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DECOMMISSIONING OF RESEARCH REACTORS AT BROOKHAVEN NATIONAL LABORATORY: Status, Future Options and Hazards

by

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Summary

Three nuclear research reactors operated at the Brookhaven National Laboratory (BNL) during the period 1950-2000. The Graphite Research Reactor (BGRR) operated from 1950-1968, the High Flux Beam Reactor (HFBR) from 1965-1996, and the Medical Research Reactor (BMRR) from 1959-2000. Decommissioning of the BGRR is under way. Limited decommissioning has occurred at the HFBR, and no decommissioning has occurred at the BMRR. This report describes the reactors, reviews the planning and status of their decommissioning, summarizes available knowledge about each reactor's radioactive inventory and the associated risks, and reviews BNL's management of decommissioning. Conclusions and recommendations are provided.

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1. Introduction

During the period 1950-2000, a half-century that saw major investment by the US government in nuclear technology, three nuclear research reactors operated at the Brookhaven National Laboratory (BNL) on Long Island, New York.[1] The BNL site is operated for the US Department of Energy (DOE) by Brookhaven Science Associates. Each of the three reactors has now been permanently shut down, and they are in varying stages of deactivation and decommissioning. Deactivation of a reactor involves actions such as the removal of fuel and the draining of nonessential systems, in order to place the facility in a stable, known condition. These actions are followed by decommissioning, which is defined by DOE as follows:[2]

> "Decommissioning: Takes place after deactivation and includes surveillance and maintenance, decontamination, and/or dismantlement. These actions are taken at the end of the life of a facility to retire it from service with adequate regard for the health and safety of workers and the public and protection of the environment. The ultimate goal of decommissioning is unrestricted release or restricted use of the site."

Two of the three BNL reactors are undergoing decommissioning, and the third is in its deactivation phase.[3] For convenience, this report discusses all three reactors on a common basis, assuming that the third reactor will be decommissioned. The report begins by describing, in Section 2, each of the three reactors. Then, in Section 3, it reviews the planning and status of decommissioning for each reactor. The inventories of radioactive material at each reactor are discussed in Section 4. Scenarios that pose a potential hazard, in terms of exposing workers or the public to radioactive material, are discussed in Section 5. BNL's management of decommissioning is reviewed in Section 6, and the report's conclusions and recommendations are presented in Section 7. A bibliography is provided in Section 8.

Information used in this report was obtained from a site visit to BNL by the author in June 2002, and from the literature cited in the report. The author did not take field measurements of radioactivity. Nor did the author make independent estimates of radioactive inventories or the risks associated with hazard scenarios. This report adds value in two respects. First, it provides a concise overview of decommissioning issues at BNL. Second, it provides an independent, critical perspective on these issues.

This report relies heavily on information contained in BNL literature. Thus, if errors or omissions exist in BNL literature, similar errors or omissions may occur in this report. Also, it is possible that the author is unaware of some relevant items of BNL literature. To the author's knowledge, the BNL website provides the only publicly-available catalog

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[1] The operating periods of the three reactors were: 1950-1968; 1965-1996; and 1959-2000.
[2] DOE, 2000a, page D-3.
[3] The Graphite Research Reactor and the High Flux Beam Reactor are undergoing decommissioning, and the Medical Research Reactor is in its deactivation phase.

of BNL documents related to decommissioning of the BNL reactors. However, the catalog available on the website is incomplete. Some of the BNL documents cited here came to the attention of the author through other channels. BNL's documentation of reactor decommissioning is discussed further in Section 6 of this report.

DOE owns the three reactors and the BNL site. Also, DOE is the decision-maker regarding decommissioning activities. When a reactor passes from its deactivation phase to its decommissioning phase, administration of the reactor passes from one DOE office to another. For convenience, when this report discusses the management of reactor decommissioning at Brookhaven, it usually refers to BNL as the management entity.

In reviewing the decommissioning of one of the BNL reactors -- the Brookhaven Graphite Research Reactor (BGRR) -- one learns that this reactor experienced damage to fuel assemblies during the 1950s, leading to release of radioactive material into the reactor's flow of cooling air. Some of this radioactive material penetrated filters in the cooling-air pathway downstream of the reactor, causing radioactive contamination of the induced-draft fans that drew cooling air through the reactor. Downstream of these fans, the cooling air was released to the environment through a stack with no further filtration. Thus, some portion of the radioactive material released from the damaged fuel must have reached the environment through the stack. This report does not examine the nature or impacts of airborne releases of radioactive material from the BGRR to the environment during the reactor's period of operation. These matters may deserve consideration in other studies.

2. Description of the Brookhaven Reactors

2.1 Brookhaven Graphite Research Reactor

The Brookhaven Graphite Research Reactor is an air-cooled, uranium-fueled reactor, moderated and reflected by graphite.[4] Its primary missions were to refine reactor technology and produce neutrons for scientific experimentation. It operated from August 1950 to June 1968 and was then shut down because the Brookhaven High Flux Beam Reactor had become available, offering a more than 100-fold increase in neutron flux and other operational advantages over the BGRR.

Initially, the BGRR was fueled with natural-uranium metal slugs in aluminum cans, but from April 1958 onward it used enriched-uranium fuel in the form of aluminum-clad, uranium-aluminum alloy plate fuel assemblies. The design power level was 28 MWt with natural-uranium fuel and 20 MWt with enriched-uranium fuel. A BNL newsletter reported that the design power level was frequently exceeded.[5] The design level of neutron flux in the core with natural-uranium fuel was 4E+12 neutrons/sq. cm/sec.

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[4] Unless otherwise indicated, information in Section 2.1 is drawn from: BNL, 1949; BNL, 2000a; Burns and Roe, 1989; and the BNL website.

[5] BNL Public Relations Office, Bulletin Board, Volume 22, Number 31, 1 August 1968.

The Graphite Pile

The central feature of the BGRR is a graphite pile in the form of a 25-foot-per-side cube that weighs 750 tons and consists of 60 thousand graphite blocks in 2,600 different shapes. The pile is divided into two equal parts by an adjustable vertical gap of approximately 3 inches; the gap crosses the pile in an East-West direction. In the North-South direction the pile is penetrated by 1,368 horizontal, cylindrical channels in a 37 x 37 square array. These channels each have a diameter of 2.7 inches and are spaced on 8-inch centers. The center hole in the 18th row is omitted because a 12-inch-square hole penetrates the pile horizontally from North to South at this location. This hole allowed the insertion of experimental devices into the pile. When these devices were not present, a 25-foot-long removable core made of graphite blocks was placed in this hole.

Fuel assemblies were placed in the 1,368 channels, except that one or more of the outer rows of channels did not receive fuel but were filled with graphite plugs. The fuel assemblies were cooled by air flowing through the channels from the central gap to the outer (North and South) faces of the pile. Thirty additional channels with a diameter of 2.7 inches also penetrate the pile horizontally from the North to the South face, each being displaced 4 inches from its companion channel in the 37 x 37 array. Only one channel of each of these pairs contained fuel, the other being filled with graphite plugs.

Experimental Access

Thirty experimental holes, each 4 inches square, penetrate the pile horizontally from East to West. These holes were used for insertion of experimental devices. When not in use, these holes were filled with graphite plugs. Two additional holes, 2 inches wide by 3 inches high, penetrate the pile from East to West, providing a passage for the target conveyor system. This system consisted of an endless chain of graphite sample holders which carried samples through the pile. A second system for inserting test specimens into the pile for short periods of irradiation employed 11 pneumatic tubes extending horizontally from the North face of the pile to the mid-pile gap. Each tube is made of aluminum and has an outer diameter of 3 inches.

Larger test specimens could be irradiated in exposure chambers below and above the pile. Below the pile, near its mid-point, are an animal chamber (3 ft long, 2 ft wide, 2 ft high) and an ion chamber (3 ft long, 1 ft wide, 1 ft high). Test specimens were transported to these exposure chambers by chain-driven cars moving through East-West tunnels under the pile. The cars could be loaded or unloaded at either side of the pile, passing through lead-doored airlocks en route to and from the pile. Above the pile, a 20-foot by 20-foot horizontal area of the radiation shield above the pile is removable in 4-foot-square sections, using an overhead crane. Removal of these sections would create an exposure chamber into which large test specimens, including animals, could be lowered.

Control Systems

The primary control system featured 16 boron-steel control rods, each 2 inches square and 27 feet long. These rods penetrate the pile horizontally from the Southeast and Southwest corners, in directions parallel to the diagonals of the pile footprint. At each of the two corners there is an array of eight rods. Viewed along the (horizontal) axes of the rods, the array has four rows and two columns, with vertical and horizontal spacing of 4 feet between rods.

Backup control was provided by two additional systems to provide emergency shutdown. One system employed two vertical boron-shot wells consisting of aluminum tubes extending 18 feet down into the graphite pile, plus two diagonal boron-shot wells of similar construction. The diagonal wells are in the gap region of the pile, inclined at 45 degrees to the horizontal. For emergency shutdown, boron-steel balls of 5/16-inch diameter would be discharged from hoppers into the shot wells. The second emergency-shutdown system involved the injection of a liquid "neutron poison" into the 11 pneumatic tubes whose primary purpose was to carry samples into and out of the pile.

Radiation Shield and Reactor Building

The pile is surrounded on all faces except its base by a radiation shield. This structure is 55 feet long, 37 feet 6 inches wide and 33 feet 7 inches high in its outer dimensions. From the inside out, the layers of this shield are: (a) 6 inches of steel plate to absorb thermal neutrons; (b) 4 feet 3 inches of high-density concrete containing iron and steel pieces as coarse aggregate and iron-rich limonite as fine aggregate, yielding an overall iron content of 70 percent by weight; and (c) an outer casing of 3 inches of steel. The inner 6-inch steel plate is supplemented by a steel band 12 inches thick and 20 inches wide girdling the gap region of the pile, where the radiation flux was highest. For each of the penetrations of the pile that is described above, such as the fuel channels, there is a corresponding penetration of the radiation shield.

The radiation shield, the graphite pile and their supporting structures are designated by BNL as Building 702. These items are enclosed within Building 701, which houses an overhead crane, protects equipment and personnel from the weather, and provides offices and other support facilities.

Cooling System

Cooling of the BGRR was provided by once-through, induced-draft circulation of air. Ambient air was drawn in through filters located on the outer East and West faces of Building 701, and traveled through ducts to the gap in the pile. After passing through the fuel channels toward the North and South faces of the pile, the air entered the North and South Plenums respectively. The design level of the air temperature upon exit from the pile was 220 degrees C for a power output of 28 MWt. The North and South Plenums are cavities between the pile and the radiation shield. Each of these two plenums communicates exclusively with an underground air duct through which the cooling air was exhausted below Buildings 702 and 701 and toward the South.

Outside Building 701 and to its South, each of the two underground exhaust-air ducts communicated exclusively with an underground filter chamber followed by an underground air-cooling chamber. In the two cooling chambers, heat exchangers transferred heat from the exhaust air to water that was in turn cooled in a cooling tower. The design level of the air temperature upon exit from the cooling chambers was 60 degrees C. Downstream of each of the two cooling chambers, an exhaust-air duct emerged above ground, the two ducts merged into one duct, and this duct entered the Fan House (Building 704). Five induced-draft fans in that house discharged the exhaust air to the atmosphere via a stack.

Spent-Fuel Canal

Fresh fuel assemblies were inserted into the pile from its South face, and spent fuel assemblies were discharged from the same face. When spent fuel assemblies were discharged, they were dropped into an opening in the floor of the South Plenum, from which a slanted chute led to the floor of a spent-fuel pit that was filled with water to a depth of about 20 feet. This pit communicated directly with a below-grade spent-fuel canal that passes under Building 701 and emerges toward the East. The canal was filled with water to a depth of about 8 feet. The volume of water in the canal and pit was about 55,000 gallons. Both the pit and the canal were made of concrete, and the pit was lined with tiles. The Canal House (Building 709) covered the exposed portion of the canal outside Building 701.

After discharge from the pile, a spent fuel assembly was stored in the pit for a cooling-down period that allowed some of its radioactive content to decay, and was then transferred to the canal for a further period of storage. Each of the fuel assemblies used in the BGRR during its first 8 years of operation consisted of 33 natural-uranium metal slugs in a finned aluminum can 11 feet long. As part of preparations for removal of spent fuel from the BGRR, the cans were chopped under water and the slugs were taken out. The lengths of empty can were removed from the canal and stored dry in underground vaults to the South of Building 709. Eventually, the slugs were taken out of the canal and removed from the BGRR. The dry-stored lengths of empty can were also removed from the BGRR. From April 1958 onward, the BGRR used enriched-uranium fuel assemblies that were 2 feet long. These assemblies did not need to be chopped. After periods of storage in the pit and the canal, these assemblies were removed intact from the BGRR.

2.2 High Flux Beam Reactor

The Brookhaven High Flux Beam Reactor (HFBR) is a uranium-fueled reactor that was moderated and cooled by heavy water.[6] It began operating in October 1965. In December 1996 it was shut down for a routine refueling but did not resume operation thereafter. In December 1999 the Secretary of Energy announced that it would be shut down permanently. During its years of operation, its primary purpose was to produce beams of neutrons that were guided from the reactor to experimental areas by nine horizontal tubes known as beam tubes. Irradiation of specimens could also occur in thimbles in or near the reactor core. Initially, the HFBR had a power rating of 40 MWt. This rating was raised to 60 MWt in 1982 and then decreased to 30 MWt in 1991. When operating at 40 MWt, the HFBR produced a peak neutron flux of about 1.6E+15 neutrons/sq. cm/sec. The neutron flux varied roughly in proportion to the power level. Peak neutron flux occurred outside the reactor core.

The Reactor Core

The HFBR core was composed of 28 vertical-axis fuel assemblies resting on a grid plate at the base of the reactor vessel. These assemblies were 57 inches long, with an active length of about 21 inches. In the active region of each fuel assembly, highly-enriched (more than 95 percent U-235) uranium was contained in 21 curved plates. Each plate was made of uranium-aluminum alloy clad with aluminum. The total U-235 loading of the core was 7.7 kg. Sixteen control-rod blades (8 main blades and 8 auxiliary blades) were arranged around the outside of the core, suspended from drive rods. Cooling water (heavy water) flowed downward through the fuel assemblies. Above the core was a stainless-steel transition plate suspended from a cylindrical flow shroud made of aluminum.

The Reactor Vessel and Cooling System

The reactor vessel is a one-piece aluminum structure about 21 ft tall. Its lower portion is a sphere with an inside diameter of 82 inches and a wall thickness of 1.75 inches. Its upper portion is a cylinder with an inside diameter of 42 inches and a wall thickness a little over 2 inches. The mid-height of the active portion of the reactor core was approximately at the center of the sphere that makes up the lower portion of the vessel.

The lower, spherical portion of the reactor vessel is penetrated by aluminum beam tubes. Each tube is open where it penetrates the vessel, and has a thin-walled hemispherical termination at its inner end, close to the reactor core. During reactor operation, neutrons would enter each tube at its inner end, travel through the tube and aligned holes in the thermal and radiation shields, and enter an experimental apparatus. The tubes form part of the pressure boundary of the reactor vessel. With one exception, the beam tubes are

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[6] Unless otherwise indicated, information in Section 2.2 is drawn from: BNL, 1964; DOE, 1999; DOE, 2000b; and the BNL website.

configured so that neutrons would emerge from the vessel along pathways roughly tangential to the reactor core. The exception (beam tube H-2) follows a radial pathway. Four removable thimbles (closed-end pipes) made of aluminum enter the core region from above. Three fixed thimbles pass through the wall of the reactor vessel. These thimbles were used for irradiation of samples.

During reactor operation, a flow of about 17,000 gpm of primary cooling water (heavy water) entered the reactor vessel through the side of its upper portion, and then passed downward within the flow shroud and into the core. Upon exiting the core, the cooling water traveled upward around the outside of the flow shroud and left the vessel through the side of the vessel's upper portion. At the vessel exit point, the cooling water had a temperature of about 130 degrees F and a pressure of about 170 psig.[7]

The primary coolant was circulated through heat exchangers that were cooled by a flow of light water on the secondary side. The secondary coolant was in turn cooled by passing through cooling towers.

Thermal and Radiation Shields

The reactor vessel is surrounded by a thermal shield whose purpose during reactor operation was to limit the energy flux into the concrete biological shield to no more than 7.5E+11 MeV/sq. cm/sec. This provision limited the temperature rise in the concrete to no more than 50 degrees F, to prevent thermal stress cracking. The energy flux on the inner surface of the thermal shield was about 5E+13 MeV/sq. cm/sec (predominantly due to epithermal neutrons) at a power level of 40 MWt.

The lower thermal shield is a series of cylindrical steel shells, some separated by cooling water and some by lead, making up a cup that surrounds the lower, spherical portion of the reactor vessel. The walls of this cup have holes that correspond to the locations of the beam tubes. Combined, the wall thicknesses in the lower shield are 3 inches of steel, 1.25 inches of water and 4.75 inches of lead. The lower shield weighs 110,000 pounds, of which 70,000 pounds is due to lead. This shield has an outside diameter of 9 feet, an inside diameter of 7.5 feet, and a height of 10.5 feet. The lower shield is surmounted by an upper thermal shield in the form of a collar around the upper, cylindrical portion of the reactor vessel. The upper shield weighs 30,000 pounds, and is made primarily of steel.

Beyond the thermal shield is a radiation shield, a structure 38 feet high with an outer surface roughly 12.5 feet from the center-line of the reactor vessel. Materials used in the radiation shield are steel plates, lead, and concrete that incorporated limonite as fine aggregate and steel punchings as coarse aggregate. At the top of the radiation shield, its cross-section is a regular octagon with a circular hole in the center. At the level of the reactor core, the cross-section of the shield is an irregular nonagon with a circular hole in the center; each face of the nonagon is pierced by one of the nine beam tubes. Each beam tube can be closed by a vertically-moving shutter at the inner side of the radiation shield.

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[7] Coolant flowrate, temperature and pressure values in this paragraph are for a power level of 40 MWt.

These shutters vary in weight from 3,000 pounds to 10,000 pounds, and are 30 inches thick in the beam direction. Sixty percent of the volume of a shutter consists of steel, and 40 percent of water.

Spent-Fuel Canal

The reactor was refueled from above. Spent fuel was removed following a 12-24 hour cooling period after reactor shutdown. During this operation the water level in the reactor vessel was raised and the vessel's lid was removed. Spent fuel assemblies were lifted out of the vessel and dropped into a discharge chute. The assemblies slid down this chute and entered the spent-fuel canal, where they were manually placed in a temporary storage rack. After a period of storage in this rack, the assemblies were cut under water to separate their active and inactive portions, and the active portions were placed in 4-slot or 6-slot storage baskets. Eventually, each filled basket was placed inside a shipping cask and removed from the HFBR. When filled, the canal held about 68,000 gallons of water. The canal is made of concrete, and its walls were glazed with tiles.

Reactor Building

The reactor, the spent-fuel canal and their supporting equipment are housed in a reactor building whose outer wall is a hemispherical steel shell. Inside the building are rows of offices and large open areas where experimental equipment was located.

2.3 Brookhaven Medical Research Reactor

The Brookhaven Medical Research Reactor (BMRR) is a uranium-fueled reactor that was cooled by light water, moderated by the light-water coolant and graphite blocks in the core region, and surrounded by an air-cooled graphite reflector.[8] Its mission was to produce neutrons for medical research. It operated from March 1959 to December 2000. Initially designed for a power level of 1 MWt, the BMRR began operating at 3 MWt during its first year of operation. Its power level was later raised to 5 MWt. Peak neutron flux in the core was 5E+13 neutrons/sq. cm/sec.

The Reactor Core

The BMRR's fuel assemblies each contained 18 curved metal plates about 25 inches long and 3 inches wide. These plates were made of uranium-aluminum alloy containing 12 percent by weight of "fully-enriched" (more than 90 percent U-235) uranium, with a thin outer layer of aluminum. The U-235 content in each plate was 7.77 grams. An aluminum grid plate 5 inches thick and 23 inches in diameter supported the fuel assemblies and other core components, providing spaces for 32 fuel assemblies and 4 control rods. Some of the fuel-assembly spaces were not used for fuel. Those spaces

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[8] Unless otherwise indicated, information in Section 2.3 is drawn from: BNL, 1960; BNL, 1956; and the BNL website.

were occupied by blocks of graphite. When operating at power levels up to 3 MWt, the core contained 17 fuel assemblies.

Four control rods were provided, consisting of one regulating rod and three safety rods. The regulating rod is a solid stainless-steel bar 0.5 inches by 2.5 inches in cross section and 26 inches long. The active length of each safety rod is a stainless-steel tube 0.75 inches by 2.5 inches in cross section and 26 inches long, lined with a cadmium tube and filled with boron carbide grains. The control rods are suspended by extension rods from control-rod drives mounted on top of the radiation shield that covers the top of the reactor tank.

The Reactor Tank

The core was located within a vertical, aluminum tank about 20 feet tall. In its lower (cylindrical) portion, the tank has an internal diameter of 23.5 inches and a wall thickness of 0.25 inches. The reactor core fitted snugly inside this portion of the tank. Cooling water entered the tank from below, passing through the grid plate and rising through the fuel elements. In its upper (cylindrical) portion, the tank is 12 feet tall and 64 inches in diameter, with a wall thickness of 0.5 inches. The upper and lower portions of the tank are joined by a conical transition piece with a wall thickness of 0.5 inches.

The top of the reactor tank communicates with the atmosphere in the reactor building but is covered by a radiation shield. Cooling water filled the tank up to about 6 inches from its top, and was discharged from the tank through a pipe that intersects the tank wall about 2 feet from its top.[9] The discharged cooling water was pumped through a heat exchanger and then returned to the base of the reactor tank. On the secondary side of the heat exchanger, heat was transferred to groundwater that was extracted from onsite wells and then returned to the ground via diffusion wells.

The Reflector

The core region of the reactor tank is surrounded by an air-cooled, graphite reflector. Between 5 and 10 percent of the fission heat released in the reactor was deposited in this reflector. Surplus graphite from the BGRR was used to construct the reflector, which consists of 724 blocks with a total weight of 10 tons. Outer dimensions of the reflector are 68 inches in height, 101 inches North to South, and 63 inches East to West.

Cooling air for the reflector was drawn from inside the reactor building. After passing through the reflector, cooling air traveled through a duct to a filter bed in the basement of the East wing of the reactor building. Two induced-draft fans drew the air through this filter bed and discharged it to the atmosphere via a stack. A separate building-ventilation system drew air from the basement of the reactor building, passed it through a filter, and discharged it via the stack. To complete these circuits, air entered the reactor building

[9] The water capacity of the reactor tank is about 2,000 gallons.

through an air shaft above the North wing of the building, passed through an air-conditioning unit, and was released in the main and upper levels of the building.

Thermal and Neutron Shield

Surrounding the reflector is a laminated, mild-steel thermal shield to protect the concrete radiation shield from overheating due to gamma irradiation. Inside this thermal shield is a neutron shield made from boral (boron carbide in an aluminum matrix) and boron carbide in aluminum cans. Above and below the reflector and at its North face, the thermal shield is 3 inches thick. At the East and West faces, the thermal shield consists of a 3-inch-thick inner plate sandwiched between two 2-inch-thick plates, with a 0.5-inch air gap between each layer. There is no thermal shield at the South face of the reflector, which is directly exposed to the Broad Beam Facility as discussed below. The neutron shield experienced a neutron flux of about 1E+13 neutrons/sq. cm/sec at a 1 MWt power level. At the East and West faces, the neutron shield consists of 1-inch thick aluminum cans containing boron-carbide grains. Elsewhere, the neutron shield consists of 0.25-inch-thick boral plates.

The Radiation Shield and Experimental Access

Surrounding the thermal shield is a radiation shield made of high-density concrete containing iron-rich limonite as fine aggregate and iron and steel pieces as coarse aggregate. About 125 cubic yards of this concrete were used. The thickness of the concrete shield varies from about 2 feet to about 5 feet. The outer, vertical faces of the radiation shield are oriented to the four points of the compass. Each face has an opening that allowed radiation flux from the reactor to reach an experimental area.

To the South, the graphite reflector opens directly onto the Broad Beam Facility. This is a small experimental room that can be accessed from outside the reactor by opening a sliding door made of high-density concrete. The radiation flux entering this room from the reactor could be tuned by placing in its path filters made of graphite, lead, cadmium or uranium. These filters allowed the intensity and spectrum of gamma or neutron radiation to be adjusted to suit a particular experiment.

To the East is an Animal Treatment Facility, and to the West is a Patient Treatment Facility. Movable, concrete shutters allowed radiation flux from the reactor to enter these chambers as required. The flux was tuned by filters to accentuate thermal neutrons and minimize fast neutron and gamma radiation. A port in the North face of the radiation shield allowed radiation to pass directly from the reactor core to an experimental chamber.

3. Planning and Status of Decommissioning

3.1 Brookhaven Graphite Research Reactor

Several decommissioning activities have been completed at the BGRR, others are in progress, and additional activities are under consideration. The ultimate scope of decommissioning at the BGRR has not been decided.[10]

Completed Activities

After the BGRR was shut down in June 1968, fuel assemblies were removed from the pile and eventually from the site. The last batch of fuel assemblies was shipped from the BGRR to the Savannah River Site in June 1972. Control rods were disconnected from their drive mechanisms and inserted into the pile. Items of experimental equipment were removed from Building 701. Penetrations of the radiation shield surrounding the graphite pile were covered up or plugged. The spent-fuel canal was pumped dry, cleaned with soap and water, and covered with concrete slabs to provide shielding. During the period 1985-1986, piping and equipment were removed from the Canal House (Building 709) and its associated Water Treatment House (Building 709A) to avoid further deterioration of these items and reduce the potential for flooding of the canal. Between 1977 and 1997, portions of Building 701 were used by the BNL Science Museum.

The present phase of decommissioning began in 2000. Early in that year, the induced-draft fans and associated equipment in the Fan House (Building 704) were removed. In March 2000, the Pile Fan Sump and contaminated soil under it were removed. The Pile Fan Sump was an underground concrete structure, with an access hatch located in the open air, that received water from various drains associated with the BGRR, the HFBR and nearby laboratory facilities. In June 2000, removal of the above-ground portion of the exhaust-air ducts was begun. The last section of these ducts was removed in March 2001. During 2003, debris was removed from the spent-fuel pit. Also, preparations were made for removal of the filters from the underground filter chambers that are in the exhaust-air pathway downstream of the pile. Filter removal began in October 2003 and was completed in January 2004.

The most challenging portions of the BGRR in terms of decommissioning are those portions that are within, under or around Building 702. These portions include the graphite pile, the radiation shield that surrounds the pile, the spent-fuel pit, the underground exhaust-air ducts, supporting structures for these items, and contaminated soils under or around Building 702. As of February 2004, no significant decommissioning work has been done on these portions of the BGRR. Work has been done to characterize these portions in terms of radioactive inventories and other characteristics that are relevant to decommissioning.

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[10] Unless otherwise indicated, information in Section 3.1 is drawn from: BNL, 2000a; BNL, 2000c; BNL, 2001b; BNL, 2001c; BNL, 2002b; Burns and Roe, 1989; Adey, 2002; Petschauer, 2003; and the BNL website.

Options for Decommissioning at the BGRR

An analysis of decommissioning options at the BGRR was published by BNL in April 2000.[11] To date, no other systematic analysis of decommissioning options has been published. A December 2003 BNL presentation to the Community Advisory Council did not mention the April 2000 analysis, implying that BNL is taking a new approach to decommissioning of the portions of the BGRR within, under and around Building 702.[12] This apparent change in BNL's thinking is discussed later in Section 3.1. The following paragraphs summarize the decommissioning options set forth by BNL in April 2000.

Some of the decommissioning options discussed in BNL's April 2000 document have been overtaken by events, in the sense that items have been removed that would, under the stated options, have remained. However, this document continues to provide a useful perspective on future decommissioning options, especially those associated with Building 702. The document describes seven broad alternatives for decommissioning of the BGRR, as follows:

Alternative 1
This is the "no action" alternative, in which all structures remain intact, with surveillance and monitoring for 50 years. Events have overtaken this option, which is discussed by BNL only as a baseline for comparison.

Alternative 2
This alternative includes a set of work elements that are common to Alternatives 2-7, plus removal of some contaminated soils and sealing of the pile. Like Alternative 1, this option has been overtaken by events.

Alternatives 3 and 4
These alternatives add to Alternative 2 a set of decommissioning activities focussed on the spent-fuel canal and the air-cooling system downstream of the reactor. Many of these activities, which involve removal of substantial amounts of contaminated material, are completed, ongoing or planned. Alternative 4 differs from Alternative 3 only in that it would encompass the removal of equipment in Building 701, including the control-rod drives and the fuel-charging elevator.

Alternative 5
In this alternative, the actions taken in Alternative 4 would be supplemented by cocooning of Building 702 and removal of Building 701.

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[11] BNL, 2000a.
[12] Petschauer, 2003.

Alternative 6
Alternative 6 would involve the actions taken in Alternative 5, except that Building 702 would be removed and Building 701 would remain in place.

Alternative 7
In this alternative, the actions taken in Alternative 6 would be supplemented by removal of Building 701.

The document states that Alternatives 2, 5 and 7 are not recommended for further consideration.[13] Thus, Alternatives 3, 4 and 6 would be open for consideration.

BNL's Current Planning for Decommissioning Associated with Building 702

BNL's December 2003 presentation to the Community Advisory Council did not mention the April 2000 document.[14] Instead, it briefly outlined three alternative options for decommissioning of the portions of the BGRR within, under and around Building 702. The options are:

Long-Term Institutional Control
The graphite pile, radiation shield, spent-fuel canal and underground air-exhaust ducts would remain in place. The various openings in these items would be capped. Building 701 would remain.

Pile and Radiation Shield Removal
The graphite pile and the radiation shield would be removed, but underground structures would remain in place. Building 701 would remain. The incremental cost of this option, compared with Long-Term Institutional Control, would be about $40 million. (Cost calculations were not provided.)

Greenfield
The graphite pile, the radiation shield, Building 701 and most underground structures (canal, air-exhaust ducts, etc.) would be removed. The incremental cost of this option, compared with Long-Term Institutional Control, would be about $96 million. (Cost calculations were not provided.)

BNL's December 2003 presentation stated that Long-Term Institutional Control is the "current planning case". This option appears to be similar to Alternatives 3 and 4 in the April 2000 document. However, a proper comparison cannot be made until more information is available about the newly-proposed options. At present, it is not clear if BNL is taking a new approach to planning and decision-making regarding decommissioning associated with Building 702.

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[13] BNL, 2000a, page ES-4.
[14] Petschauer, 2003.

3.2 High Flux Beam Reactor

The HFBR was shut down for a normal refueling in December 1996.[15] During that refueling, data from newly-installed monitoring wells showed that water leaking from the spent-fuel pool had created an underground plume of tritium-contaminated groundwater. The leakage rate was estimated at 6-9 gallons per day.

The Tritium Remediation Project

After discovering groundwater contamination by tritium, BNL initiated the Tritium Remediation Project (TRP). Its purpose was to identify and isolate the source of the tritium plume, seeking to ensure that tritium levels (in groundwater) above the drinking-water standard did not occur beyond the site boundary. BNL removed all spent fuel from the HFBR and shipped it to the Savannah River Site during 1997. According to a DOE report, 14 reactor control-rod blades and other highly-activated reactor-vessel internals that had been stored in the spent-fuel pool were placed into dry-storage containers on the site.[16] This statement invites question.[17]

The spent-fuel pool was drained in January 1998, marking completion of the TRP. Approximately 68,000 gallons of pool water was pumped into onsite storage tanks. Then, the pool was equipped with a new, double-walled, stainless-steel liner. Piping systems, tanks and sumps that could contain radioactively-contaminated water were upgraded to reduce the potential for leakage that could cause contamination of groundwater. This work and the installation of the pool liner were needed to bring the HFBR into conformance with the Suffolk County Sanitary Code. The control room and the operations-level crane were scheduled for reinforcement to ensure that they would survive a design-basis earthquake.

*The Decision to Shut Dow*n

In November 1997, pursuant to Congressional direction, DOE initiated the preparation of an environmental impact statement (EIS) for the HFBR. A draft EIS was published in April 1999. It considered four major alternatives: (i) no action (although the steps described in the two preceding paragraphs would be completed); (ii) resume operation at a power level of 30 MWt or 60 MWt; (iii) resume operation with enhancements that could include replacement of the reactor vessel; and (iv) permanent shutdown with eventual decontamination and decommissioning.

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[15] Unless otherwise noted, information in Section 3.2 is drawn from: BNL, 2000b; DOE, 1999; DOE, 2000b; Lang, 2000; and the BNL website.

[16] DOE, 2000b, page 6.

[17] During a normal refueling of the HFBR, 14 of the reactor's 28 fuel assemblies would be removed and placed in the spent-fuel pool. There are 16 control-rod blades, and it is not clear that any of these blades would be removed during a normal refueling. Thus, the DOE report might be incorrect on this point.

In December 1999 the Secretary of Energy decided that the HFBR would be shut down permanently. The EIS process was terminated. In April 2000, control of the HFBR was transferred to DOE's Office of Environmental Management. A project for stabilization and decontamination and decommissioning (the "HFBR-S/D&DP" project) was initiated, with the objectives: (i) protect human health and the environment; (ii) achieve future land-use options for BNL; (iii) remove or permanently isolate contaminants; (iv) minimize the impacts of transporting and disposing of contaminated media; and (v) conform to federal and state environmental laws.

Stabilization

Following the shutdown decision, experimental apparatus was dismantled and removed from the HFBR. Much of this apparatus was scheduled for re-use at BNL or elsewhere. A volume of 10,900 gallons of primary coolant (heavy water contaminated with tritium) was drained from the reactor and associated systems, and was shipped to the Savannah River Site. The cooling towers and a steel 275,000-gallon cooling-water-holdup tank were removed. Lead shielding, acid and other chemicals were removed.

As of November 2000, planned stabilization actions included: (i) removal of remaining heavy water; (ii) cleanup of the Experimental Floor; (iii) removal of filters and resins from process systems; (iv) removal of beam plugs and their placement in storage; and (v) preparation for layup of the reactor. Available documents do not provide a completely clear picture of planned or achieved actions regarding layup. The following two paragraphs summarize the author's understanding of this issue.

The reactor vessel has, after removal of the heavy-water coolant, been partially re-filled with 1,400 gallons of light water.[18] According to a BNL author:[19] "The light water will provide shielding for personnel working on stabilization activities in the reactor pit area. These activities may include layup of the control rod drives or removal or installation of equipment to facilitate decommissioning. Following completion of these activities, the water will be pumped from the vessel as part of the vessel long-term layup or removal plan." The same author states:[20] "In the proposed layup configuration, the reactor vessel will be drained and dried once the light water is no longer needed for shielding, and the activated internals will remain installed until decommissioning is undertaken. Current plans call for maintaining the system dry in an inert atmosphere to eliminate the hazard of a coolant loss from a vessel or piping failure and to provide a stable environment that minimizes corrosion."

During the Tritium Remediation Project that was conducted in 1997, control-rod blades and other highly-activated reactor-vessel internals might have been removed from the HFBR spent-fuel pool and placed into dry-storage containers, as discussed above. If so,

[18] During normal operation, the reactor vessel contained 2,175 gallons of heavy water, while the total inventory of heavy water in the primary cooling system was 7,073 gallons. See: BNL, 1964, Table 5.2-1.
[19] Lang, 2000, page 6.
[20] Lang, 2000, page 5.

it appears that these items were subsequently returned to the reactor vessel, where they are now located.

Decommissioning Alternatives

BNL is considering nine alternative options for decommissioning the HFBR.[21] These nine alternatives are grouped in four classes, as follows:

I No-Action Alternatives

I-1 Surveillance and Maintenance (S&M): Minimum stabilization activities would be performed as a "holding action". Decommissioning decisions would be deferred.
I-2 Deactivation: As for the S&M alternative except that facility utilities would be turned off and disconnected. The building would be "cold and dark".

II Building Re-Use Alternatives

II-1 Radiological Facility: The reactor vessel would be removed; the building would be cleaned up to an extent that would allow its limited re-use. Minimal S&M activities would continue.
II-2 Industrial Facility: The reactor vessel and most other materials containing significant radioactivity would be removed. The building would be suitable for industrial re-use. Minimal S&M activities would be necessary. This alternative is identified by BNL as the "current planning case".
II-3 Free-Release Facility: The reactor vessel and all other accessible, radioactively-contaminated systems and structures would be removed. The building would be suitable for unlimited re-use. Criteria for "free release" limits of contamination have yet to be negotiated.

III Property Re-Use Alternatives

III-1 Brownfield: The building and its contents would be removed down to the foundation.
III-2 Greenfield: The building, its contents and the foundation would be removed. Unrestricted use of the grounds could occur.

IV Entombment Alternatives

IV-1 Building Remains in Place: The reactor vessel would be filled with grout or some other material. Limited re-use of the building could occur.
IV-2 Building is Removed: The reactor vessel would be entombed within a monolithic structure. The building would be removed. Limited re-use of the grounds could occur.

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[21] BNL, 2000b.

Five of these nine alternatives involve removal of the reactor vessel. This could be done in two ways: as a single pull; or with segmentation before removal. According to BNL, the major factors determining the choice between these two options would relate to transportation and disposal of the vessel. A single pull would allow the vessel to leave the site in one shipment. However, a special shipping cask would have to be designed, built and licensed, and a transport route negotiated. Also, no disposal site has yet agreed to accept the intact vessel. By contrast, vessel segments could be shipped to disposal sites in already-licensed casks, via already-approved transport routes.

3.3 Brookhaven Medical Research Reactor

The BMRR operated until December 2000. The author was informed by BNL personnel, during a site visit in June 2002, that all the fuel at the BMRR would be shipped to the Savannah River Site before the end of 2002. BNL personnel indicated that the transfer of fuel into shipping casks would be done without immersing the fuel in water. Previously, the spent-fuel pool at the HFBR had been used in the fuel-transfer process. BNL personnel further indicated that characterization of the reactor, to determine its radioactive inventory and other parameters relevant to decontamination and decommissioning, would begin after removal of the fuel from the site.

To the author's knowledge, there is no published analysis, plan or discussion of options for decommissioning of the BMRR. Development of such studies would be a task beyond the scope of this report.

4. Inventories of Radioactive Material

4.1 Brookhaven Graphite Research Reactor

The core and radiation shield of any operating nuclear reactor contain radioactive material in the form of neutron-activation products. These are created by neutron bombardment of the materials used in the reactor. After the reactor is shut down and the fuel is removed, activation products are major contributors to the reactor's radioactive content. As expected, activation products are present at the BGRR. In addition, dispersal of radioactive material from BGRR fuel assemblies, especially those used during the period 1950-1958, caused radioactive contamination that remains in the graphite pile, the exhaust-air system, the spent-fuel canal and contaminated soils.[22]

Estimated radioactive inventories at the BGRR are provided in Table 1. For each of a set of radionuclides, this table shows the inventories (in Ci) for three components of Building 702, and for all other components of the BGRR combined. The latter components are

[22] Unless otherwise indicated, information in Section 4.1 is drawn from: BNL, 2000a; BNL, 2000c; BNL, 2001b; BNL, 2001c; BNL, 2002a; BNL, 2002b; BNL, 2003a; and BNL, 2003b.

grouped under the heading, "Balance of Plant".[23] The inventory values in Table 1 are taken from a BNL report published in January 2002.[24] This is the most recent BNL report that provides overall inventories of radioactive material at the BGRR. Data that could be used to update these values are available from characterizations of the BGRR that were completed during 2002 and 2003.[25] During these characterizations, samples were taken that provide concentrations (e.g., in pCi/g) of radioactive material at various locations. The resulting data could be compiled into estimates of overall inventories of radioactive material at the BGRR, thereby updating Table 1. However, BNL has not published such a compilation.

Some information that is available reveals discrepancies between the inventories in Table 1 and those developed from the recent characterizations. Table 1 shows that the graphite pile contains a combined inventory of H-3 and C-14 amounting to about 4,500 Curies, with no inventory of these isotopes in the radiation shield. By contrast, a December 2003 BNL presentation to the Community Advisory Council stated that the graphite pile and shield contain about 8,100 Curies of radioactive material, predominantly H-3 and C-14.[26]

Dispersal of Radioactive Material from Fuel

From its initial operation in August 1950 until April 1958, the BGRR employed fuel assemblies consisting of finned aluminum cans 11 feet long, each can containing 33 natural-uranium slugs. From April 1958 onward, the BGRR employed fuel assemblies containing enriched uranium in the form of aluminum-clad, uranium-aluminum alloy plates. These assemblies were 2 feet long.

Fuel assemblies of the first type were prone to stress-related failures of the aluminum cans while the assemblies were in the reactor. In 28 instances, can failures resulted in oxidation of uranium slugs, with release of fuel fragments, fission products and transuranic radionuclides into the cooling air. In other instances, fission products were released into the cooling air through leaks in cans or by fission of uranium residues on the outside of cans. Fuel elements of the second type were more durable, but there were two instances of cladding failure during the period 1960-1962.

Radioactive material released during these various instances caused contamination of fuel channels, the North and South Plenums, the underground exhaust-air ducts, the underground filter chambers and cooling chambers, the above-ground exhaust-air ducts, and the Fan House (Building 704). Some portion of the radioactive material released from the fuel must have passed through the Fan House and been released to the atmosphere through the stack. Material released through the stack would have been in the form of gases or comparatively small particles. Larger particles or condensing vapors

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[23] For the Balance of Plant, Table 1 shows the amounts of each radionuclide that have been removed during decommissioning (the Removed column), and the amounts that remain in place (the Residual column).

[24] BNL, 2002a.

[25] See, especially: BNL 2003a; and BNL, 2003b.

[26] Petschauer, 2003.

would have settled out or adhered to surfaces before reaching the stack. The largest particles would not have traveled far from their point of origin. For example, it has been presumed that fuel parts containing natural or enriched uranium were deposited in or close to the graphite pile.

Characterization of the graphite pile has revealed the presence of debris, sometimes with high concentrations of radioactivity. For example, a debris sample from the floor of the North Plenum was found to contain Cs-137 at a concentration of 246,000 pCi/g and Co-60 at 9,200 pCi/g.[27] A debris sample from the interior of an experimental channel opening onto the West face of the pile was found to contain Cs-137 at 3.36 million pCi/g and Co-60 at 124 million pCi/g.[28] The profile of beta and gamma radiation readings taken within the fuel channels indicates that a considerable amount of debris exists under the pile. This debris has not been sampled directly.[29]

The fuel assemblies used from 1950 to 1958 were prone to experiencing corrosion of their aluminum cans during storage in the spent-fuel canal, resulting in radioactive contamination of the canal water. Also, as part of preparations for removing this fuel from the BGRR, the cans were chopped under water, the uranium slugs were taken out, and the lengths of empty can were removed from the canal and stored dry in underground vaults. This practice released a substantial amount of radioactive material into the canal water. By contrast, the fuel assemblies used from April 1958 onward experienced comparatively little corrosion while stored in the canal, and did not need to be chopped before they were removed from the canal. Thus, they caused much less contamination of the canal water.

Contamination of Soils

Some of the radioactive material that was dispersed into the cooling system and the canal from BGRR fuel was subsequently carried by water into soils in the vicinity of the BGRR. One pathway was leakage of canal water. Another pathway was leakage of water that accumulated in the underground exhaust-air ducts. Much of this water came from ruptured heat-exchanger tubes in the underground cooling chambers. An especially important pathway for soil contamination was leakage from the Pile Fan Sump. Water reaching the Pile Fan Sump from the Fan House carried radioactive material that had been dispersed into the cooling air from BGRR fuel. Groundwater plumes of Sr-90 in the vicinity of the BGRR show their highest concentrations – up to 500 pCi/liter of Sr-90 according to measurements in 1998 – close to the Pile Fan Sump.[30]

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[27] BNL, 2003a, Volume 1, page 35. An offsite laboratory analysis of this sample yielded concentrations 16-17 percent lower than the values given above. The onsite analysis is cited in order to provide comparability between the two debris samples discussed in this paragraph.
[28] BNL, 2003a, Volume 1, page 94.
[29] BNL, 2003a, Volume 1, pp 88-91.
[30] BNL, 2001b, Figure 5.

4.2 High Flux Beam Reactor

BNL has estimated the radionuclide inventories of various components of the HFBR.[31] These estimates are shown in Table 2. It appears that the primary source for these estimates is a BNL document that calculates the concentrations of neutron-activation products for a given neutron flux and period of exposure.[32] This primary document begins with the statement: “Estimates are made of the long lived radionuclide activities in the HFBR structures. These values are to be used for scoping cost estimates related to decommissioning. Note that the calculations have not been independently checked so they should be used for preliminary estimates only.”

The first inventory column of Table 2 provides estimates of radionuclide inventories in the control-rod blades and the transition plate. These items were inside the reactor vessel during reactor operation. The control-rod blades consist of neutron poisons (europium and dysprosium) in a stainless-steel matrix with a stainless-steel cladding. The transition plate is made from stainless steel. Radionuclides were created within these items by neutron activation. Apparently, these items are currently located inside the reactor vessel, submerged under light water.

No estimates are provided in the second inventory column of Table 2, which accounts for the aluminum portions of the reactor vessel and its internals. However, neutron-activation products, especially Fe-55, are said to be present in these aluminum components. The total inventory of activation products in the reactor's aluminum components is said to be "only ten percent of the total radionuclide inventory".[33]

The third and fourth inventory columns provide estimates of neutron activation products in the thermal and radiation shields. In both instances, the activation products (Fe-55) arise from activation of steel. The thermal shield is made of lead and steel, and the radiation shield contains 60 percent by weight of steel punchings.

The fifth inventory column provides an estimate of the inventory of the neutron activation product Co-60 within the H-6 beam plug. This item is stored in a shielded storage structure (the "cheese box") on the Equipment Level. Other components containing activation products are also stored in this structure. Beam plugs similar to the H-6 plug have been disposed of previously. BNL has considered the removal from the radiation shield of other beam plugs that contain activation products, followed by their storage in the cheese box.

Radionuclide inventories shown in the final column are for materials that have two attributes: (i) they are outside the reactor and its thermal and radiation shield and are therefore designated as "balance-of-plant" materials; and (ii) they are judged to be "at

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31 Lang, 2000.
32 Tichler, 1995.
33 Lang, 2000, page 5.

risk" because they could be dispersed into the environment by credible mechanisms. These materials include 300-500 gallons of heavy water contaminated by tritium (H-3); this water consists of dregs in storage tanks and residues in resin beds, filters and piping. Also, resin beds contain Co-60, and various activated wastes contain Fe-55 and Ni-63.

4.3 Brookhaven Medical Research Reactor

To the author's knowledge, there is no published estimate of the inventories of radioactive material at the BMRR. Preparation of an independent estimate would be a task beyond the scope of this report.

4.4 Comparison with Radioactive Waste Tanks at Hanford

Table 3 provides estimated radionuclide inventories in four radioactive waste tanks at the Hanford Site. These tanks each have a nominal capacity of 55,000 gallons. They now contain thousands of gallons of radioactive liquid and sludge.[34] Some tanks at Hanford contain much larger inventories of radioactive material. Decommissioning of the Hanford tanks poses major technical challenges.

The Hanford tank inventories provide an illustration of the amounts of radioactive material that must be dealt with in a decommissioning situation unlike that at BNL. However, direct comparison of the Hanford tank and BNL reactor inventories does not indicate the comparative risks associated with these facilities. A risk comparison would consider the probabilities for dispersal of radioactive material and the pathways by which the dispersed material would expose people to radiation.

5. Hazard Scenarios

5.1 Brookhaven Graphite Research Reactor

A hazard analysis has been published for the BGRR.[35] However, its scope is limited, as discussed in Section 6, below. The analysis considers the following types of event as potential causes of dispersal of radioactive material: (i) earthquake; (ii) hurricane; (iii) graphite-dust detonation; (iv) loss of ventilation or filtration for the pile negative-pressure system; (v) crane load drop; and (vi) fire. Also considered is the potential exposure of workers to toxic material. Risks are assessed using a methodology that is primarily qualitative. There is no discussion of the potential for dispersal of the loose, radioactive debris that exists in and around the graphite pile.

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[34] Tank B-201 contains 10,000 gallons of liquid and 30,000 gallons of sludge, Tank B-202 contains 8,000 gallons of liquid and 29,000 gallons of sludge, Tank B-203 contains 13,000 gallons of liquid and 51,000 gallons of sludge, and Tank B-204 contains 13,000 gallons of liquid and 50,000 gallons of sludge. Each tank has a nominal capacity of 55,000 gallons. Data were provided to the author by Robert Alvarez, drawing from: Hanlon, 2002.

[35] BNL, 2002a.

5.2 High Flux Beam Reactor

A hazard analysis has been published for the HFBR.[36] Its scope is limited, as discussed in Section 6, below. The analysis considers the following types of event as potential causes of dispersal of radioactive material: (i) earthquake; (ii) hurricane or tornado; (iii) fire; and (iv) explosion. Discussion of the risk of dispersal is cursory and qualitative.

5.3 Brookhaven Medical Research Reactor

To the author's knowledge, there is no published document that identifies or analyzes potential hazard scenarios at the BMRR. Preparation of an independent analysis would be a task beyond the scope of this report.

6. Management of Decommissioning at Brookhaven

Sections 1 through 5 of this report summarize available information about the BNL reactors and their decommissioning. Those sections of the report do not evaluate BNL's work. Here, in Section 6, BNL's management of the decommissioning of the BNL reactors is evaluated. This evaluation represents the author's best judgment, based on a site visit and review of BNL literature.

BNL's management of reactor decommissioning is competent in many respects. There are, however, deficiencies in four areas: (i) documentation; (ii) hazard analysis; (iii) consistency of planning standards and procedures; and (iv) learning from experience. These areas are inter-related, because obtaining the needed improvements in each area will require parallel improvements in the other three areas.

Documentation

The standard for documentation of the BNL reactors was set by reports prepared during the reactors' design phases.[37] In general, these reports are clearly written, internally consistent, accurate and informative. The authors took obvious pride in the engineering effort that went into these reactors.

BNL literature about decommissioning of the BNL reactors does not stand up well by comparison. There are internal inconsistencies and inaccuracies. In some cases, documents are voluminous and repetitive without necessarily being informative. In other cases, important issues are addressed in a cursory fashion. A reader of a BNL document can be left with the impression that the document was written to complete a bureaucratic requirement rather than to perform an incisive analysis or convey information. Decommissioning information on the BNL website can be out of date and inaccurate. There appears to be no complete, publicly-available catalogue of documents related to

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[36] Lang, 2000.
[37] See, for example: BNL, 1949; BNL, 1956; BNL, 1960; and BNL, 1964.

decommissioning of the BNL reactors. Documents can be difficult to obtain when they have been identified.

These deficiencies are not simply matters of public relations. They can be symptomatic of a management approach that is not as cost-effective and productive as it should be. In the context of reactor decommissioning, there is no reason to withhold information from the public for reasons of security or commercial confidentiality. Thus, there should be only one set of documents, serving a dual purpose: (i) providing BNL's organizing framework for decommissioning activities; and (ii) informing interested parties outside BNL. Both purposes require high-quality documentation.

Decommissioning activities are not as sophisticated as reactor design, and therefore do not require documentation with the same degree of detail. However, decommissioning documents should be clear, concise, internally consistent and accurate. Lessons learned from experience (see below) should be clearly spelled out. Each document should have a summary prepared according to a standard format. That summary should be placed on the BNL website when the report is issued, to ensure that the website remains current. Full text of each document should be accessible through the website (e.g., as a PDF file). A complete, current catalogue of decommissioning documents should be available.

Hazard Analysis

Hazard analyses, in a decommissioning context, are available for the BGRR and the HFBR.[38] Both analyses assess risks in a cursory and qualitative manner, as discussed in Section 5. Better methods of risk assessment are available and should be used. Also, the present hazard analyses are too limited in scope. Notably, they do not examine the comparative risks associated with: (i) leaving radioactive material in place; or (ii) removing the material to a disposal site.

If a decommissioning option leaves radioactive material in place, then a hazard analysis for that option should assess risks over the time period required for the radioactivity to decay to a negligible level. The analysis should consider, for example, the long-term potential for contamination of groundwater, and the resulting potential for exposure of people to radiation. BNL has done analyses of this kind for portions of the BGRR, using the RESRAD code.[39]

The comparative risks of leaving radioactive material in place or removing it are central to debates about decommissioning options. These risks should be carefully assessed using tools such as the RESRAD code. If radioactive material is left in place, the risk assessment should address the long-term risks of dispersal of the material and exposure of people to radiation. If radioactive material is transported to a disposal site, the risk assessment should address the short-term risks associated with removal and

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[38] BNL, 2002a; Lang, 2000.
[39] BNL, 2001b, Appendix C.

transportation of radioactive material, and the long-term risks associated with the material's burial at a disposal site.

Consistency of Planning Standards and Procedures

BNL's December 2003 presentation on options for decommissioning of the BGRR reveals inconsistencies in the standards and procedures that BNL uses in its planning and decision-making.[40] As mentioned in Section 3.1, above, BNL published in April 2000 a systematic analysis of decommissioning options for the BGRR. This analysis was ignored in BNL's December 2003 presentation. An impression is left that BNL has recently changed its thinking about decommissioning of the BGRR. This impression is reinforced by other developments.

In a March 2003 report describing the characterization of the graphite pile and related components of the BGRR, BNL explained that the characterization would inform two subsequent documents: (i) a BGRR Risk Assessment; and (ii) a BGRR Feasibility Study.[41] One could reasonably expect that the BGRR Risk Assessment would address, among other matters, the long-term potential for exposure of people to radiation as a result of dispersal of radioactive material from the BGRR. This potential was addressed in an August 2001 BNL report that evaluated options for decommissioning of the BGRR spent-fuel canal and associated items.[42] However, BNL may not proceed with the BGRR Risk Assessment.[43]

As discussed in Section 3.1, above, BNL's "current planning case" for future decommissioning of the BGRR involves Long-Term Institutional Control. In this option, BNL would seal the openings in the radiation shield, spent-fuel canal and air-cooling ducts, making no further effort to remove radioactive material from the central portions of the BGRR. The radioactive material left in place would include loose, radioactive debris in and around the graphite pile. Consideration of this option should be preceded by a thorough assessment of its risks. Yet, BNL may not proceed with a risk assessment. A suspicion arises that BNL, faced with the most demanding phase of its decommissioning of the BGRR, intends to walk away from the task without assessing the risk implications of this action. One assumes that this suspicion is unjustified. If so, BNL should make its intentions clear.

Learning from Experience

In a human enterprise such as decommissioning of the BNL reactors, there will be numerous opportunities to learn from experience. Estimates made during the planning of an activity might not be borne out when the activity is implemented. Unanticipated problems might arise. Improved methods might be developed in the course of an activity.

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40 Petschauer, 2003.
41 BNL, 2003a, Executive Summary.
42 BNL, 2001b, Appendix C.
43 Email to the author from John Carter, BNL, 13 February 2004.

At present, decommissioning programs at BNL lack a systematic process for capturing and using lessons of this kind.

The learning process that is needed would establish a set of indicators for accomplishing decommissioning. Before an activity is implemented, the value of each indicator would be estimated. During and after implementation, actual values of the indicators would be measured. At predetermined intervals, the estimated and measured values would be compared. Implementation of the activity could be adjusted in light of this comparison. If adjustments were made, the reasons would be documented.

Appropriate decommissioning indicators would include:

(i) Physical indicators

- Radionuclide inventories encountered, removed or left in place
- Volumes and masses of contaminated material removed, transported or left in place
- Employee exposure to radiation

(ii) Program indicators

- Timing
- Cost
- Number, type and destination of contaminated-material shipments

Over time, the documented findings of such a learning process would become very useful in planning future decommissioning activities at BNL. These findings would also assist future decommissioning programs within the DOE complex and around the world. In addition, the findings could help to convince the public that decommissioning activities at BNL are well managed.

7. Conclusions and Recommendations

Major conclusions are:

C1. Decommissioning of the BNL reactors is a demanding but achievable task. There is no technical factor preventing the decommissioning of each reactor to a point where subsequent use of the site would be unrestricted. Decommissioning to this extent would require removal from the site of substantial amounts of radioactively-contaminated structural materials and soils.

C2. Decommissioning of the BGRR has proceeded to the point where BNL now faces the most demanding phase of this reactor's decommissioning. This phase involves the removal of radioactive material from the central portions of the reactor, namely the portions in, under and around the graphite pile. Included in this material is loose, radioactive debris from oxidation of uranium fuel slugs during the 1950s.

C3. BNL has conducted a substantial program of sampling and measurement to estimate the radioactive-material concentrations and other characteristics of the central portions of the BGRR. Much of the loose, radioactive debris in the reactor was not accessible within the scope of this program. BNL has not compiled the findings of this program to provide an overall estimate of the remaining radioactive inventory in the BGRR.

C4. BNL, having reached the most demanding phase of decommissioning of the BGRR, has adopted a "current planning case" in which no effort would be made to remove radioactive material from the central portions of the reactor. Instead, BNL would seal the openings in the radiation shield, spent-fuel canal and air-cooling ducts, leaving the radioactive material in place. There are indications that BNL does not intend to assess the risks associated with this course of action.

C5. A limited amount of decommissioning work has been done on the HFBR, and no decommissioning work has been done on the BMRR. Neither reactor has been characterized through a process of sampling and measurement. The radioactive inventory of the HFBR has been estimated, but this estimate does not purport to be definitive. There is, at present, no basis for assessing the risks associated with decommissioning options for the HFBR and the BMRR.

C6. BNL's management of reactor decommissioning is competent in many respects, but there are deficiencies in: (i) documentation; (ii) hazard analysis; (iii) consistency of planning standards and procedures; and (iv) learning from experience. Obtaining the needed improvements in each area will require parallel improvements in the other three areas.

Major recommendations are:

R1. BNL should make its intentions clear regarding future decommissioning actions at the BGRR and the assessment of risks associated with those actions.

R2. BNL should correct the deficiencies summarized in Conclusion C6.

8. Bibliography

(Adey, 2002)
Chuck Adey, Reactor Projects Group Manager, BGRR Working Group, "Overview of Cleanup Activities", slides for a presentation, 24 September 2002.

(BNL, 2003a)
Brookhaven National Laboratory, Brookhaven Graphite Research Reactor Decommissioning Project: Final Draft Characterization Report for the 701 Below-Ground Structures, 702 Pile, and Remaining Soils, BGRR-055, Rev. C (3 Volumes), March 2003.

(BNL, 2003b)
Brookhaven National Laboratory, Brookhaven Graphite Research Reactor (BGRR) Decommissioning Project: Final Draft Characterization Report for Building 701 Above Ground Surfaces, Systems, and Structures, BGRR-054, Rev. A, DRAFT (2 Volumes), May 2003.

(BNL, 2002a)
Brookhaven National Laboratory, Hazard Classification and Auditable Safety Analysis for Brookhaven Graphite Research Reactor (BGRR) Decommissioning Project, BGRR-002, Rev. 4, 10 January 2002.

(BNL, 2002b)
Brookhaven National Laboratory, Brookhaven Graphite Research Reactor Decommissioning Project: Final Completion Report for Removal of the Above-Ground Ducts and Preparation of the Instrument House (708) for Removal, BGRR-039, Rev. 0, 26 April 2002.

(BNL, 2001a)
Brookhaven National Laboratory, Site Environmental Report 2000, September 2001.

(BNL, 2001b)
Brookhaven National Laboratory, Brookhaven Graphite Research Reactor Decommissioning Project: Lower Canal and Water Treatment House, Equipment, and Associated Soils - Engineering Evaluation/Cost Analysis, BGRR-033, Rev. F, 15 August 2001.

(BNL, 2001c)
Brookhaven National Laboratory, Brookhaven Graphite Research Reactor (BGRR) Decommissioning Project: Removal Action Completion Report for Pile Fan Sump, Piping, and Soils Removal, BGRR-024, Rev. 0, 24 January 2001.

(BNL, 2000a)
Brookhaven National Laboratory, Brookhaven Graphite Research Reactor Decommissioning Project: Removal Action Alternatives Study, BGRR-015, Rev. 0, 13 April 2000.

(BNL, 2000b)
Brookhaven National Laboratory, "HFBR Roundtable Summary Report", a summary of presentations and discussion at a public meeting held at BNL on 30 November 2000.

(BNL, 2000c)
Brookhaven National Laboratory, Brookhaven Graphite Research Reactor (BGRR) Decommissioning Project: Activity Closure Report for Primary Air Cooling Fans and Materials Removal, BGRR-020, 28 April 2000.

(BNL, 1994)
Brookhaven National Laboratory, Levels 2 and 3 External Events PRA for the High Flux Beam Reactor, Rev. 1, June 1994.

(BNL, 1964)
Brookhaven National Laboratory, Final Safety Analysis Report on the Brookhaven High Flux Beam Research Reactor, BNL 7661, April 1964.

(BNL, 1960)
Brookhaven National Laboratory, Description of Facilities and Mechanical Components: Medical Research Reactor (MRR), BNL 600 (T-173), February 1960.

(BNL, 1956)
Brookhaven National Laboratory, Medical Research Reactor Hazard Report, May 1956.

(BNL, 1949)
Brookhaven National Laboratory, Book No. 1, Design Manual, Volume No. 3, Pile, 1 April 1949.

(Burns and Roe, 1989)
Burns and Roe Company, Decommissioning Evaluations and Plan for the Brookhaven Graphite Research Reactor, Revision 2, March 1989.

(Carter, 2003)
John Carter, Director of Community Affairs, BNL, letter of 18 March 2003 to IRSS with enclosed slides for a BNL presentation, “BGRR Characterization” (undated).

(DOE, 2000a)
US Department of Energy, Decommissioning Handbook: Procedures and Practices for Decommissioning, DOE/EM-0383, January 2000.

(DOE, 2000b)
US Department of Energy, DOE Safety Evaluation Report (SER): Brookhaven High Flux Beam Reactor Stabilization and Decontamination and Decommissioning Project, 26 September 2000.

(DOE, 1999)
US Department of Energy, Draft Environmental Impact Statement (DEIS) for the High Flux Beam Reactor Transition Project at the Brookhaven National Laboratory, Upton, New York (DOE/EIS-0291), April 1999.

(DOE, 1997)
US Department of Energy, DOE Standard: Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports, DOE-STD-1027-92, issued December 1992, revised September 1997.

(ERDA, 1977)
Energy Research and Development Administration, Final Environmental Impact Statement: Brookhaven National Laboratory, Upton, New York, ERDA-1540, July 1977.

(Hanlon, 2002)
B. M. Hanlon, Waste Tank Summary Report for Month Ending December 31, 2002, HNF-EP-0182, Rev. 177, CH2MHill Hanford Group, Richland, WA.

(Hanson et al, 1994)
A. Hanson, J. Boccio and W.T. Pratt, Levels 2 and 3 External Events PRA for the High Flux Beam Reactor, Rev. 1, Brookhaven National Laboratory, June 1994.

(Lang, 2000)
Paul Lang, Reactor Division, BNL, Reduction in HFBR Hazard Category, 9 August 2000.

(Minns et al, 2000)
John L. Minns et al, Staff Responses to Frequently Asked Questions Concerning Decommissioning of Nuclear Power Reactors, NUREG-1628, US Nuclear Regulatory Commission, June 2000.

(Petschauer, 2003)
Fed Petschauer, “Brookhaven Graphite Research Reactor: Update and Risk-Based End State Vision”, slides for a presentation to the Community Advisory Council, 11 December 2003.

(Tichler, 1995)
P. Tichler, BNL, Estimates of Neutron Activation in the HFBR Reactor Vessel, Vessel Internals and the Thermal and Biological Shield, 25 January 1995.

(Travers, 2002)
William B. Travers, Executive Director for Operations, US Nuclear Regulatory Commission, Memorandum to the Commissioners, "Status of the Entombment Option for Power Reactors, SECY-02-0191", 25 October 2002. Attached to this Memorandum is the paper, "Staff Analysis of Advance Notice of Proposed Rulemaking Comments for Entombment of Power Reactors".

(TWINS, 2003)
Tank Waste Inventory Network System (TWINS), an information system operated for the US Department of Energy by the Pacific Northwest National Laboratory to provide information on the radioactive inventories of waste tanks at the Hanford site, data downloaded by Robert Alvarez on 8 September 2003 and provided to the author.

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Tables 1 through 3 are on the following pages.

Table 1
Estimated Radionuclide Inventories at Brookhaven Graphite Research Reactor

		Radionuclide Inventory (Ci)				
		Building 702			Balance of Plant	
Radionuclide	Half-Life (yr)	Graphite Pile	Control Rods	Radiation Shield	Removed	Residual
H-3	1.23E+01	3.40E+03			1.16E-02	3.22E-02
C-14	5.72E+03	1.05E+03			1.67E-04	5.59E-03
Fe-55	2.70E+00	3.14E-02			1.17E-06	2.04E-05
Co-60	5.27E+00	1.25E+01	3.32E-01		3.06E-03	8.13E-03
Ni-63	1.00E+02	9.73E+01			2.53E-03	8.41E-02
Sr-90	2.88E+01	4.21E-01			4.22E-01	2.97E+00
Y-90	7.31E-03	4.21E-01			4.22E-01	2.97E+00
Tc-99	2.14E+05	4.96E-03			1.25E-04	1.23E-03
I-129	1.57E+07	5.13E-03			3.08E-05	4.07E-04
Cs-137	3.02E+01	1.26E+00	1.75E-03		8.13E-01	1.03E+01
Eu-152	1.30E+01	3.95E-02	8.57E-04		4.68E-04	5.49E-03
Eu-154	8.50E+00	5.66E+00	1.80E-03		2.38E-04	2.66E-03
Eu-155	4.90E+00	3.55E-01	1.14E-02		5.51E-05	3.04E-06
Ra-226	1.60E+03	6.88E-03			5.90E-01	3.67E-03
Th-232	1.41E+10	2.07E-03			5.57E-03	6.55E-05
U-233	1.59E+05				1.08E-01	
U-234	2.45E+05	7.16E-03			5.85E-04	9.54E-04
U-235	7.04E+08	6.93E-04			2.59E-05	3.89E-05
U-238	4.47E+09	6.07E-04			2.75E+00	9.06E-04
Np-237	2.14E+06				1.00E-04	
Pu-238	8.77E+01	4.81E-02			1.58E-04	1.78E-03
Pu-239	2.44E+04	7.61E-02			7.12E-03	1.09E-01
Pu-240	6.57E+03				7.12E-03	
Pu-241	1.44E+01	1.09E-01			9.32E-03	1.56E-01
Am-241	4.33E+02	1.66E-01	2.53E-02		2.41E-02	2.57E-02
Cm-244	1.81E+01				6.93E-09	
Cf-252	2.64E+00				1.93E-02	

SOURCE: BNL, 2002a.

Table 2
Estimated Radionuclide Inventories at Brookhaven High Flux Beam Reactor

		Radionuclide Inventory (Ci)					
Radionuclide	**Half-Life (yr)**	**Control Rods & Transition Plate (stainless steel)**	**Reactor Vessel and Internals (aluminum)**	**Thermal Shield (steel and lead)**	**Radiation Shield (steel and concrete)**	**H-6 Beam Plug**	**Balance-of-Plant "Material at Risk"**
H-3	1.23E+01						2.91E+03
C-14	5.72E+03						
Fe-55	2.70E+00	4.10E+04		<7.20E+05	3.60E+02		<1.00E+00
Co-60	5.27E+00	2.00E+04				2.10E+01	<1.00E+00
Ni-63	1.00E+02	8.19E+03					<1.00E+00
Sr-90	2.88E+01						
Y-90	7.31E-03						
Tc-99	2.14E+05						
I-129	1.57E+07						
Cs-137	3.02E+01						
Eu-152	1.30E+01						
Eu-154	8.50E+00	2.32E+02					
Eu-155	4.90E+00	1.42E+01					
Ra-226	1.60E+03						
Th-232	1.41E+10						
U-233	1.59E+05						
U-234	2.45E+05						
U-235	7.04E+08						
U-238	4.47E+09						
Np-237	2.14E+06						
Pu-238	8.77E+01						
Pu-239	2.44E+04						
Pu-240	6.57E+03						
Pu-241	1.44E+01						
Am-241	4.33E+02						
Cm-244	1.81E+01						
Cf-252	2.64E+00						

SOURCE: Lang, 2000.

Table 3
Estimated Radionuclide Inventories in Some Hanford Radioactive Waste Tanks

		Radionuclide Inventory (Ci)			
Radionuclide	**Half-Life (yr)**	**Tank B-201**	**Tank B-202**	**Tank B-203**	**Tank B-204**
H-3	1.23E+01	2.58E+00			
C-14	5.72E+03				
Fe-55	2.70E+00				
Co-60	5.27E+00	1.10E-01			
Ni-63	1.00E+02	1.00E-02		1.00E-02	1.00E-02
Sr-90	2.88E+01	2.67E+02	4.37E+02	3.44E+02	3.15E+02
Y-90	7.31E-03	2.67E+02	4.37E+02	3.44E+02	3.15E+02
Tc-99	2.14E+05		7.00E-01		
I-129	1.57E+07				
Cs-137	3.02E+01	2.11E+01	1.11E+01	7.65E+00	6.67E+00
Eu-152	1.30E+01				
Eu-154	8.50E+00	2.50E-01			
Eu-155	4.90E+00	1.90E-01	1.00E-02	1.00E-02	1.00E-02
Ra-226	1.60E+03				
Th-232	1.41E+10				
U-233	1.59E+05				
U-234	2.45E+05	1.00E-02	2.00E-02	1.00E-02	1.00E-02
U-235	7.04E+08				
U-238	4.47E+09	1.00E-02	1.00E-02	1.00E-02	1.00E-02
Np-237	2.14E+06				
Pu-238	8.77E+01	9.30E-01	2.80E-01	2.90E-01	3.50E-01
Pu-239	2.44E+04	1.09E+02	1.73E+01	4.46E+01	5.38E+01
Pu-240	6.57E+03	6.27E+00	2.94E+00	3.90E+00	4.72E+00
Pu-241	1.44E+01	1.55E+00	1.25E+01	9.21E+00	1.11E+01
Am-241	4.33E+02	4.13E+00	8.80E+00	4.80E-01	5.80E-01
Cm-244	1.81E+01				
Cf-252	2.64E+00				

SOURCE: TWINS, 2003.