

## **CHARACTERIZATION OF THE BROOKHAVEN GRAPHITE RESEARCH REACTOR – THE FIRST NUCLEAR REACTOR BUILT FOR PEACEFUL SCIENTIFIC RESEARCH**

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### **ABSTRACT**

This paper describes the historical significance, construction, and the methodology, techniques, and results obtained while characterizing the Brookhaven Graphite Research Reactor (BGRR) at Brookhaven National Laboratory (BNL). The BGRR, similar in design to the X-10 reactor at Oak Ridge National Laboratory, was the world's first nuclear reactor dedicated to the peaceful exploration of atomic energy. A comprehensive review of the facilities history was performed prior to characterization activities to determine the contaminants of concern and potential environmental release pathways. The unique history, design, and construction of the BGRR provided significant challenges in accessing and obtaining meaningful samples for characterization of the reactor pile, underground structures, and soils beneath them.

### **INTRODUCTION**

Brookhaven National Laboratory was one of the first three National Laboratories created to build and operate research facilities that were too large for single universities to finance and operate. The other two national labs, Argonne National Laboratory and Oak Ridge National Laboratory, were constructed during World War II to develop the atomic bomb and were converted to peacetime purposes after the war. Of those three facilities, Brookhaven was the only laboratory constructed as a civilian institution with its core program solely for scientific research [1].

The team selected to design, build, and eventually operate the Brookhaven Graphite Research Reactor was assembled during the summer of 1946 and was led by Lyle Borst. Borst was a senior physicist at the X-10 reactor and he was pressed by Congress to have an operational research reactor facility as quickly as possible. Probably for that reason alone, the BGRR is very similar in design to the X-10 with some key improvements to better its use as a research tool versus plutonium production.

Some of the modifications were minor and were aimed at improving facility safety but others were very significant. One major modification was the design of the biological shield that surrounded the reactor pile. Unlike the X-10 reactor, researchers performing experiments would be positioned near the reactor face over extended periods. The BGRR biological shield was developed from high density concrete; thus providing the additional radiation shielding necessary to protect the facility users. The high-density concrete would later provide significant challenges while characterizing the biological shield.

Another significant modification was dividing the graphite pile into two sections. Dividing the pile increased the amount of airflow through the reactor allowing an increase in the flux thus producing more neutrons for the researchers. While designing this modification, the engineers realized that it would be desirable to alter the width of the gap to obtain optimal airflow conditions within the reactor pile. To achieve this, one half of the pile was placed on a track that could be moved in and out. An additional design change staggered the air gap to prevent the escape of neutrons; a design that enhanced the performance of the reactor but ultimately created significant problems in characterizing the graphite pile.

During operation the reactor coolant, air, was drawn through the pile from the top, bottom, east, and west sides of the reactor and exited the north and south faces into air plenums. The air was directed from the plenum into two large below ground ducts approximately the size of subway tunnels. The air traveled along the ducts through exit air filters, coolant coils, and eventually into a 300-foot stack where it was expelled to the environment. The air leaving the reactor was so hot that it would have rapidly damaged the concrete walls of the belowground ducts. A series of metal liners and a secondary cooling air system were installed within the duct to cool the exhaust air and insulate the concrete. The design of the belowground duct liner is presented in Fig. 1.

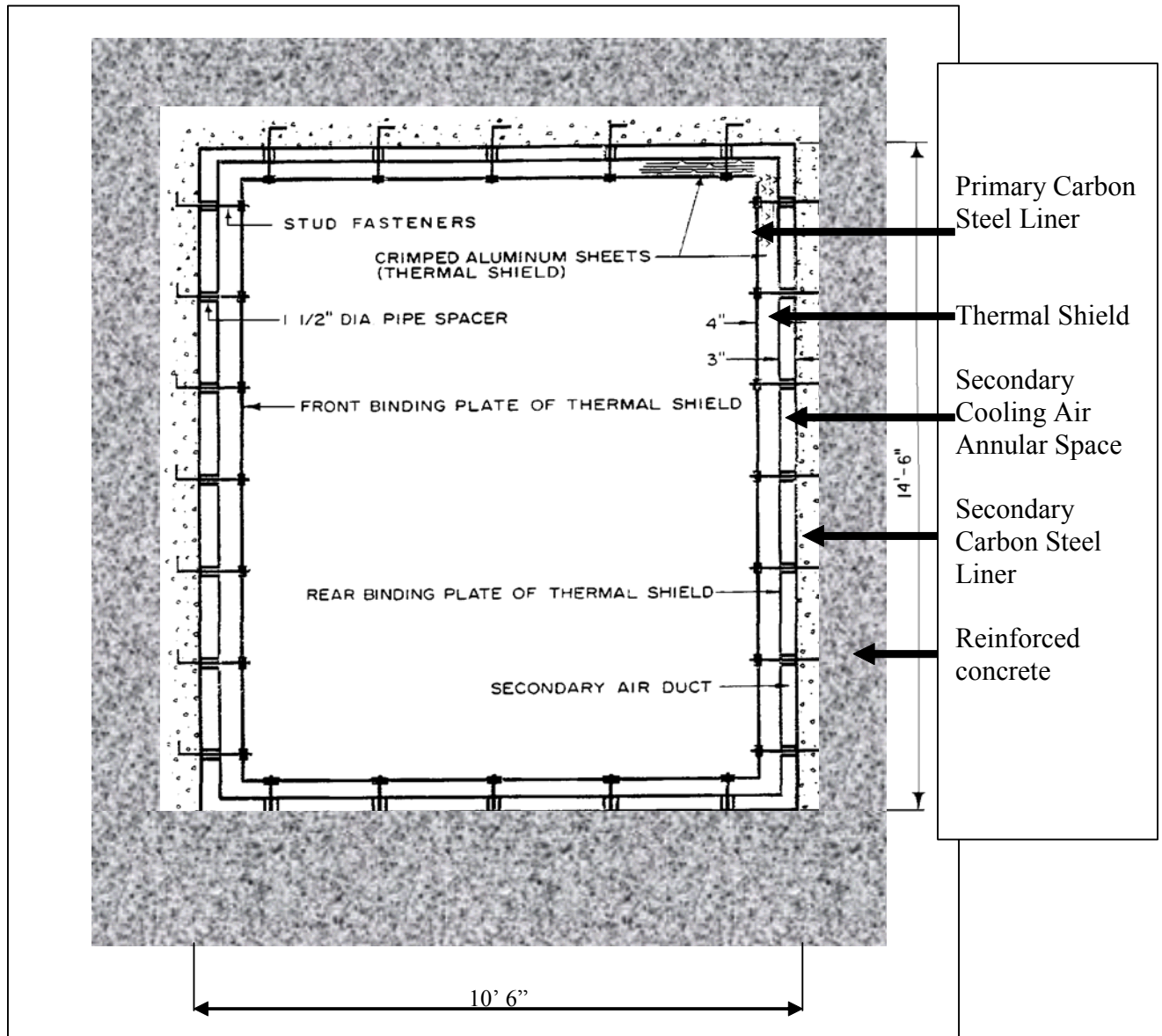


Fig. 1 Belowground ducts interior construction cross-section

An outer (secondary) steel liner was positioned directly on the concrete. During construction this liner served as an interior concrete form when pouring the walls and ceiling of the concrete duct. A thermal shield (primary liner) consisting of multiple layers of crimped aluminum sheets sandwiched between

inner and outer steel thermal binding plates was suspended within the secondary liner. Suspension of the thermal shield created a 3" air gap between the primary and secondary liner. The air gap allowed secondary cooling airflow, which removed the heat dissipated by the thermal shield. The design of the belowground duct system would become an essential piece in planning the characterization of the soils and groundwater associated with the BGRR.

Construction on the BGRR was completed in August 1950, and initial criticality of the reactor was achieved the same month. During its years of operation, it was one of the principal research reactors in the United States. Some of the research performed at the BGRR included the creation of the "Barn Book", a reference document used extensively in the studies of neutron absorption properties, medical radioisotope production, early boron neutron capture therapy, and its use in the Atoms for Peace program in which international researchers shared their work publicly. The science mission of the BGRR concluded in June 1963, and all research ceased when operation of the reactor was terminated and deactivation of the facility was initiated. In March 1972, the last fuel element was removed from the reactor. Shipment of the nuclear fuel to the DOE Savannah River Site was completed shortly thereafter. The BGRR complex was described as being in a "safe shutdown" condition by the U.S. Atomic Energy Commission and became a "Surplus Facility" within the DOE complex. From 1977 until 1997, portions of the facility were used as the BNL Science Museum.

In 1997, a BGRR site assessment was performed. During the assessment a deep drain sump connecting the low point drain lines from the north and south pile exhaust ducts was visually inspected. During the inspection it was noted that the deep drain sump had a significant level of water within it. Subsequent inspections of the below ground ducts found that they were flooded with approximately 57,000 gallons of contaminated water [2]. As a result of this finding and the likelihood that contamination from the water within the duct migrated into the soil, the BGRR was identified as Area of Concern 9 in an Interagency Agreement between the EPA Region II, the DOE, and NYSDEC initiating the BGRR Decommissioning Project.

## **CHARACTERIZATION PLANNING**

In March of 2002, a series of workshops were held at BNL to develop a comprehensive and clearly defined path forward that would meet the requirement of site closure by September 2005. Prior to these workshops, the BGRR decommissioning project was being performed in a series of removal actions without looking at the facility and its' components holistically. BNL recognized that a complete characterization of the reactor facility was needed in order to prepare a comprehensive Risk Assessment, Feasibility Study, and Proposed Remedial Action Plan to reach a consensus agreement with the interested parties on the final disposition of the BGRR. .

To fully understand the potential pathways for contaminant release to the environment, a comprehensive review of the facility construction, operational history, logbooks, and monthly reports were conducted. Fortunately, a small constituency of BGRR operations staff still resides in the local area and they were able to provide very useful information relating to some of the releases at the facility. Also, Robert Powell, a member of Borst's design team who would later become the BGRR plant manager, made several trips to BNL and provided in-depth knowledge of the facility construction, components, and operational history.

The sampling and analysis plans drafted to characterize the facility were designed to sample both biased and unbiased locations and concentrated around the main components: graphite pile, biological shield, belowground ducts, fuel canal, groundwater, building drain system, and reactor auxiliary systems. Also, the aboveground surfaces in common areas, office spaces, and workshops were fully characterized to provide the information needed to decide on the future use of the building. Biased locations allowed for

the study of known contaminated areas or leakage paths. The unbiased locations were useful to confirm or deny the existence of contamination in locations thought to be clean via historical and anecdotal assumptions.

## **CHARACTERIZATION RESULTS**

Over a 12-month period the BGRR characterization team obtained and analyzed in excess of 4,000 samples. Samples were obtained from surface and deep soil, reactor pile and belowground duct steel, biological shield concrete, aluminum thermal shield, reactor pile graphite, and groundwater. Samples were analyzed for radionuclides and hazardous constituents. Dose rate and smear surveys were performed in 75 fuel channels and 25 experimental ports. In the event contamination was detected, additional bounding samples were obtained to quantify the areal extent of contamination. In addition to the data collected, a considerable amount of knowledge about the facility construction, and release pathways was obtained during the performance of the characterization survey. The experience gained will be of great value when implementing the remaining remedial actions following the BGRR's final end-state decision.

### **Graphite Pile Characterization**

The characterization of the pile and biological shield included surveys and samples of the following components:

- Pneumatic transfer tubes
- Experimental and instrumentation ports
- Fuel channels
- Graphite moderator
- Graphite plugs
- Biological shield concrete
- Biological shield steel liner

The characterization of the pile, biological shield, and associated equipment was primarily a radiological characterization. The facility operating procedures were written to prevent introduction of hazardous materials so that the internals of the pile did not become chemically contaminated. Additionally, the heat generated within the pile during operations and subsequent continuous ventilation of the pile is expected to have removed any volatile and semi-volatile material.

The initial fueling utilized 11-foot long aluminum fuel elements, each loaded with 33 natural uranium fuel slugs. During the early operation of the reactor using the natural uranium fuel, 28 fuel failures were recorded. The first fuel element failure occurred in January 1952. Logbook entries indicate that as the workers attempted to remove the failed fuel element from the pile, gross contamination of tools and clothing occurred. Contamination was spread throughout the facility resulting in the facility shutdown and subsequent decontamination [3]. Historical records indicated that widespread contamination events were not uncommon in the early operating days of the BGRR. A modification to the reactor fuel was made in 1958, replacing the natural uranium fuel with 24-inch long aluminum clad enriched uranium fuel elements. There were no further fuel element failures recorded after replacing the fuel. Written records in the form of logbook entries, memos and general correspondence was extremely useful in designing the characterization approach.

As part of the pile characterization, seventy-five fuel channels, including the twenty-six channels where fuel element failures occurred, were inspected and surveyed. Inspection included a visual surveillance by remote camera, radiation dose rate surveys and removable contamination surveys. Access to the fuel

channels was gained through the south biological shield that historically was used to remove and replenish reactor fuel. Modular scaffolding was erected to reach the elevated fuel channels. For each penetration, the steel cover plate was removed and the biological shield plugs were withdrawn. The two inner graphite plugs were removed with a long-reach extension. Then a twenty-foot bridge tube was inserted into the penetration to span the plenum gap. The internal three-inch steel plate supported the inner end of the bridge tube. The fuel channels were accessed by removing the fuel anchors that were installed after the last defueling. The non-fuel channels were accessed by removing the retaining clips, springs, and graphite plugs [4]. Several of the miscellaneous items removed from the fuel channels exhibited neutron activation and were subsequently evaluated using an In-Situ Object Counting System (ISOCS®). Table I lists the ISOCS® results for a fuel anchor, two retaining springs, and several other components.

Table I Activity Concentration on Samples from Plenum Access - BGRR Graphite Pile Characterization, Gamma Spectrum Analysis with the ISOCS® Instrument

Description	Fine graphite powder in 500 ml bottle West Control-Rod Drive Top Plate	Plug Retainer Spring Scanner Slot #4 North Plenum	Fuel Anchor Portion (Hot end only, in 1L bottle)	Thermocouple wire, folded in 1L bottle scanner slot #4 North Plenum
Sample Mass	~ 1 g	71.6 g	45.1 g	31.6 g
Am-241 (pCi/g)	ND [ 74,000 ]	ND [ 79,100 ]	ND [ 4,200 ]	ND [ 69,400 ]
Cs-137 (pCi/g)	22,000,000 [ 51,000 ]	7,800 ± 7,100 [ 11,800 ]	7,600 ± 1,700 [ 2,470 ]	ND [ 7,670 ]
Co-60 (pCi/g)	578,000 ± 29,600 [ 21,000 ]	3,000,000 ± 120,000 [ 11,000 ]	732,000 ± 29,000 [ 2,110 ]	2,800,000 ± 100,000 [ 5,530 ]
Eu-152 (pCi/g)	ND [ 42,000 ]	ND [ 9,000 ]	ND [ 2,240 ]	ND [ 5,330 ]
Eu-154 (pCi/g)	ND [ 34,000 ]	ND [ 19,200 ]	ND [ 3,830 ]	ND [ 8,750 ]

ND = Not Detected

[ ] = Minimum Detectable Concentration in pCi/g

The large amount of Co-60 activity remaining within the retaining spring after 40 years of radioactive decay was surprising. Calculations indicate that each retainer spring would have contained 22 millicuries of Co-60 at the time of reactor shutdown.

Several of the fuel channels that had fuel element failures were visually inspected using a remote camera prior to performing radiological surveys. The visual inspection provided clear evidence of graphite deformity and damage from overheating. Figure 2 is a photograph that shows the effects of a fuel element failure in channel B-4-5. The groove at the bottom of the channel was a result of having to drill out the failed fuel element that had become lodged in the channel.

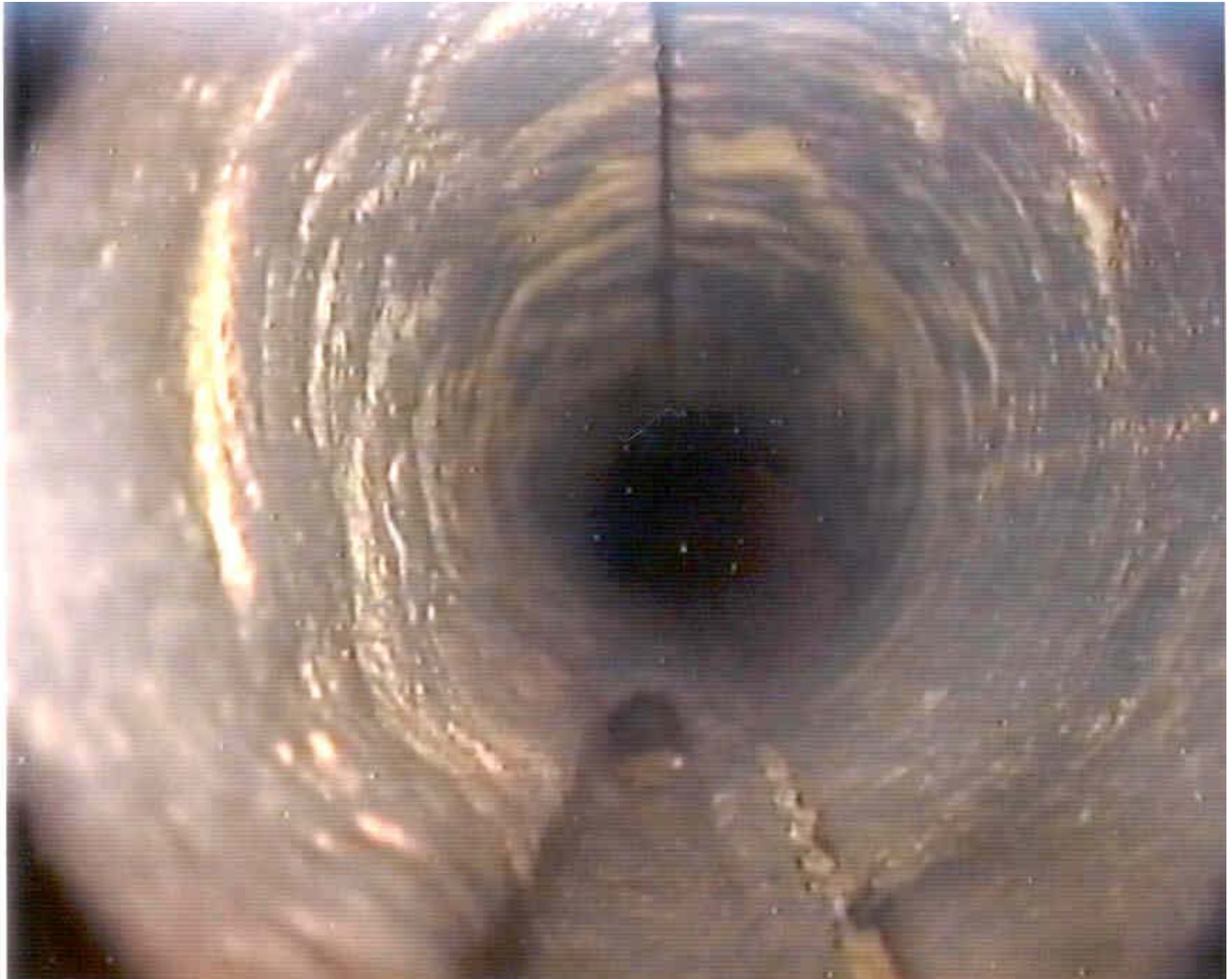


Fig. 2 Graphite damage in fuel channel B-4-5

Following the video inspection, radiation dose-rates were recorded at one-foot intervals throughout the fuel channel providing a radiological profile of the gamma and beta radiation levels within the channel. The highest gamma dose rate observed during channel profiling was in channel B-4-5 at 9.74 roentgens per hour (R/hr). Dose-rate surveys were performed on all seventy-five fuel channels that were inspected. Removable contamination smear surveys were taken from channels exhibiting elevated dose-rate readings.

Six visualizations were produced from the beta and gamma dose rate measurements obtained during the fuel channel profiling surveys. Each colored dot represents a one-foot increment where the dose rates were recorded. Of particular note are the high dose rates at the bottom of the pile within non-fuel bearing channels. It is assumed that the high dose rates are the result of debris collecting at the base of the pile near the air gap. To remove failed fuel from the north side of the pile the fuel assembly was pulled from the south side of the pile. While passing over the 3-inch air gap, debris collected was able to fall into the air gap and accumulate at the base of the pile. The configuration of the pile and biological shield prohibited obtaining a sample at the base of the air gap. Analytical results of similar type debris obtained in the north air plenum indicated high levels of Cs-137, Sr-90, Pu-239/240, Pu-238, Co-60, U-234, and U-235. Based on these results, it is expected that similar radionuclides exist in the debris at the base of the Pile. Figure 3 is a visualization of the gamma dose rate profile obtained from the surveyed fuel channels.

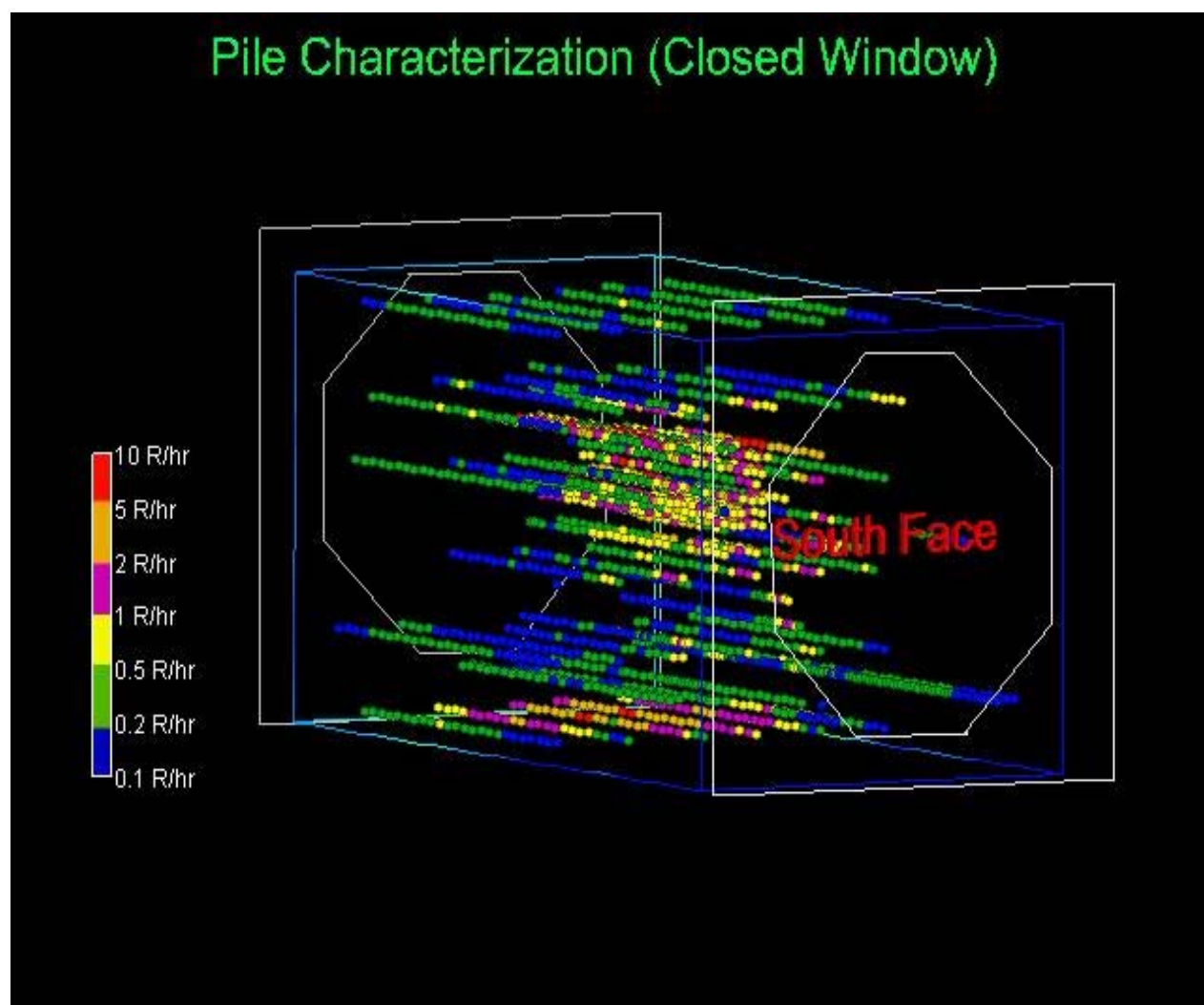


Fig. 3 Gamma dose rate readings from surveyed fuel channels

The graphite pile is divided into four concentric zones, A, B, C and D. The four zones reflect the four purity grades of graphite with the highest grade being "A" and the lowest "D". Samples of each grade of graphite were collected, including a sample of grade "AA" used in manufacturing the graphite plugs.

A right-angled air-operated die grinder with a ball-end center-cutting mill was used to obtain graphite samples within the pile. To collect the samples, the sampling device was inserted in an experimental port

and positioned within the pile at a location corresponding to the desired graphite type. The assembly was housed in a box and when in position was gradually rotated on axis to mill out the graphite sample. The operation was performed with long extension poles and monitored with a remote video camera. Analysis of the graphite indicated the primary radionuclides were C-14 and H-3 and to a lesser extent, Co-60, Cs-137, Sr-90, Ni-63, Eu-152, Eu-154, and Eu-155. The C-14 and H-3 constituents accounted for roughly 99.5% of the 3,300 curies in the pile. With a radioactive decay half-life of over 5,700 years, C-14 will be a key issue when evaluating decommissioning alternatives.

### **Biological Shield Characterization**

The protective shield surrounding the reactor pile is made of high-density concrete containing scrap iron and limonite, a mineral with a high content of water hydration. The concrete is rated at 350 pounds per cubic foot compared to 145 pounds per cubic foot for conventional concrete. The shield is four feet, three inches thick on four sides plus the top and has a cumulative weight of approximately 5,000 tons.

The overall dimension of the reactor with the shield in place is 55 feet long north and south, by 37 feet 6 inches wide east and west, and 33 feet 7 inches high. Proceeding from the graphite outward, the shield consists of six inches of steel plate (in some places it is two separate plates), four feet three inches of high-density concrete, and an outer casing of three inches of steel plate. The six inches of steel plate is supplemented with a band of 12 inches thick by 20 inches wide steel girdling the graphite around the gap. To complicate matters, the 3-inch thick plates used in the construction of the shield are armor strength steel salvaged from a World War II project to protect the Panama Canal.

The east and west faces of the graphite are covered with a close-fitting membrane of quarter-inch steel, and the top face is covered with eighth-inch aluminum plate that provides a seal between the inlet air and the seams in the graphite. Three-inch thick steel plates cover the north and south faces of the graphite pile [5].

A total of eight cores were made through the biological shield outer steel plate, concrete, and inner steel linings to determine the extent of activation in the components. The extremely heavy concrete required the project to have special drill bits manufactured to core through the concrete and iron punchings. Cooling water was pumped into the drill region and vacuumed out in a re-circulation system that minimized the volume of wastewater produced. Two cores were made through the east face, three through the west, two through the north, and one through the top. The locations of the core bores were chosen to determine the extent of neutron activation in high and low flux regions surrounding the pile.

The sample with the highest level of activation was located in line with the three-inch air gap that splits the pile into two halves. The neutron flux was highest at this location due to lack of moderation from the absence of graphite. The biological shield construction is slightly different at this location due to the higher flux. Because of the expectation of higher neutron flux in this region, designers incorporated six three-inch steel plates versus the typical arrangement of two three-inch plates. The additional thickness of steel provided the necessary shielding for the higher neutron flux. All six inner steel plates have detectable levels of activation with the plate nearest the graphite having the highest level of 300,900 pCi/g of Co-60. As was experienced when analyzing the fuel retaining spring, a significant level of Co-60 remains even after eight half-lives of radioactive decay. Steel shavings from the inner steel plate were sent to an off-site laboratory for full radiological analysis. C-14, Fe-55, Ni-59, Ni-63, and Co-60 were the principle radionuclides detected. Analysis of the high density concrete indicate that the twelve inches of concrete nearest the graphite exhibited elevated levels of Co-60 and Eu-152 activation but the remaining outer thirty-nine inches were less than MDA.



The cores made at or near the center of the pile had very similar results. Also, cores made through the biological shield near the edge of the fuel region had, as expected, very little activation of the concrete and steel. Estimates are that biological shield steel and concrete has a total curie content of approximately 4800 curies.

Lessons learned while performing the concrete and steel cores relating to the steel slugs within the high-density concrete and toughness of the 3-inch thick armor steel plates will be invaluable when determining the methods and approach of dismantling the biological shield.

### **Below-Ground Duct Characterization**

The primary goal of the below-ground duct characterization was to determine the nature and extent of radiological and non-radiological contaminants in the below-ground duct concrete, carbon steel liners, thermal shield, and associated soils.

The BGRR historical site assessment identified contamination inside the below-ground ducts. During operation, fission products, activation products, and transuranics from the pile were deposited on duct inner surfaces (the primary carbon steel liner). In addition, water leaked into the below-ground ducts both during operation and in the 30 years since the reactor was shut down. Based on evidence that this water leaked out, the Sampling and Analysis Plan identified extensive sampling of soils adjacent to the below-ground ducts for analysis.

Approximately 57,000 gallons of water were pumped from the below-ground ducts in 1997 and 1998 and the ducts have remained dry since. Each of the ten tankers into which water was pumped was sampled and analyzed for radionuclides. The specific activity in the water ranged between 0.01 to 0.02 microcuries per milliliter ( $\mu\text{Ci/ml}$ ). Two samples of sludge and debris were collected from the ducts and four samples from the filter area. The debris and sludge sample results indicated specific activity of the materials ranged between 0.3 to 7.4  $\mu\text{Ci/ml}$ . Spectroscopy results indicated that the predominant radionuclides were Strontium-90 (Sr-90) and Cesium-137 (Cs-137). These two radionuclides comprised over 90 percent of the total radioactivity in the debris and over 99 percent of the total radioactivity in the water [6].

A visual inspection via remote video equipment was performed at various locations throughout the system. Visible water lines on the walls of the duct liner and significant deterioration of steel components within the duct indicated that it was very likely that contaminated water had leaked from the ducts into the surrounding soils. While performing the visual inspection of the north duct, an undocumented stack of concrete shield blocks was observed in an area just beneath the south side of the pile. Discussions with Mr. Powell revealed that this was very likely the remnants of an early neutrino experiment performed by BNL researcher Dr. Ray Davis who later went on to win the Nobel Prize in Physics for detecting solar neutrinos.

The characterization of the concrete and steel structure consisted of taking 75 concrete and 79 samples of steel, aluminum, and residual corrosion products from 13 locations within the below-ground ducts. Four of the sample locations were taken at the secondary cooling air supply bustle located adjacent to Building 701 foundation. At this location, the base of both the north and south duct concrete were submerged making it a likely location for radionuclide migration through the concrete. Six of the samples were taken at locations adjacent to the four main expansion joints, another likely location for radioactive liquid to migrate from the ducts and contaminate the surrounding soil. The final three concrete sample locations were downstream of the coolers where the inside of the duct does not have a carbon steel liner.

To ensure that cross contamination was minimized and that the integrity of the ducts was maintained, a system of aluminum pipes were installed to guide the core drill and to provide a method of sealing the ducts. After the top of the duct was cored, the annular space was vacuumed to remove any debris found between the liners. An aluminum pipe was placed inside the annular space and fixed to the bottom carbon steel liner with hydraulic cement. The vacuum was then inserted into the pipe to remove any loose debris. Epoxy was poured into the pipe to fix any remaining loose contamination. After the epoxy hardened, the core drill was inserted into the pipe and the carbon steel liner was drilled to expose the duct concrete. The concrete was then cored to obtain samples and to allow the Geoprobe<sup>®</sup> access to the soil below. At the completion of Geoprobe<sup>®</sup> sampling, pipe plugs were installed and the tops of the sampling pipes were capped.

The primary, secondary, and thermal shield liner were all contaminated with up to 175,000 pCi/g of Cs-137. The contamination of the secondary liner in the north duct is on both sides of the liner indicating that water escaped the liner and entered the concrete. The south duct secondary liner did not show contamination on the concrete side, which indicates its integrity, was maintained. The concrete of the north duct had Cs-137 contamination to a depth of 15" and the south duct concrete was non-detect for Cs-137. The data clearly indicated water leaked from the north duct into the soil and it was unlikely that the south duct was a contributing factor [7]. Soil samples obtained near the north and south ducts further proved this conclusion.

The soil investigation of the BGRR below-ground ducts consisted of the advancement of 51 deep soil borings via Geoprobe<sup>®</sup> adjacent to and under the below-ground ducts and an additional 43 surface soil samples.

The deep soil contamination is in three distinct areas:

- The area adjacent to and below the north duct at the secondary cooling air bustle located 7 feet south of Building 701
- The area adjacent to and below the north duct innermost expansion joint located approximately 5 feet south of the pile
- The area associated with the north and south duct cooler drain sumps

As previously discussed, visual inspections and characterization results from the north duct indicate that contamination leaked from the primary liner into the duct's secondary air annular space, through the secondary steel liner working its way eventually through the concrete and into the soil. Leakage at the secondary cooling air bustle appears to have originated at a construction joint in the concrete. Characterization results indicate there is approximately 30 cubic yards of contaminated soils with specific activity levels up to 89,000 pCi/g of Cs-137 and 11,200 pCi/g of Sr-90. Sampling the soil beneath the inner most expansion joint posed a serious challenge requiring several difficult cores through up to seven feet of concrete. Once accessed, the joint was visually inspected and appeared significantly deteriorated. Because of the similarities in the leakage path and, concentrations of radioactivity, it was assumed that the extent of the contamination within the soil at the innermost expansion joint is similar to that found at the secondary air bustle.

The function of the north and south cooler drain sumps was to provide a collection point for cooling media leaks and tube condensation. During reactor operation, the sump level was monitored and water was pumped when it reached a pre-designated level. During an inspection performed in 1997 the sumps were found filled to capacity. It was determined that a source of water leaked into the cooler area and filled the cooler drain sumps which then overflowed into the lower portions of the below-ground ducts, eventually accumulating in the ducts lowest point, from the secondary cooling air bustle area to the north

and south pile exhaust plenums. The north and south cooler sumps are constructed of unlined concrete and over time water migrated through the concrete and into the surrounding soil.

A debris sample from the bottom of the cooler drain sump was obtained and analyzed. Results indicated the predominant nuclide was Cs-137 at a specific activity of 6,250 pCi/g in the north sump and 178,000 pCi/g in the south sump. A sample of the soil beneath the cooler drain sump indicated elevated levels of Cs-137 as expected.

### **Fuel Canal Characterization**

The fuel canal at the BGRR consists of a fuel chute connecting the south pile plenum to a deep pit where the spent fuel elements were stored to decay, then moved to a shallow portion of the canal where the fuel elements were segmented, packaged and stored awaiting shipment. The fuel chute, deep pit and inner portion of the canal are supported by the reactor foundation while the remainder of the canal is supported by its own foundation. A lead expansion joint connects the inner and outer canal; additional embedded steel construction joints are present within the extensive concrete run of the outer shallow canal. Water used within the deep pit and shallow canal was cycled through a water purification system connected to the fuel canal via a sump within the deep pit sump. The water purification system was housed adjacent to the deep pit.

Results of the characterization indicated pockets of soil contamination, up to 20,000 pCi/g of Cs-137 and Sr-90, were located along the main expansion joint seams along both walls and across the floor. Historical records also indicated that during early years of facility operation, the deep pit sump leaked. Several attempts were made to stop the leak, including the installation of a steel liner, but the sump continued to leak. Eventually, use of the sump was ceased and in 1961 the sump was filled with concrete in an attempt to stabilize the contamination within the sump. The unique design of the pile and building foundation made it extremely difficult to obtain a representative sample of soil below the deep pit sump. The bottom of the sump foundation is at the same elevation as the pile and deep pit foundations, thirty-five feet below the main floor of 701. In an attempt to obtain a sample, the main building 701 main floor and canal walkway floor were cored through and a Geoprobe<sup>®</sup> sample was obtained as close to the sump as possible. Results indicated elevated levels of Cs-137 and Sr-90 approximately five feet below and twelve inches west of the sump.

Surveys performed on the inner concrete surface of the canal indicated the concrete is contaminated with Cesium-137, Strontium-90, and to a lesser extent, Americium-241 and other transuranics. Analyses of concrete bore samples indicate that the contamination does not extend through the full thickness of the concrete structure with the exception of isolated areas in the vicinity of concrete mating surfaces created when initially pouring the canal structure [8].

### **Groundwater Characterization**

Previous groundwater investigations performed as part of the Operable Unit (OU) III Remedial Investigation identified a plume of Sr-90 groundwater contamination in the vicinity of the Waste Concentration Facility and the BGRR [9]. The previous investigation associated with the BGRR was conducted between 1995 and 1997 and samples collected from both groundwater monitoring wells and Geoprobess<sup>®</sup>. Results of the OU III groundwater investigation indicated a Sr-90 groundwater plume down gradient from the BGRR, Waste Concentration Facility, and pile fan sump area. The highest Sr-90 concentration detected in 1997, down gradient from the BGRR, was 53.9 pCi/l.

While the BGRR was believed to be a contributor to the Sr-90 groundwater contamination, the precise source within the BGRR complex was not identified. The below-ground ducts and canal were considered

as potential sources. Fifty-three groundwater samples were obtained in the vicinity of the BGRR to validate this assumption. Three samples obtained during the validation process indicated elevated levels of Sr-90. One sample, taken adjacent to the south duct at #3 expansion joint, had the highest concentration of Sr-90 at 540 pCi/l. The other two locations with elevated Sr-90 are in the same vicinity but their levels of Sr-90 were significantly lower with 11.7 pCi/l and 24.4 pCi/l. Since that time, additional groundwater samples have been taken down gradient of the below-ground ducts detecting Sr-90 in the groundwater up to 3000 pCi/L. Figure 4 is a summary of the soil and groundwater contamination found at the BGRR.

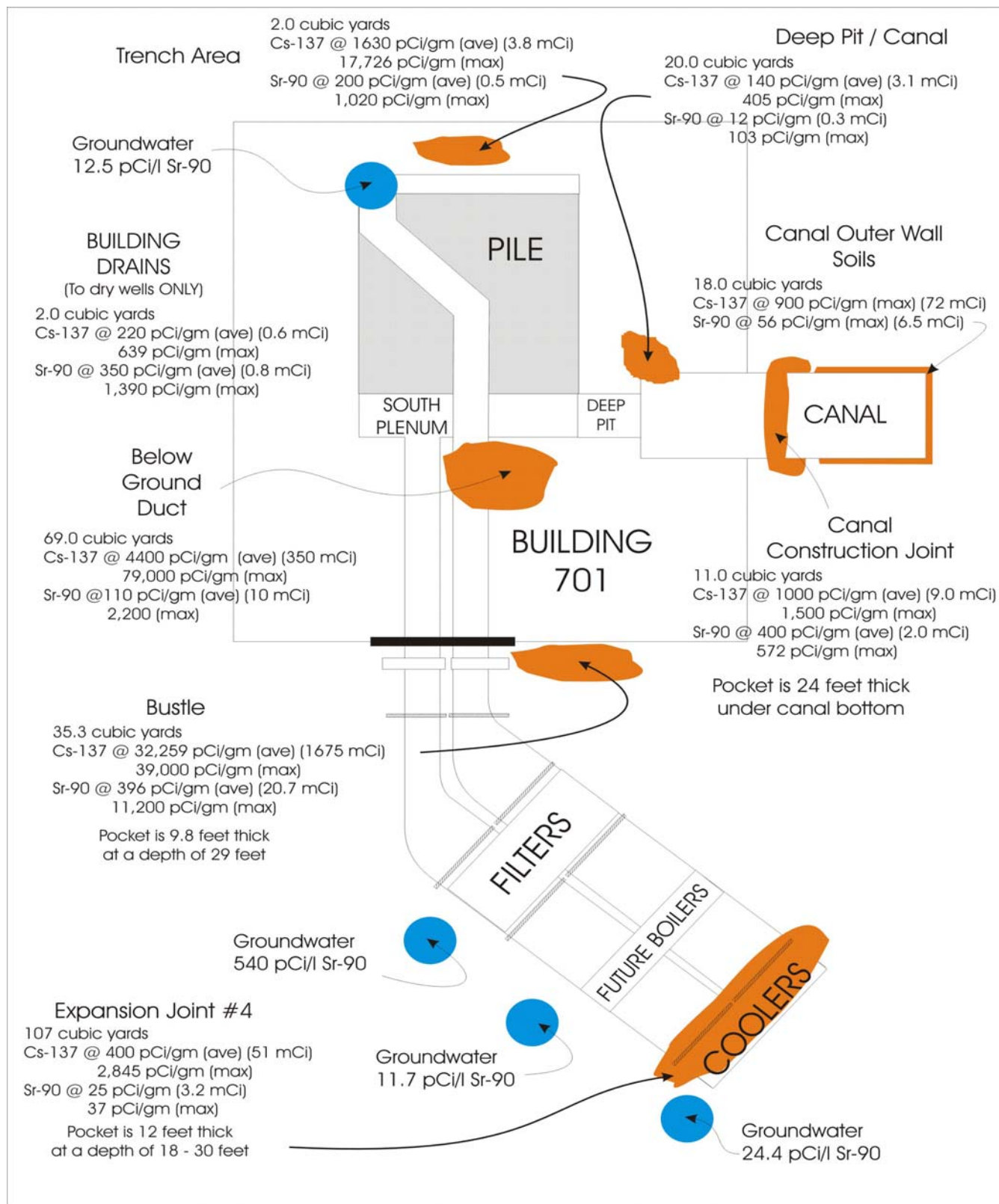


Fig. 4 BGRR soil and groundwater contamination

## **Building Drains**

There are a total of sixteen individual drain systems at the BGRR. During the characterization each of the systems were visually inspected, surveyed and sampled for radiological and non-radiological hazardous materials. Of greatest concern were the eight systems that terminated in dry wells near and below the building.

The drain systems associated with the three elevators all terminate in drywells. Results indicated that the drain systems from the passenger and freight elevators were free of contamination but the reactor fuel charging elevator indicated elevated levels of Cs-137 and Sr-90. Contamination was expected in this drain based on the review of the operating logs and the numerous events of facility contamination associated with fuel handling evolutions conducted from the charging elevator [10]. Due to the facility construction and piping arrangement, representative soil samples could not be obtained from the charging elevator drywell.

The majority of floor drains throughout the facility were dry systems used primarily for collection of residual liquids and no contamination was detected within the piping or near their drywells. The drains from the base of the intake air plenums presented a different problem; water was found in the drain lines that went to drywells. This was unexpected because the intake air plenums have been sealed for several years and there was no evidence of water intrusion detected. The conceptual model developed to explain the presence of water in the plenum drain drywells has several components. First, while coring through the main floor of Building 701 it was observed that the soil settled several inches beneath the concrete. Secondly, it is believed that water is able to migrate beneath the building by traveling along the underground concrete structures such as the below-ground ducts and air intake plenums. Lastly, engineering studies performed on the BGRR super structure and substructures concluded that the soil beneath the concrete foundation has undergone densification. These conditions working together create a pathway for water to migrate beneath the building and accumulate in low points such as the intake plenum drywells. A sample of the sludge within each the intake plenum drywells was sent for off-site laboratory analysis. Results indicated elevated levels of Cs-137 at 400 – 450 pCi/g, Co-60 at 3 –5 pCi/g and Sr-90 at 1500 – 1900 pCi/g. The evidence of water accumulating in these drywells and the presence of elevated radioactivity will certainly affect the decommissioning decisions to be made in the near future.

## **Above-ground Areas and Systems**

The characterization approach for the BGRR interior was selected to provide the radiological and non-radiological hazardous material data that is necessary to determine an acceptable end state of the facility. The characterization effort included the review of historical data, the conduct of surveys, and analysis of radiological and hazardous samples from 67 Survey Units within the BGRR complex.

The results of the hazardous material characterization indicate that asbestos, Polychlorinated Biphenyls (PCB's), and metals are present in many locations throughout the facility. Asbestos is present in floor tiles, insulation, ceiling tiles, and plasters. PCBs are present in locations where oil has contacted floor tiles under the Control Rod Drive Motor's, within the personnel elevator, the charging elevator, and the freight elevator. PCB's were also identified in scraping samples of interior wall coatings and paints. Lead and other heavy metals are present in original wall coatings and paints.

The results of the radiological characterization indicate that dispersible radioactive materials within Building 701 have previously been identified and contained through radiological posting and controls. Fixed contamination, radiation areas, and radioactive material storage areas are controlled consistent with ongoing work. The characterization survey concludes:

- Office areas contained no fixed or removable contamination above normal operational release limits
- The external surfaces of the reactor and support structures contained no fixed or removable contamination above normal operational release limits outside of existing radiologically posted areas. Within posted contamination areas, sporadic removable contamination was identified between 1,000-5,000 dpm / 100 cm<sup>2</sup> and discrete locations exhibited fixed beta contamination levels ranging between 5,000-50,000 dpm/100 cm<sup>2</sup>.
- Isolated areas of removable contamination were identified on the interior surface of the reactor building roof, "I" beams, and catwalks. Discrete locations exhibited fixed beta contamination levels ranging between 5,000-50,000 dpm/100 cm<sup>2</sup>.

Of the fifteen systems evaluated in this characterization report, seven are defined as radioactive, and seven are contained within controlled areas and must be suspected as radioactive. The remaining system, electrical distribution, is housed outside the confines of the building and is considered free of radioactivity [11].

## CONCLUSION

The BGRR was the first reactor built for peaceful scientific research. The research performed at the BGRR and the discoveries made are still being expanded upon today. The reactor was designed, built, and operated during a different era with significantly different operational and environmental rules and regulations. Accidents and environmental releases that occurred within the facility were routine and would not have been acceptable today. The results of the characterization indicate that there have been several environmental releases that have impacted the soil and groundwater in the vicinity of the BGRR complex. Also, the extent of contamination with BGRR related components such as the below-ground ducts, graphite pile, and biological shield will make full decommissioning very difficult and extremely expensive. At present, members of the Department of Energy, United States Environmental Protection Agency, and the New York State Department of Environmental Conservation are reviewing the characterization data to determine a suitable end state for the BGRR.

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