRL facility. The GeoProbe® technology uses hydraulically driven rods to advance sample collection sleeve into the soil to a desired depth. To collect the samples, the sample sleeve is brought to the surface and the soil from the desired depth range is removed.

Samples collected outside the NRL building were collected at 4-to-5, 8-to-9, and 12-to-13-foot depths below the ground surface (surface grade). Samples collected inside the reactor room, through the concrete floor were taken at 0-to-1, 3-to-4, 6-to-7, 9-to-10, and 12-to-13-foot depths below the floor. The reactor room floor is about 6 feet below the ground surface and the floor is

2 feet thick. Therefore, the 0-to-1-foot sample from inside the building corresponds to the same depth a 8-to-9 foot sample from outside the building.

A breakdown of the material samples collected is provided in Table II. Table II also provides the analyses that were performed on the samples. The sample analytical methods are described in the following section.

Table II. Sample Collected During the NRL Facility Characterization

Location	Media	Number of Samples	Technology	Analyses <sup>a</sup>
Reactor Room Floor <sup>b</sup>	Concrete	16	TruPro®	On-site LSC Gamma Spec.
	Soil	4	TruPro®	On-site LSC Gamma Spec.
	Soil	19	GeoProbe®	Off-site tritium
Tunnel	Concrete	8	TruPro®	On-site LSC Gamma Spec.
	Soil	8	TruPro®	On-site LSC Gamma Spec.
	Soil	2	GeoProbe®	Off-site tritium
Bioshield	Concrete	30	TruPro®	On-site LSC Gamma Spec.
Bulk Shielding Tank Floor	Concrete	3	TruPro®	On-site LSC Gamma Spec.
Building Perimeter	Soil	39	GeoProbe®	Off-site tritium
Background	Concrete	2	TruPro®	On-site LSC Gamma Spec.
	Soil	2	Manual	On-site LSC Gamma Spec.
	Soil	4	Manual	Off-site tritium
Large Thermal Column	Graphite	4	TruPro®	On-site LSC Gamma Spec.
Top Grid Plate	Aluminum	2	Manual and TruPro®	Gamma Spec.
Top Grid Plate Bolt	Stainless Steel	1	Manual and TruPro®	Gamma Spec.
Emergency Spray Pipe	Aluminum	4	Manual and TruPro®	Gamma Spec.
Reactor Tank Wall	Aluminum	1	TruPro®	Gamma Spec.

<sup>a</sup> Eleven concrete and two soil samples were sent off-site for Fe-55 and Ni-63 analysis.

<sup>b</sup> Includes 1 sample from the Mechanical Equipment Room, 2 samples from the loading bay, and 3 samples from the shielding door trench.



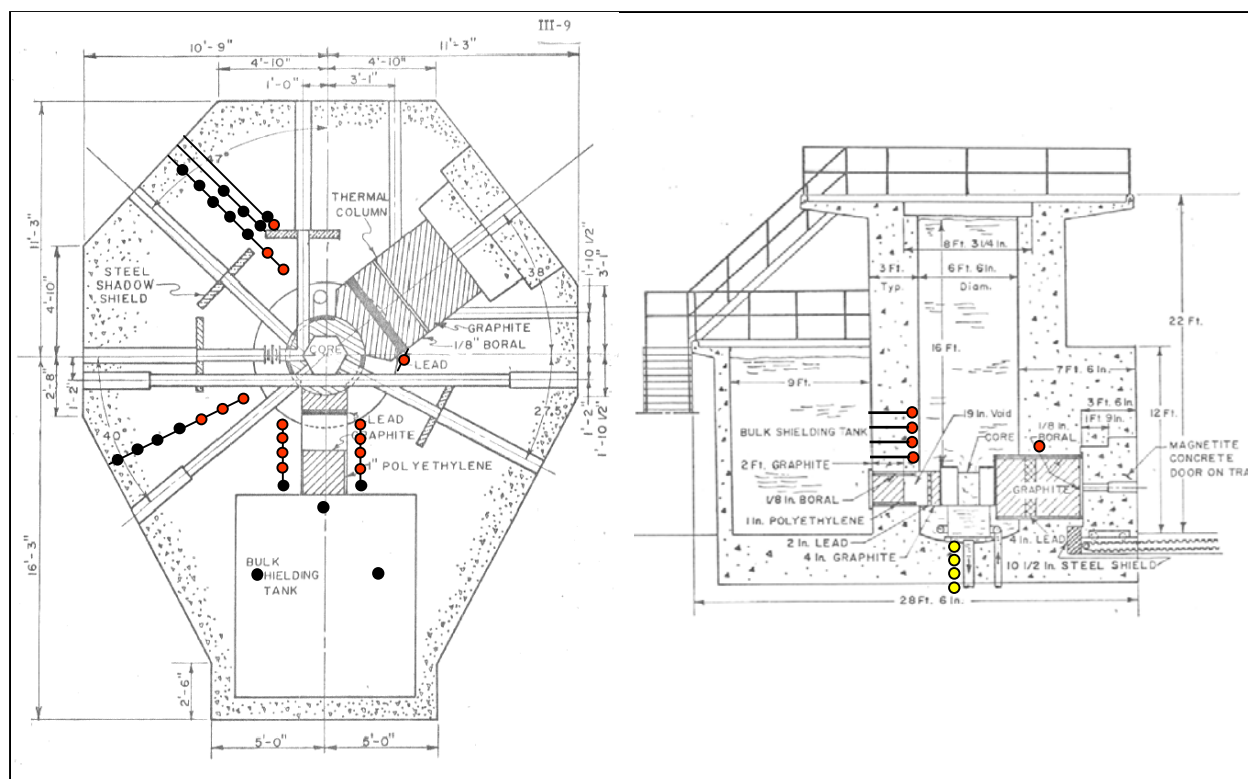


Fig. 3. Bioshield concrete sample locations

## SAMPLE ANALYSIS

Several analytical techniques were used to examine samples for radionuclides of concern. 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 some soil samples collected were also analyzed for tritium and total beta activity using a Beta Scout® portable liquid scintillation counter (LSC). The subsurface soil samples and several concrete samples were sent off site to Severn Trent Laboratories (STL) in Earth City, Missouri for tritium analysis. These soil samples were not analyzed on-site using gamma spectroscopy because activation products were not considered contaminants of concern for the subsurface soils. STL also performed analysis for Fe-55 and Ni-63 on selected concrete samples.

## On-site Liquid Scintillation Counting

Two Beta Scout® portable LSCs were used on-site to analyze samples collected using the TruPro® technology for tritium and total beta activity. The samples were prepared by adding 0.1 grams of sample material (concrete, soil, or graphite) along with 5 milliliters (ml) of InstaGel® scintillation fluor and 5 ml of deionized water to a 20 ml scintillation vial. The samples were then counted for 1 minute and the tritium and total beta count rates were recorded in counts per minute (cpm).

The use of the less efficient portable LSCs combined with a small sample size results in a significantly high minimum detectable activity for tritium on the order of several hundred Bq/g.

However, because one of the primary goals of the characterization was to identify impacted areas, the tritium and total beta count rate data was useful in predicting whether a sample was greater than background in a qualitative assessment.

A Beckman 6500LS laboratory-quality LSC was used in a nearby University facility. This LSC was used to count removable contamination smears, a sample of mineral deposits present along a crack in the exterior of the reactor bioshield, and water (groundwater infiltration) from the vault containing the N-16 decay tanks.

### **On-site Gamma Spectroscopy**

A Canberra Industries (Canberra) HPGe gamma spectroscopy system was used to analyze concrete, graphite, metal, and some soil samples on-site. Samples consisted of a small volume of sample material, typically 15 to 25 grams, in a plastic sample container. The masses of the metals samples were much lower. The samples were placed in a lead sample well to reduce the sample counting background and counted for 10 minutes (some background samples were counted for 20 minutes). Canberra's Genie2000® gamma spectrum and analysis software was used to process and analyze the spectra.

Because the samples were not a standard size and no reference source of the same geometry was available, the Canberra *In-Situ* Object Counting System (ISOCS)® Geometry Composer software was used to model the sample geometries to provide efficiency files. The efficiency files were used by Genie2000® to perform the detector efficiency calibrations. The energy calibrations were performed using a Eu-152 multi-peak gamma radiation standard.

The gamma spectra analysis used a limited radionuclide library that contained naturally-occurring radionuclides, the primary radionuclides of concern (Co-60, Eu-152, and Eu-154), the fission product Cs-137, naturally-occurring K-40 and Ra-226 decay daughters. MDAs for a 20 gram background concrete sample with a 10 minute count time were less than 0.2 Bq/g (~5 pCi/g) for Co-60 and Cs-137 and less than 0.4 Bq/g (~10 pCi/g) for Eu-152.

### **Off-site Radiochemistry Analysis**

Sixty soil samples and eight concrete floor samples were sent to STL for tritium analysis. These samples were analyzed using the U.S. Environmental Protection Agency (EPA) method 906.0 MOD. This method uses distillation of the sample and LSC analysis of the distillate. As part of its quality assurance (QA) assessment, the laboratory performed analyses on method blanks, laboratory control samples, matrix spike samples, and ran duplicate analysis on 3 of the 60 soil samples. The results of the QA tests demonstrate that the data was of an acceptable quality.

Eleven concrete and two soil samples were also sent to STL for Fe-55 and Ni-63 analysis. These isotopes are hard to detect using field methods. STL used U.S. Department of Energy (DOE) method STL-RC-0055 for the analyses. Both analyses involved liquid scintillation spectroscopy. QA analyses included a method blank, a laboratory control sample, two duplicate Fe-55 analyses, and one duplicate Ni-63 analysis. The results of the QA tests showed that the data was of an acceptable quality.

## CHARACTERIZATION RESULTS

### Activated Materials

One of the primary goals of the characterization effort was to define the “radius of activation” within the reactor bioshield. Because of the history of the reactor operating at multiple power levels, the non-symmetrical placement of components within the bioshield such as the bean ports and shadow shields, and the presence of unknown impurities in the bioshield concrete, computer modeling of the activation was believed to be a costly and time consuming effort that would yield questionable results. By collecting concrete samples at various depths and at various locations around the reactor and analyzing them on-site using the portable LSCs and gamma spectroscopy, it was determined that the “radius of activation” was about 0.75 meters from the reactor tank wall. It was also determined by taking concrete samples along the reactor centerline at various elevations that the concrete is activated up to about 1.2 m from the core centerline.

The presence of Co-60 and Eu-152 in the high-density bioshield concrete samples was the primary indicator of activation. Gamma spectroscopy analysis indicated that these two radionuclides are typically present in the concrete at approximately equal concentrations and Eu-154 is typically present at 5% to 15% of the Eu-152 concentration. The maximum measured activity in a concrete sample was taken from just outside the tank wall on the core centerline with Co-60, Eu-152, and Eu-154 activities equal to 311 Bq/g (8,400 pCi/g), 333 Bq/g (9,000 pCi/g), and 21 Bq/g (580 pCi/g) respectively.

The extent of activation of the aluminum reactor tank was determined by removing an aluminum pipe from the tank and collecting samples of the pipe. The pipe was part of the emergency spray system and ran along the side of the tank wall. The samples, which consisted of drill cuttings, were analyzed with the on-site gamma spectroscopy system to identify and quantify Co-60 activity. The pipe sample Co-60 concentrations and an extrapolation of the data predict that the tank may be activated to a height of up to 2 m from the reactor's top grid plate.

Samples were also taken from two points in the top grid plate (made of aluminum), from a stainless steel bolt, and the TruPro® technology was used to collect four samples of graphite from the large thermal column. Each of these samples was analyzed on-site using gamma spectroscopy. The analysis indicated that the top grid plate contained as much as 3,700 Bq/g (0.1 uCi/g) Co-60 and that the stainless steel bolt contained about 2.2 MBq/g (60 uCi/g) Co-60. The on-site analysis also indicated that all of the graphite in the large thermal column was activated and contained Eu-152 in concentrations that ranged from 0.6 Bq/g (0.15 pCi/g) in the sample collected furthest from the reactor core to 630 Bq/g (1,700 pCi/g) in the sample collected near the lead brick wall in the thermal column.

As previously described, the reactor's rotary specimen rack was estimated to contain about 0.15 TBq (4 Ci) of Co-60 based on underwater dose rate measurements and MicroShield® dose modeling. Fig. 4 provides the modeled and measured dose rates.

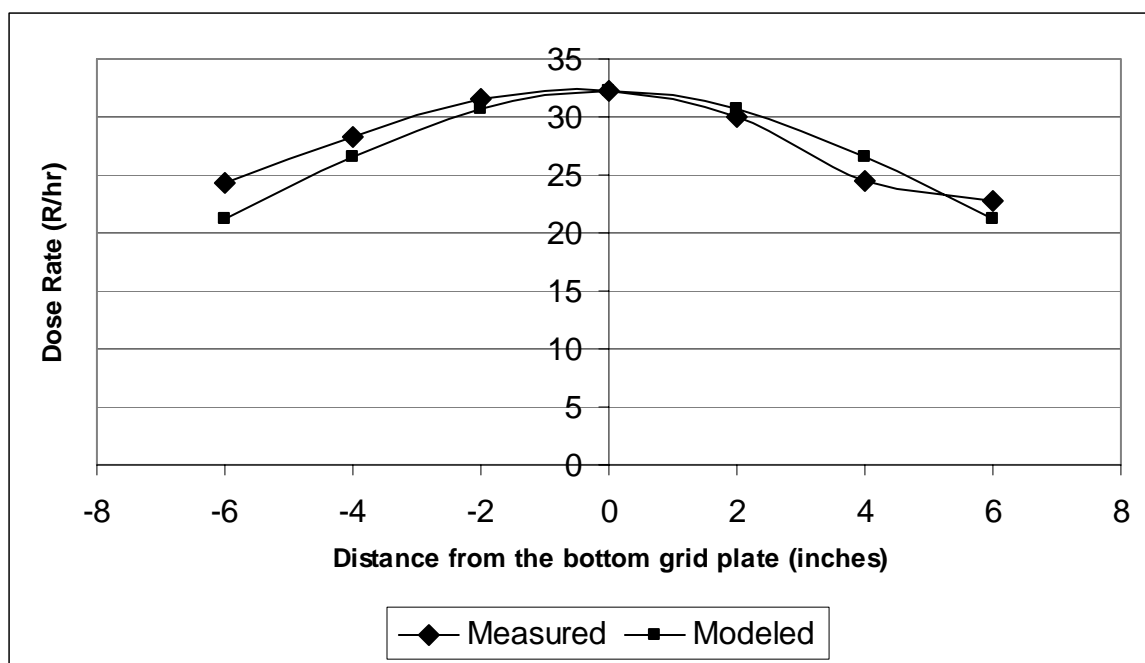


Fig. 4. Modeled and measured dose rates for reactor rotary specimen rack

The concentrations of other activation products were estimated based on the concentration of the measured radionuclides and the expected ratios of the measured radionuclides to those that were not measured. These expected ratios were derived from various sources.

### Contaminated Materials

Contaminated materials were identified in many areas of the NRL facility. Contaminated equipment was identified using direct and removable alpha/beta contamination surveys and gross gamma activity surveys. Removable tritium contamination was identified in the Mechanical Equipment Room and the Loading Bay using LSC analysis of surface smears.

On-site portable LSC analysis of concrete samples for tritium and total beta activity indicated volumetric contamination in the concrete floor of the reactor room and in the pipe chase tunnel under the reactor. Off-site laboratory analysis confirmed that some concrete is volumetrically contaminated with tritium. Off-site laboratory analysis for Fe-55 in several concrete samples also indicated that it is present as a potential concrete contaminant in some locations. The mineral deposits below the crack in the bioshield also indicated the presence of tritium. Analysis of the water samples collected in the N-16 decay tank vault in the University's LSC did not indicate the presence of tritium or other beta-emitting radionuclides.

### CONCLUSION

The University of Illinois TRIGA research reactor was successfully characterized in a short period of time (less than 3 weeks) using a variety of analytical methods and tools. The TruPro® concrete sampling technology allowed for fast sample collection with minimal sample preparation time proving far superior to traditional concrete coring. On-site analytical methods were maximized to help direct additional sample and survey efforts and needs for off-site sample

analyses. The results of the characterization effort allowed for estimates of low-level radioactive waste classifications and volumes. The information derived from the characterization is being used to prepare the facility Decommissioning Plan and decommissioning cost estimates.

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