Nuclear Reactor Laboratory
University of Illinois at Urbana-Champaign
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Reactor Administrator: Richard L. Holm

March 28, 2006
Docket No. 50-151

U.S. Nuclear Regulatory Commission
Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: Technical Specification Change and Decommissioning Plan submittal.

Please find attached the following documents requiring NRC approval:

- A request for a change to the facility technical specifications in support of decommissioning, and
- The Decommissioning Plan with supporting documents for the University of Illinois Nuclear Reactor Laboratory.

Upon approval of the Decommissioning Plan and Technical Specification changes we will commence the decommissioning of the facility.

I would also request that upon NRC approval of the Final Status Survey that the facility license be terminated.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 28, 2006.

Sincerely,

Richard L. Holm
Reactor Administrator

c: Alexander Adams, Jr., USNRC
File

A020

UIUC Technical Specifications Changes in Support of Decommissioning
March 2006

Section 3.7 Ventilation System

Due to the potential for airborne radioactivity due to activities during decommissioning the following substantive addition to specification 3.7 is requested:

> *"During decommissioning activities where the potential exists for an airborne release of radioactivity to the environment, appropriate ventilation will be operating in a manner such that the release pathway is via a monitored, and filtered, ventilation system exhaust."*

Justification: This change to the specification will ensure that activities during decommissioning that have the potential for creating an airborne release of radioactivity to the environment will occur in a controlled and monitored manner. The original specification is no longer valid as the reactor is inoperable and fuel removed, and has been deleted.

The page is attached showing the editorial and substantive changes, also attached is a "to be inserted" page.

Section 5.2 Reactor Building

This section is entirely deleted. The objective of this specification was to assure that provisions were made to restrict the release of radioactivity into the environment. In fact, the specification applied to gaseous releases. Section 3.7 Ventilation System has been revised to ensure that potential airborne releases of radioactivity to the environment are thru a controlled and filtered pathway. The specifications listed in Section 5.2 no longer apply:

a. The reactor shall be housed in a closed room of a building designed to restrict leakage.

   This specification existed in support of reactor operation for confinement purposes. The reactor is inoperable and fuel is not present.

b. The minimum free volume of the reactor room shall be 70,000 cubic feet.

   This specification existed to minimize and provide dilution effects in the event of a fuel element failure or other gaseous release. The reactor is inoperable, the fuel is not present and there are no other potentials for gaseous release.

c. The reactor building shall be equipped with a ventilation system capable of exhausting air or other gases from the reactor room from a stack at a minimum of 60 feet above ground level.

   This specification existed to minimize and provide dilution effects in the event of a fuel element failure or other gaseous release. The reactor is inoperable, the fuel is not present and there are no other potentials for gaseous release.

3.7 Ventilation System

Applicability

This specification applies to the operation of the reactor facility ventilation system *and temporary systems utilized during decommissioning activities.*

Comment [rlh1]: Added to cover decommissioning activities.

Objective

The objective is to assure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from ~~reactor operation or during fuel movements~~ *decommissioning activities.*

Comment [rlh2]: Deleted since no longer applicable.

Comment [rlh3]: Added to cover decommissioning activities

Specification

~~The reactor shall not be operated and fuel shall not be moved unless the facility ventilation system is in operation, except for periods of time not to exceed two days to permit repairs to the system. During such periods of repair:~~

~~a. The reactor shall not be operated at power levels above 1 Mw;~~
~~b. The reactor will not be operated in the pulse mode; and~~
~~c. The reactor shall not be operated with experiments in place whose failure could result in the release of radioactive gases or aerosols, and~~
~~d. Fuel shall not be moved.~~

Comment [rlh4]: Deleted since no longer applicable.

During decommissioning activities where the potential exists for an airborne release of radioactivity to the environment, appropriate ventilation will be operating in a manner such that the release pathway is via a monitored, and filtered, ventilation system exhaust.

Comment [rlh5]: Substantive change to support decommissioning activities.

Basis

~~It is shown in Section XIV of the SAR that operation of the ventilation system sufficiently reduces off-site doses to below 10 CFR Part 20 limits in the event of a TRIGA fuel element failure. The specifications governing operation of the reactor while the ventilation system is undergoing repair preclude the likelihood of fuel element failure during such times. It is shown in Section IV of the SAR that, if the reactor were to be operating at a power level of 1 Mw, fuel element failure will not occur, even if all the reactor tank water were to be lost.~~ *This specification ensures that activities during decommissioning that will cause the potential for airborne radioactivity will be performed in such a manner that the airborne release pathway from the building will be through a monitored and filtered ventilation system.*

Comment [rlh6]: Deleted since no longer applicable

Comment [rlh7]: Justification for change.

5.2 ~~Reactor Building~~ *Reactor Building intentionally deleted.*

~~Applicability~~

~~This specification applies to the building which houses the reactor.~~

~~Objective~~

~~The objective is to assure that provisions are made to restrict the amount of release of radioactivity into the environment.~~

~~Specifications~~

~~a. The reactor shall be housed in a closed room of a building designed to restrict leakage.~~

~~b. The minimum free volume of the reactor room shall be 70,000 cubic feet.~~

~~c. The reactor building shall be equipped with a ventilation system capable of exhausting air or other gases from the reactor room from a stack at a minimum of 60 feet above ground level.~~

~~Bases~~

~~In order that the movement of air can be controlled, the building contains no windows that can be opened. The room air is exhausted through absolute filters and discharged through a stack to provide dilution.~~

Comment [rlh8]: This section is no longer applicable. Potential airborne radioactivity releases controlled per changes to section 3.7.

PAGES TO BE INSERTED

3.7 Ventilation System

Applicability

This specification applies to the operation of the reactor facility ventilation system and temporary systems utilized during decommissioning activities.

Objective

The objective is to assure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from decommissioning activities.

Specification

During decommissioning activities where the potential exists for an airborne release of radioactivity to the environment, appropriate ventilation will be operating in a manner such that the release pathway is via a monitored, and filtered, ventilation system exhaust.

Basis

This specification ensures that activities during decommissioning that will cause the potential for airborne radioactivity will be performed in such a manner that the airborne release pathway from the building will be through a monitored and filtered ventilation system.

Section 5.2 Reactor Building intentionally deleted.

UNIVERSITY OF ILLINOIS
AT URBANA-CHAMPAIGN

University of Illinois

Nuclear Reactor Laboratory

Decommissioning Plan

And

Supporting Documents

Docket: 50-151

March 2006

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HISTORICAL SITE ASSESSMENT

NUCLEAR RESEARCH LABORATORY
UNIVERSITY OF ILLINOIS AT CHAMPAIGN-URBANA

Prepared by:

Scientech, LLC
143 West Street
New Milford, CT 06776

June 2005

Project Application	*Prepared By*	*Date*
23566	Kevin Taylor, PE, CHP	6/21/05

APPROVALS:

Title	*Signature*	*Date*
Technical Reviewer	Kenneth M. Kasper, CIH, CHP	6/21/05
D&D Operations Manager	Lee G. Penney	6/21/05

Revision Log

Revision Number	Affected Pages	CRA Number	Approval

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TABLES

APPENDIX

ACRONYMS AND ABBREVIATIONS

ACM	asbestos containing materials
ASTM	American Society for Testing and Materials
Ci	curie
DCGL	Derived concentration guideline level
dpm/100cm^2	disintegrations per minute per 100 square centimeters
FSS	Final status survey
g/cm^3	grams per cubic centimeter
HEPA	High-Efficiency Particulate Air
HSA	Historical site assessment
ID	inner diameter
kg/m^3	kilograms per cubic meter
kW	kilowatt
lb/ft^3	pounds per cubic foot
LOPRA	Low Power Reactor Assembly
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
mCi	millicurie
NRC	U.S. Nuclear Regulatory Commission
NRL	Nuclear Research Laboratory
OD	outer diameter
Scientech	Scientech, LLC
uCi	microcurie
University	University of Illinois at Champaign-Urbana

1.0 EXECUTIVE SUMMARY

The University of Illinois at Champaign-Urbana (University) discontinued operations of its Advanced TRIGA nuclear research reactor at its Nuclear Research Laboratory (NRL) in 1998. Prior to shutdown of the reactor and the removal of the reactor fuel elements in 2004, the University has decided to decommission and dismantle the NRL and terminate U.S. Nuclear Regulatory Commission (NRC) Facility Operating License No. R-115. Prior to license termination, the NRL facility will be completely dismantled and components will be disposed of according to their radiological condition. As such, the final status survey (FSS) for license termination is expected to include only the site location and none of the facility structure or components.

The Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) is the current regulatory guidance document for characterizing and surveying radiologically impacted sites. Per the guidance in the MARSSIM, the initial phase of the NRL site characterization and eventual decommissioning is the Historical Site Assessment (HSA). The NRL HSA, its methods, and its findings are described in this document.

The results of the HSA suggest that much of the NRL and its components are known to be or are potentially impacted by radioactive contamination or activation. Many of the aluminum reactor components will be activated along with a portion of the heavy concrete bioshield. The most highly activated component is expected to be the Lazy Susan (specimen rotary rack) because of its proximity to the core and because it contains Stellite and stainless steel components. Activated components will also include the beam ports and thermal column components, including some of the graphite. Known contaminated materials include equipment that was used to manage and transfer specimen containers that were irradiated in the reactor core or the Lazy Susan. Other contaminated materials include concrete that were exposed to reactor primary coolant water and in which tritium may have built up over the years. Tritium surveys were not part of the NRL's routine contamination surveys. Routine removable contamination surveys do show, however, that there is no wide-spread contamination of other higher-energy beta or alpha-emitting radionuclides which would likely be present with tritium contamination.

The distinction of impacted and non-impacted structures, systems, and components is not of great importance to the planning of the FSS because the FSS will be conducted following facility dismantlement. The results of the HSA will be important in designing the next phase of the site decommissioning, the site characterization.

The site characterization is expected to take place in July 2005. The information obtained during the site characterization phase will be used to prepare the facility Decommissioning Plan and provide estimates of the volumes of contaminated, activated, and clean waste.

2.0 PURPOSE OF HISTORICAL SITE ASSESSMENT

The HSA was conducted as part of the site assessment and characterization phase leading to the decommissioning of the NRL and NRC License No. R-115. The HSA process, as described in the MARSSIM, is designed to:

- Identify potential, like, or known sources of radioactive material or radioactive contamination or activation based on existing or derived information,
- Provide an assessment of the likelihood of contamination migration,
- Provide information useful for the design and planning of the characterization surveys, and
- Provide initial classification of the areas of the NRL as impacted or non-impacted.

To meet these goals, several actions were performed as described in the following sections of this document. Section 3 describes the physical setting, structure, and components of the NRL as described in historical documents and as observed during site visits. Section 4 describes the HSA methodology and details the information gathering process. Section 5 describes the site history and current use of the facility and surrounding areas. Section 6 presents the findings of the HSA.

The HSA was also used to identify the project planning team and other stakeholders in the reactor decommissioning project. Currently, the project planning team consists of Mr. Richard Holm, Reactor Administrator and Assistant to the Dean of the College of Engineering, and the site characterization and planning contractor, Scientech, LLC (Scientech). The Scientech personnel responsible for the HSA, site characterization, and development of the facility Decommissioning Plan are Lee Penney, Scientech Decommissioning Department Manager, Mr. Kevin Taylor, Scientech Senior Health Physicist and Project Manager, and Kenneth Kasper, Scientech Chief Technical Officer and Corporate Radiation Safety Officer. Mr. Holm is supported by the University Reactor Committee on an as needed basis. The members of the Reactor Committee are as follows:

- Mr. Rich Holm, Reactor Administrator
- Mr. Dan Hang, Professor Emeritus of Nuclear Engineering
- Mr. David Scherer, Campus Radiation Safety Officer
- Dr. Barkley Jones, Chair, Professor of Nuclear Engineering
- Mr. Mark Kaczor, Reactor Health Physicist
- Dr. Jonathan Nadler, Adjunct Assistant Professor of Nuclear Engineering and former licensed Senior Reactor Operator on the facility

Mr. Holm has the responsibility of communicating with all other stakeholders. Other stakeholders in the NRL decommissioning project include:

- U.S. Nuclear Regulatory Commission – Mr. Alexander Adams, Jr., Project Manager
- Illinois Department of Nuclear Safety
- The City of Urbana, Illinois
- University Administration
- College of Engineering

3.0 PROPERTY IDENTIFICATION

The following sections provide the physical and environmental setting of the University of Illinois NRL. The information provided will be used to prepare the Conceptual Site Model for development of the derived concentration guideline levels (DCGL) for site release. The DCGLs will be presented in the facility Decommissioning Plan. A description of the NRL building and its interior features and components are also provided.

3.1 PHYSICAL SETTING

3.1.1 Site Location

The NRL is located in the City of Urbana, Illinois on the campus of the University of Illinois at Champaign-Urbana. Urbana is located about 110 miles south-west of Lake Michigan and about 35 miles from the Illinois-Indiana border. Urbana is approximately in the center of Champaign County, which covers an area of about 990 square miles in the east-central section of Illinois.

The University of Illinois campus is located in the adjoining cities of Urbana and Champaign and is centered on the dividing line of these cities.

The NRL building is located in the northeastern portion of campus between Springfield Street to the north, Green Street to the south, Goodwin Street to the west, and Gregory Street to the east. The building is surrounded by research buildings associated with the University's College of Engineering. The most prominent buildings within 800 feet of the NRL building are the Materials Research Laboratory, Loomis Laboratory, Coordinated Science Laboratory, and Ceramics Building. Beyond the immediate surroundings, structures include University dormitories and private dwellings (homes and apartments).

3.1.2 Geology

The bedrock formations in east-central Illinois are layers of sandstone, limestone, dolomite, shale, and coal arranged one above the other. The rock layers, now firm and compact, were originally deposited as unconsolidated sediments in costal marshes or in shallow seas, repeatedly invaded the continent. These sedimentary layers hardened into rock and were later warped and tilted so that today they are no longer horizontal.

3.1.3 Topography and Drainage

The terrain in the vicinity of Champaign-Urbana varies from generally flat to very gently rolling. The soil is primarily silty loam, though in some areas it is silty clay. The general flow of runoff is to the south to Boneyard Creek. Boneyard Creek is located about 300 feet south of the NRL. Boneyard Creek drains to the east to the Salt Fork and Vermillion Rivers. The immediate location surrounding the NRL building, mostly paved areas, slopes to the south and surface runoff is collected by several grated storm sewer openings.

The storm sewers of the University and the City of Urbana flow into Boneyard Creek. The surface runoff runs directly into the storm sewer system via grated openings to the southeast of the building. The building roof drains tie directly into the storm sewer system near the southwestern corner of the building and near the southeastern corner of the heat exchanger room roof. Utility drawings indicate an 8" storm sewer line running north to south several feet from the western side of the building. Two 24" sanitary sewer lines run west to east, one under the northern half of the building and the other several feet north of the building under the cooling towers.

3.1.4 Water Supplies

In Champaign County the fill of the Mohomet Valley contains sand and gravel deposits suitable for development of high-capacity wells. The water for the towns of Champaign and Urbana is obtained from wells that penetrate into the Illinois drift and in some cases into the deeper Kansas drift. The bulk of the local water supply is provided by 4 to 6 wells that are approximately 300 feet deep. Other local supply wells are 120 to 150 feet deep.

3.2 NRL STRUCTURE

The NRL building is a steel frame concrete block building that is approximately 80 feet east-west by 45 feet north-south. The building is supported by 30 metal-shells, cast in place concrete piles with minimum lengths of 40.5 feet. A 6.5-foot deep by 1-foot wide concrete footing, which is laid on concrete pile caps, supports the walls.

The Mechanical Equipment Room containing the heat exchanger and the primary and secondary cooling system pumps is located off the south side of the building and is accessed through a door on the reactor room floor. The exterior dimensions of the room are approximately 18 feet by 22 feet. The exterior height is less than 6 feet above grade. Located below the heat exchanger room is a vault containing two N-16 delay tanks.

The interior of the building contains three levels: the mezzanine or main level, the storage level (located above the mezzanine), and the lower level (reactor room). The mezzanine floor is 10 feet above the reactor room floor and the storage floor is 21 feet above the reactor room floor. The clearance from the reactor room floor to the roof supports is 35 feet. The reactor room is about 44 feet wide by 57 feet long. The mezzanine floor, storage floor, and roof are placed on standard bar joists which are tied to horizontal I-beams and the main support columns.

The reactor room floor is a six inch concrete slab laid on undisturbed earth. The mezzanine and storage floors are 2.5-inch concrete slabs poured on corrugated steel plate. The roof is composed of gravel on a gypsum roof deck which is covered with four-ply asphalt paper. A special concrete base is used to support the reactor and the thermal column door railway. Ten piles similar to those used to carry the building are used to support this special base.

3.2.1 Concrete

There are primarily two different types of concrete used in the NRL facility: (1) Class A (light) concrete and (2) heavy-weight concrete.

Class A concrete aggregates conformed to American Society for Testing and Materials (ASTM) designation C 33-57 and contained both fine and course aggregates. The density of the Class A concrete is 146 pounds per cubic foot (lb/ft^3) or 2,330 kilograms per cubic meter (kg/m^3). Class A concrete was used as mass concrete in floors on all levels, reactor floor walls, and building footings.

Heavy weight concrete is composed of both fine and course aggregates of magnetite ore. The density of the heavy weight concrete is 220 lb/ft^3 or 3,530 kg/m^3. Heavy weight concrete is used as mass concrete in the reactor bioshield, trenches, the 5,000-gallon waste water tank pit, and the concrete thermal column door.

All cement for mixing concrete conforms to ASTM Designation C 150-56 for Portland cement.

Reinforcement bars for concrete conformed to ASTM Designation A 15-57T for billet-steel bars. Deformed bars that conformed with ASTM Designation A 305-56T were also used.

3.2.2 Steel

Steel materials in the NRL include sheet steel plate and structural steel that conform to ASTM Designation A 7-56T. Steel bolts conform to Standard B-18.2-1955 of the American Standards Association. Steel pipe is Grade B, seamless, galvanized (zinc) and conform to ASTM Designation A 120-57T.

Some other steel products include the shadow shields imbedded in the bioshield, beam ports, grating bars, metal strips, anchors, hold-down bolts, and the concrete door frame. Pipe rails are fabricated of 1.25-inch black iron pipe. The concrete door rollers are cast iron.

3.2.3 Other Construction Materials and Components

Other construction materials and component materials include aluminum (type 6061-T6 wrought allow), boral sheets, cadmium plating, lead, graphite, concrete block, and wood.

Aluminum components include the reactor tank, reactor components (such as the grid plates, reflector assembly housing, reactor specimen rack, thermal column housing, beam port liners, and safety plate), studs, bolts, nuts, washers, and pipes. The aluminum reactor tank, the largest aluminum component, is 0.25-inches thick with an outer diameter of 6.5 feet and 20.5 feet deep. The density of the aluminum is 2.72 grams per cubic centimeter (g/cm^3). The large thermal column and the small thermal column of the bulk shielding tank are constructed with a welded aluminum can of 0.5-inch thick aluminum. The metallic of the aluminum alloy required in the original construction specifications is provided below. Aluminum components installed during the 1968 upgrade may not have the same composition, but the composition should be similar.

Silicon	0.40 – 0.80%
Iron	0.15 – 0.40%
Copper	0.15 – 0.40%
Manganese	0.15%
Magnesium	0.8 – 1.2%
Chromium	0.15 – 0.35%
Nickel	0
Zinc	0.25%
Titanium	0.15%
Others	0.5% each

Boral is a mixture of boron carbide (B_4C) and aluminum powder compacted in an aluminum frame, clad with aluminum sheets, and hot rolled to a final thickness of 0.125 inches. The composition by weight is 15.7% boron, 4.3% carbon, and 80.0% aluminum. It has a density of 2.5 g/cm^3 or 1.67 pounds per square foot for 0.125-inch-thick sheet. Boral components include the 46-inch by 49-inch boral curtain of the large thermal column and the lining of the aluminum thermal column ports (large and small thermal columns).

[redacted]

- Room 102 – Office and Reactor Control Room
- Room 103 – Toilet
- Room 104 – Janitor's Closet
- Room 105 – Toilet – southwestern corner

From the main level, two flights of stairs can be taken down to the reactor floor level. One set of stairs is located to the east of the northern door. The second set is located on the south side of Room 102 near the southern roll-up door. The area of the reactor floor beneath the mezzanine floor contains the mechanical and electrical equipment rooms, a clean-up sink, a decontamination shower, and a "cold' fuel storage room.

3.3.1 Ventilation and Air Conditioning

The entire building is exhausted through the building ventilation system that is located on the storage floor. The system contained a 5-micron pre-filter section and a special 0.3 micron high-efficiency particulate air (HEPA) filter section. An activated charcoal filter is located down stream from these filters in a duct parallel with the main exhaust duct. A damper in the duct containing the charcoal filter is normally closed. In case of a fission product release containing iodine isotopes, the damper in the main exhaust duct was closed automatically and the damper in the duct containing the charcoal filter would be opened, thus forcing the air through the activated charcoal filter.

The blower, which is located on the roof, is capable of changing the building's air in about one hour. The exhaust is discharged through a 15-foot stack at a flow rate of 2,400 cubic feet per minute to a height of about 60 feet above grade. When the facility doors are closed, all incoming air is diverted through an air conditioning system that is used to cool and heat the air as needed. The air conditioning unit is positioned on an elevated platform in the northeastern corner of the facility.

3.3.2 Source and "Hot" Fuel Storage and Beam Catchers

Four source and "hot" fuel storage pits are located in the floor of the truck bay. Each of these pits is 1-foot in diameter, 8 feet deep, and lined with 3/8" steel plate. The pit covers are 1-foot thick concrete plugs.

Four hollow concrete pipe beam catchers are cast into the footing of the building across from the original reactor beam ports. These beam catchers have a 2-foot inner diameter and are set 6 feet into the earth outside the footing. The concrete pipes are 2 to 3 inches thick.

3.3.3 Liquid Waste Disposal

On the mezzanine floor, the two restrooms and the janitor's closet are the only rooms requiring drainage. Vent pipes are provided for this system which drains directly into an existing 24-inch sanitary sewer line. Prior to May 1996, four floor drains, one trench drain, the clean-up sink, and the decontamination shower on the reactor floor level all drained into a 500-gallon retention tank located below the floor on the western side of the reactor floor. Waste in the tank would be sampled prior to being discharged directly into the existing 24-inch sanitary sewer line. Besides clean-up water, any radioactive solutions from experiments were put into this system.

Most recently, waste water was collected in the retention rank. When the tank was full it was pumped over to a polyethylene holdup tank located on the reactor room floor. The water passed through a course and a fine filter assembly on route to the holdup tank where it was then sampled. The water was discharged from the holdup tank into the sanitary sewer system when the soluble activity results were satisfactory and it was verified that no insoluble activity was present. If insoluble activity was detected before the discharge then the contents of the holdup tank were recirculated through a 0.4 micron process filter until the insoluble activity was removed and it was verified that no insoluble activity was present.

3.3.4 Gas Discharge

To dispose of any argon-41 that may have formed during reactor operations, an off-gas piping system was provided for the beam port tubes, the thermal column, and the thermalizing column of the bulk shielding facility. In each case, the outer end or surface of the experimental facility is connected by small-diameter piping to an off-gas manifold on the western side of the shielding structure. This manifold is connected to the normal building exhaust system via a 3-inch vent line and discharges through a 0.3 micron filter section at a point just past the HEPA filter section of the ventilation system. The vent from the waste water retention tank is also tied into the reactor vent line.

3.4 TRIGA REACTOR COMPONENTS

The TRIGA reactor is located at the bottom of the reactor tank that is contained inside the concrete bioshield. The open-top aluminum reactor tank is 0.25-inches thick with an outer diameter of 6.5 feet and 20.5 feet deep. The reactor tank was installed during NRL facility construction in 1959. The current reactor core assembly was installed in 1968 to replace the original reactor assembly.

A bridge extends above the reactor tank for positioning and operating the control rods, fuel rods and experimental components. The bridge consists of two parallel 8-inch steel channels 98.25 inches long spaced about 17 inches apart. The steel channels are fastened together with five 0.675-inch thick steel cover plates which form a continuous deck. The bridge is designed to hold a 3.5-ton shielded cask placed over the specimen removal tube.

The reactor bioshield is constructed of heavy concrete and is 22 feet high measured from the reactor room floor to the top of the reactor platform. The lower section of the bioshield is 12 feet high and at least 7.5 feet thick. The upper section of the bioshield is 3 feet thick. Under the bioshield, the concrete is about 2 feet thick below the reactor room floor.

The core assembly currently consists of core support (plenum), the control rods safety plate, the reflector, and grid plates. The core support is a cylindrical aluminum structure bolted to a platform on the bottom of the reactor tank. The core support is 31.5 inches high and 30 inches in diameter. The aluminum control rod safety plate is located below the core support. Surrounding the core support is the aluminum primary coolant return spray header.

The reflector surrounding the core consists primarily of a circular block of graphite with a hexagonal center cutout. The graphite reflector block has an outer diameter of 41.675 inches with a radial wall thickness of 8.6 to 10.3 inches and a height of 22 inches. A 2.25-inch-thick annulus of lead surrounds the graphite. The lead shield is cut out at the beam port penetrations. The reflector and lead assembly is sealed in a welded aluminum container. The reflector and core support are bolted together with stainless steel bolts.

There are two aluminum grid plates in the core assembly. The top grid plate is 1 inch thick and the bottom grid plate is 1.25 inches thick. The top grid plate has three removable sections. The grid plates are about 2 feet in diameter.

Several other reactor components and pieces of support equipment are still in the reactor tank. These include two control rods, five ion chambers, one fission chamber, and submersible lights. The Adjustable Transient Control Rod and the Fast Transient Control Rod are the two control rods that did not contain a fuel trailer like the other control rods. These two rods are described in Table 3-1. The controls rods with the fuel trailers were removed with the reactor fuel.

TABLE 3-1
TRANSIENT CONTROL RODS

	Adjustable	Fast
Exterior material	1.25" OD anodized aluminum	1.25" OD anodized aluminum
Poison Section	15" borated graphite	30" borated graphite
Air follower	20"	20"
Guide tube	1.37" ID aluminum; 5' 4" long	1.37" ID aluminum; 8' 9" long

The drive assemblies for the control rods are fastened to a mounting plate located on the bridge. The standard control rods have electrically-driven rack-and-pinion drives; the adjustable transient rod has a pneumatic-electromechanical drive; and the fast transient drive is pneumatically driven. An electrical connector is provided within the standard drive assembly to permit the easy disconnection and removal of the motor potentiometer, magnet, and the major moving part of the drive.

3.5 EXPERIMENTAL SYSTEMS

Experiments with the TRIGA reactor were carried out primarily in conjunction with the following facilities:

- Isotope-production facility
- Pneumatic transfer system
- Central thimble
- Beam port facilities
- Thermal column
- Thermalizing column (small thermal column) and bulk shielding tank

3.5.1 Isotope-Production Facility

The isotope production facility consists primarily of five components:

1. The rotary specimen rack (Lazy Susan).
2. The specimen – removal tube assembly.
3. The tube-and – shaft assembly that connects the rotary specimen rack with the drive-and – locator assembly.
4. The drive-and-locator assembly, which is mounted on the center channel at the top of the tank.
5. The specimen-filling device, which is used for the insertion and removal of the specimen containers.

The Lazy Susan consists of an aluminum rack for holding specimens during irradiation. The rack is located in a seal-welded aluminum housing attached to the core reflector assembly, by captive screws. The rack can be rotated on stainless steel ball bearing assembly consisting of Stellite balls, type 304 stainless steel races, type 302 stainless steel spring-type spacers.

3.5.2 Pneumatic Transfer System and Central Thimble

Two pneumatic transfer systems were used for the fast insertion and removal of specimens to be irradiated. One of the systems is completely contained within the reactor building. This system consists of a blower, filter, solenoid valves, air tubes, a control box, and specimen tube (in the reactor pool).

The second transfer system terminates in Room 222 in the Material Research Laboratory (a building adjacent to the NRL). This system also consists of a blower, filter, solenoid valves, air tubes, a control box, and specimen tube (in the reactor pool). The transfer tube from the NRL to the Material Research Laboratory is in an underground pipe chase.

The central thimble experimental facility is located at the center of the core. An access tube, consisting of a 20-foot long watertight upper section and a 30-inch long lower section which is perforated, is utilized to place samples in the core center. The upper section of the access tube is bent and the top of the tube is capped and locked.

3.5.3 Beam Ports and Thermal Column

There are 9 beam ports that penetrate the bioshield at the reactor core level. Three ports are associated with the thermal column. The inner portions of the other six beam ports are 6 inches in diameter and 8 inches in diameter on the outer section. One beam port goes completely through the bioshield tangential to the core and has opening on both the north and south ends. The west beam port extends through the reflector to the core surface. The beam ports contain shielding plugs of various construction containing borated concrete, steel, lead, and in some cases wood.

The thermal column, which penetrates the northeastern bioshield and core reflector assembly, is 4 feet square by 5.5 feet in length. The outer thermal column is covered by the 3.5-foot thick heavy concrete door. The graphite in the thermal column is stacked block contained in a welded aluminum can, fabricated of 0.5-inch thick sheets of aluminum and lined with 0.125-inch-thick boral sheets. A 4-inch thick block internal lead shield (stacked bricks) is positioned about 50 inches from the outside surface of the column. Approximately 3 feet from the outside surface is 46-inch by 49-inch boral curtain fabricated from a single 0.25-inch thick boral plate.

3.5.4 Bulk Shielding Tank and Thermalizing Column

The bulk shielding experimental tank is 12 feet deep, 8 feet wide, and 9 feet long. The south wall of the tank is pour heavy concrete about 1 foot thick. The floor of the tank is about 2 feet thick. The east and west walls are part of the reactor bioshield and have varying thickness from about 1 foot to more than 4 feet. The north wall, which is adjacent to the reactor and is also part of the bioshield, is 3 feet thick.

A smaller thermal column (thermalizing column), 2-feet square and 4.33 feet long, extends from the north wall of the bulk shielding tank to the core reflector assembly. The column contains a welded aluminum can, fabricated of 0.5-inch thick sheets of aluminum and lined with 0.125-inch-thick boral sheets. A 0.675-inch thick aluminum cover plate separates the bulk shielding tank and the thermal column.

At the inner end (nearest to the core) the column is filled with graphite blocks to an axial thickness of 8 inches followed by two inches of lead. Following the lead is an 18-inch air space and 24 more inches of stacked graphite block extending to the bulk shielding tank.

The bulk shielding tank was also used for wet fuel storage from 1998 to 2004.

3.6 COOLING SYSTEM

Since 1969, NRL TRIGA reactor used a forced circulation cooling system that included primary and secondary coolant loops. The primary and secondary coolant loops are described in the following sections.

3.6.1 Primary Coolant System

The primary coolant system used the water from the reactor tank as the primary coolant fluid. The water would flow through the reactor core and out the plenum at the bottom of the tank and into a 10-inch pipe. From there, the water would flow first to a 3,000-gallon delay tank and then directly into a second 1,500-gallon delay tank. The water travels from the reactor tank to the delay tanks through about 19 feet of 10-inch diameter schedule 40 6061-T6 aluminum pipe. The pipe runs through a tunnel located under the reactor room floor. The delay tanks are located in a 10-foot by 21-foot vault located under the Mechanical Equipment Room. The vault has 1-foot thick concrete walls. The tunnel and the vault have water collection sumps.

From the second delay tank, the primary coolant water is pumped via the primary pump to the heat exchanger (both located in the Mechanical Equipment Room). It is then returned to the reactor tank via the primary coolant return spray header through 10-inch aluminum piping in the tunnel.

The total equivalent of straight pipe in the primary coolant loop (including elbows and valves) is 394 feet.

3.6.2 Secondary Coolant System

The secondary coolant system is a closed loop system that consists of the cooling towers located outside on the north side of the building and the secondary pump and the shell side of the heat exchanger located in the Mechanical Equipment Room. Eight-inch diameter schedule 40 carbon steel pipes are used for the secondary system. The two cooling towers are Marley Series HCT Steel Double –Flow towers constructed of hot-dipped galvanized steel.

4.0 HSA APPROACH AND METHODOLOGY

The primary components of the HSA are described in the following sections. These include a description of the documents reviewed, details of the site inspections, and the information on interviews conducted with personnel familiar with the NRL operations.

The HSA was designed to gather specific information on:

- Site setting
- Description of NRL facility
- Description of major NRL components
- History of spills, leaks, or releases of radioactive materials
- Current locations of radioactive materials
- Likely and potential contaminated areas
- Likely activated materials
- Likely non-impacted areas

4.1 DOCUMENTS REVIEWED

Many documents were reviewed as part of the HSA. These included Reactor Committee meeting minutes from 1960 through 1978, Reactor Annual Reports from 1969 through 2004, hazard and safety analysis documents, technical specification documents, and various drawings and figures. Many files containing technical information on facility components, such as manufacturers manuals for support equipment were also reviewed. A list of the documents and files reviewed are provided in Appendix A.

The most informative documents included the 1959 Hazard Analysis Report and the 1967 Safety Analysis Report. These documents provided information on the original construction and 1968 reactor modifications and described the reactor and support systems. The Reactor Annual Reports and Reactor Committee meeting minutes provided information on unusual occurrences, experiments that could have released radioactive materials, routine contamination levels throughout the facility.

4.2 SITE INSPECTION

Two site inspections were conducted as part of the HSA. The first inspection consisted of a facility tour provided to Mr. Lee Penney and Mr. Kevin Taylor of Scientech by Mr. Rich Holm on May 17, 2005. Mr. Taylor returned to the facility on May 31 and was present on site to review documents, conduct interviews, and inspect the facility through June 3.

Visual observations during the site inspections identified several areas of concern for possible radioactive contamination. Many of these areas are listed in Section 6.2. The site inspections also allowed for the identification of potential pathways for contamination outside the facility.

4.3 PERSONAL INTERVIEWS

Formal interviews were conducted by Mr. Taylor with Mr. Dan Hang and Dr. Barclay Jones on June 2 and Mr. Mark Kaczor on June 3, 2005. Mr. Holm was also available during the site visit to answer questions as they arose and he was present during all interviews. Each of those interviewed is on the current Reactor Committee and has a long history at the NRL. Mr. Hang and Dr. Barkley have been involved with the NRL since its original construction, through the modifications, and on through the operating life of the reactor. Mr. Kaczor has been the Reactor Health Physicist since 1990. Mr. Holm became Reactor Supervisor in December 1991. He was previously a Staff Senior Reactor Operator for 2 years.

The interviews focused on trying to gain knowledge of information about possible contamination or activation that may have occurred at the NRL. For example, it was asked if there were any experiments that were stopped due to contamination or activation issues. Questions were also directed to gather information about the construction of the facility that may be important to the dismantlement and demolition of the reactor. For example, it was asked how the reactor vessel was fixed inside the reactor tank, which was not apparent from facility drawings.

Interview discussions also focused on radioactive materials that are currently present in the facility and what the future of these materials may be. Items discussed were natural uranium plugs, an antimony-beryllium neutron source, a possible radium-beryllium neutron source, and activated lead bricks. These items will all likely be in place during the site characterization.

While tritium contamination in the soil has not been confirmed at the NRL, tritium contamination at other reactor facilities has been a problem. Potential pathways for tritium contamination to the soil underlying and surrounding the NRL were discussed. Potential pathways identified included the air conditioning condensate discharge and seeps or cracks in the concrete floor of the tunnel and other subsurface locations.

Mr. Kaczor's interview also included a facility walkthrough where he identified the known locations of radioactive materials. These materials included radioactive check sources, contaminated equipment, waste materials, and activated materials. These items are described in Sections 6.2 and 6.5.

No revelations of undocumented contamination or other radiological issues came to light during the interviews conducted as part of this HSA.

5.0 HISTORY AND CURRENT USAGE

The following sections present a brief description of the more than 35-year history of the NRL and the TRIGA research reactor. A description of the current facility is also provided along with a description of the surrounding property.

5.1 HISTORY OF THE NRL

Construction began on the NRL in the summer of 1959 to house a TRIGA Mark II training and research nuclear reactor that was later built by the General Atomic Division of General Dynamics Corporation. The construction of the building and installation of the reactor was performed by the University. By 1960, the walls of the NRL were complete and the foundation for the reactor and bioshield and the thermal column trench were complete. By the spring of 1960, the reactor and reactor tank were installed along with the beam ports and the forms for bioshield concrete. The NRL was completed in the summer of 1960 and the reactor first went critical on August 16, 1960.

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

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

Additional modifications that were made during the upgrade were the installation of additional beam ports, an underground vault to house the heat exchanger and nitrogen-16 delay tanks, and a tunnel to shield the primary water before it reaches the delay tanks.

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

The bulk shielding tank, located on the south side of the reactor allowed for neutron beam experiments to be conducted underwater for additional shielding. The bulk shielding tank was also the home of the Low Power Reactor Assembly (LOPRA) which was a subcritical assembly that used TRIGA fuel. The LOPRA operated under its own NRC license (No. R-117) beginning in 1971. The R-117 license governed the use of the LOPRA until 1995 when the fuel and subcritical assembly were transferred to the NRL's current R-115 license.

On August 6, 1998, nearly 30 years since its initial start-up, the NRL TRIGA reactor was shut down permanently. In 1999, the reactor was officially placed in a SAFSTOR condition while waiting for arrangements to be made to remove and ship the reactor fuel. The bulk shielding tank was used for wet storage of the fuel following shutdown. On August 18, 2004, the reactor fuel was removed and shipped to the U.S. Department of Energy's Idaho National Environmental and Engineering Laboratory.

5.2 CURRENT USE OF THE FACILITY

Since the reactor was shut down in 1998, the facility has been used primarily as a storage facility for NRL equipment. No experimental activities have taken place in the NRL since reactor shutdown. The most significant event to occur since shutdown was the transfer of the reactor fuel from the reactor core, to the bulk storage tank for wet storage in 1998 and then off-site shipment in August 2004.

Stored equipment includes all reactor components (excluding the fuel and the dummy elements), experimental equipment, records and files, office equipment, mechanical equipment, analytical and sampling equipment, radiation survey instruments, and other operable and inoperable equipment. Plans are to salvage and reuse some of the equipment outside the NRL; however, much of the equipment has little value.

There are also some radioactive sources and materials stored in the reactor facility. These are described in detail in Sections 6.2 and 6.5.

5.3 ADJACENT LAND USAGE

The areas outside the NRL are open to public access. Facilities adjacent to the NRL are academic facilities that include class rooms, administrative offices, and laboratories. To the southeast of the facility is a permit-required parking area. The nearest residences, apartments that house mostly University students, are about 300 feet directly north of the facility. It is extremely unlikely that any adjacent or nearby facilities were affected by or contributed to the physical or radiological condition of the NRL facility.

6.0 FINDINGS

The following sections describe the findings of the HSA. Details are provided on the possible contaminated materials and activation products. Activation is caused by neutrons changing the elemental nature of materials such as the transformation of stable cobalt-59, present as an impurity in carbon and stainless steel and as the primary metal in the Stellite alloy, to cobalt-60. Through material degradation, such as rusting carbon steel, activation products can be transferred as surface contamination. Sources of contamination may have also been leakage from sealed radioactive sources, leakage from glove boxes, leaks and spills of radioactive or contaminated liquids, and the emission of radioactive particles during experimental procedures.

6.1 LIKELY CONTAMINANTS

Contaminants at the NRL may include a variety of radioactive isotopes. These isotopes are primarily activation products, tritium, and other fission products. Activation products will appear in many of the materials used in and around the NRL reactor. Potential activation products include carbon-14 in graphite; iron-55, cobalt-60, and nickel-63 in aluminum and stainless steel; and iron-55, cobalt-60, cesium-134, and europium-152 in the heavy concrete.

Tritium is produced in a small fraction of thermal neutron fissions of uranium-235. The method of tritium generation common to all nuclear reactors has been found to be ternary fission. In this relatively rare mode of fission, the fissile nucleus breaks into two heavy fragments and one light fragment. The resulting light fragment can be any one of several light unstable or stable nuclei. In about 1 out of 10,000 fissions, tritium is formed as a product of ternary fission. Tritium is also produced in the bombardment of deuterium in deuterium oxide (heavy water). While the NRL reactor was not moderated with heavy water, a small amount of heavy water occurs in water naturally and heavy water was used in at least one experiment at the site.

Tritium is known to have been present in waste water collected and discharged to the sanitary sewer system. While discharges met all appropriate concentration limits, the accumulation of tritium in concrete may produce a contamination problem. The HSA identified primary coolant water leaks that occurred in 1971, 1974, 1976, 1978, from either the reactor tank or the primary coolant system. Impacted areas include the reactor room floor, including the thermal column door track, and the delay tank vault and sump. There were also leaks reported from the Bulk Shielding Tank.

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

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

6.2 POTENTIAL CONTAMINATED MATERIALS

Experimental equipment and reactor components have become contaminated or may be contaminated primarily with activation products. Tritium is also a major potential contaminant of concern.

6.2.1 Potential and Known Impacted Areas

There is a lot of equipment and supplies stored in the NRL facility. Some of this equipment is known to be contaminated. The known contaminated equipment include experimental equipment including a large and a small glove box (there are items in the large glove box that are likely contaminated), the specimen transfer station on top of the reactor, specimen transfer tubes, and specimen holders. Sections of the beam ports may also be contaminated due to the removal of oxidized activated metals and surface wearing.

All air filter media may be contaminated with fission products. These include the particulate and HEPA filters of the main exhaust system, at the activated charcoal bed in the charcoal filter. All other air and water filters, including fibrous and resin bed filters, located throughout the NRL may also be contaminated.

Due to the nature of the materials stored in the radioactive materials storage areas, include temporary storage areas on the reactor top, the south eastern corner of the reactor room, the beam catchers, and the dedicated radioactive materials storage cage, located in the southwestern corner of the reactor room, each of these areas are potentially impacted.

6.2.2 Non-Impacted Areas

A review of the routine survey data presented in the facility annual reports to the NRC, suggest that the mezzanine level offices, control room, and restrooms are non-impacted areas. In the period from 1980 to 2000, the highest reported removable contamination level was 600 disintegrations per minute per 100 square centimeters (dpm/100 cm^2). Surveys in the last 4 years all indicate removable contamination less than 1,000 dpm/100 cm^2. However, it should be noted that while these areas are unlikely to be impacted by tritium contamination, the surveys did not include tritium analysis.

Due to the exhaust rate of the air exhaust system, the roof of the NRL facility is expected to be non-impacted. The exterior surfaces of the walls are also expected to be non-impacted.

Leaks were never identified that would indicate cross contamination or mixing of the primary and secondary coolant water. In the event of a leak, pressure differentials in the two systems would have limited cross contamination. Therefore, the secondary coolant system, including the interior pipe and pump surface and the cooling towers are expected to be non-impacted. The exterior surfaces of the pipes and secondary pump should, however, not be considered non-impacted.

6.3 POTENTIAL ACTIVATED MATERIALS

Many of the primary reactor components are constructed on aluminum alloy. Neutron activation of aluminum may result into formation of several radioactive isotopes including aluminum-28, sodium-24, and magnesium-27 through a number of neutron capture processes. Fortunately, each of these isotopes has a very short half-life of less than 15 hours and they are of no concern. However, the aluminum alloy contains impurities in the form of other elements that can become activated by the reactor neutron flux and form radioactive isotopes. Cobalt-60 (half-life = 5.3 years) and iron-55 (half-life = 2.7 years) are the primary isotopes of concern in activated aluminum. Other activation products may include nickel-63 (half-life = 100 years), zinc-65 (half-life = 244 days), manganese-54 (half-life = 313 days), and europium-152 (half-life = 13.6 years). Some known activated aluminum tubes, a flux trap, and a removable section of the reactors top grid plate are stored in the radioactive materials storage area.

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

Stainless steel and carbon steel are also activated when exposed to neutrons. The primary activation products in reactor components constructed of these materials will be cobalt-60, iron-55, nickel-63, and manganese-54.

Graphite will also become activated and contaminated from surrounding activated materials. A graphite blind element from a German TRIGA reactor became activated and contaminated with carbon-14, cobalt-60, europium-152, and europium-154.

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

Table 6-1 provides a summary of the expected activation products and there location.

TABLE 6-1
POTENTIAL ACTIVATION PRODUCTS

MATERIAL	PRIMARY RADIONUCLIDES	LOCATION
Graphite	Carbon-14	Thermal columns and reflector assembly
Aluminum	Cobalt-60, iron-55, nickel-63, zinc-65, manganese-54, and europium-152	Reactor tank, core assembly, Lazy Susan, reflector assembly
Stainless steel	Cobalt-60, iron-55, nickel-63, and manganese-54	Beam ports, Lazy Susan
Stellite (alloy w/ more than 50% cobalt)	Cobalt-60	Lazy Susan bearings
Carbon steel	Cobalt-60, iron-55, nickel-63, and manganese-54	Beam ports, beam port plugs, shadow shields, thermal column supports
Re-bar	Cobalt-60, iron-55, nickel-63, and manganese-54	Bioshield
Heavy concrete	Cobalt-60, iron-55, europium-152, cesium-134, and europium-154	Bioshield

6.4 POTENTIAL CONTAMINATED ENVIRONMENTAL MEDIA

There is a potential for tritium to have impacted the soils under the NRL facility. While no contamination has been confirmed, the potential exists because there were known leaks of the primary coolant water onto concrete surfaces. If soils are impacted, there is potential for shallow groundwater contamination. The groundwater under the NRL is quite shallow, and during periods of heavy rain, groundwater will seep into the tunnel under the reactor room floor and delay tank vault.

Condensate from the building air conditioner compressors located outside on the north side of the building, east of the cooling towers, drains directly to Boneyard Creek. This is a potential pathway for airborne tritium removed from the building through the air conditioning system.

6.5 RADIOACTIVE SOURCES AND MATERIALS

There are numerous radioactive sources still housed in the NRL. These include a neutron generator, sealed radioactive check sources, calibration/check sources in analytical equipment, discrete activated sources, and radioactive lead. Many of the radioactive sources are stored in a controlled and locked radioactive material storage area located in the southwestern corner of the reactor room. Table 6-2 provides a list of the radioactive sources that are known to be present at the NRL at the time of the HSA. Some of the sources will be transferred to the University's broad scope license and others will be disposed of.

6.6 RELATED ENVIRONMENTAL CONCERNS

Hazardous materials may have been used at the NRL in small quantities for experimental purposes. There is currently a supply of mercury thermometers in a desk drawer in Room 102.

An asbestos survey, performed at the facility in December 2000, identified asbestos containing materials (ACM) in insulation around the magblock and cardboard pipes in the reactor room and the magblock tank insulation. The survey also assumed that the 9-inch by 9-inch floor tiles in the mezzanine level floor and in the mastic under the vinyl baseboards in the same area contained asbestos. The report also assumed that the roof material was ACM but a later analysis of the roof material showed it to be free of ACM.

Due to the nature of the operations of the reactor and the low levels of hazardous materials that may have been used during experiments, there are no suspected non-radiological environmental concerns. The ACM will require special care in handling and disposal during facility dismantlement and demolition.

TABLE 6-2
RADIOACTIVE SOURCES AT THE NRL

ISOTOPE	APPROXIMATE ACTIVITY	DESCRIPTION/ IDENTIFICATION	LOCATION
Cobalt-60	0.3 Ci	Sealed source	RAM Storage (Photo 1)
Cobalt-60	7 mCi	Particles A & B	RAM Storage
Cobalt-60	14 mCi	Particles C, D, E, F	RAM Storage
Cobalt-60	1.5 mCi	Particle G	RAM Storage
Cesium-137	25 mCi	#K-53	RAM Storage
Uranium-235	4.32 uCi (2 g)	Fission Chamber #WL-6376	RAM Storage
Uranium-235	4.32 uCi (2 g)	Fission Chamber #WL-23094	RAM Storage
Uranium-235	Unknown	Fission chamber	In reactor pool
Activation products	Unknown	Westinghouse nuclear instrument detectors (5)	RAM Storage
Radium-226	100 mCi	Sealed source	RAM Storage
Radium-226	5 mCi	Sealed source	RAM Storage
Radium-226	0.1 mCi	Sealed source	RAM Storage
Radium-226	10 uCi	In Tri-Carb 300 LSC (Photo 2)	Storage floor
Activation products	Unknown	Activated material	Lead brick shield on top of reactor
Activation products	Unknown	Activated material	Lead brick shield on top of reactor
Activation products	Unknown	Activated material	Lead brick shield on top of reactor
Radium-226	0.5 Ci	Radium-Beryllium neutron source	Reactor tank
Strontium-90	0.7 uCi	Sources in Eberline SPING-3 Air Monitor (Serial #153)	West side of reactor room
Activation products	Unknown	Activated dummy rods	Northwest beam catcher
Strontium-90 (3 or more sources)	0.2 uCi each	Source in Victoreen Model 89720 alarming radiation monitor	Outside room 100
Krpton-87	Unknown	Contaminated lead bricks	Reactor room floor
Source Storage Box	Unknown	Contaminated box	RAM Storage
Activation products	Unknown	5 activated aluminum components (5)	RAM Storage
Uranium-238, uranium-238	Unknown	349 sub-critical fuel assemblies	RAM Storage
Various	Various	Check sources	Shop area of reactor room

APPENDIX A

DOCUMENTS REVIEWED DURING THE HSA

DOCUMENTS AND FILES REVIEWED AS PART OF THE UNIVERSITY OF ILLINOIS NUCLAER RESEARCH LABORATORY HISTORICAL SITE ASSESSMENT

DOCUMENT	IDENTIFIER	DATE
TRIGA Reactor Mechanical Maintenance & Operations	GA-8698	March 1967
Hazard Analysis of TRIGA Reactor		October 1959
TRIGA Rx Instrumentation Maintenance	GA-8691	June 1968
Specifications for Nuclear Reactor Laboratory		June 1959
Irradiation Facilities		
Specifications for Construction	8017	December 1959
Aux. System Drawings		
Univ. of Ill Nuclear Rx Lab Site Description		January 1990
REPORTS, NOTES, AND FILES	**IDENTIFIER**	**DATE**
Accident Reports	E-4230	November 1976
Accident Reports	Goldman	February 1968
Accident Reports	Harrison	November 1962
Accident Reports	Harrison	June 1964
Accident Reports		December 1966
Report of Fires Extinguished	3096 E	November 1968
1960 History	File	
Supplier literature on neutron detectors	File	
Cooling tower manufacturer's information	File	
Air compressor information	File	
Reactor pumps manufacturer's information	File	
Nuclear insurance documents	File	
Research records for Hopke	File	1978 - 1987
Building exhaust filter information	File	
Heat exchanger specifications	File	1968
Large file of various engineering/product info	File	
Fuel shipment file - to University of Illinois	File	1996
Fuel shipment file - to University of Texas		2002
Reactor Committee Meeting Minutes	Files	1960 - 1978, 1984
Experimental Research Applications	Files	Various
TRIGA Annual Reports - TRIGA	Files	1969 - 2004
LOPRA Annual Report	Files	1974 - 1977
Monthly Reports	Files	Various

SCIENTECH

SITE CHARACTERIZATION REPORT

NUCLEAR RESEARCH LABORATORY
UNIVERSITY OF ILLINOIS AT CHAMPAIGN-URBANA

Prepared by:

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October 2005

Project Application	*Prepared By*	*Date*
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APPROVALS:

Title	*Signature*	*Date*
Technical Reviewer	Kenneth M. Kasper, CIH, CHP	10/6/05
D&D Operations Manager	Lee G. Penney	10/6/05

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APPENDICES

ACRONYMS AND ABBREVIATIONS

2x2 NaI	2-inch by 2-inch sodium iodide
ACM	asbestos containing materials
Canberra	Canberra Industries
Ci	curie
cpm	counts per minute
cy	cubic yards
dpm/100cm^2	disintegrations per minute per 100 square centimeters
DOE	U.S. Department of Energy
EPA	U.S. Environmental Protection Agency
g/cm^3	grams per cubic centimeter
HPGe	high-purity germanium
HSA	Historical site assessment
ISOCS	*In-Situ* Object Counting System
kg	kilograms
kg/m^3	kilograms per cubic meter
kW	kilowatt
lb/ft^3	pounds per cubic foot
L_D	limit of detection
LLRW	Low-Level Radioactive Waste
LOPRA	Low Power Reactor Assembly
LSC	liquid scintillation counter
m^3	cubic meters
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
mCi/g	millicurie per gram
MDA	minimum detectable tritium activity
MDC	minimum detectable concentration
ml	milliliters
NMNT	New Millennium Nuclear Technologies
MWhrs	mega-watt-hours
NRC	U.S. Nuclear Regulatory Commission
NRL	Nuclear Research Laboratory
OD	outer diameter
pCi/g	picocurie per gram
QA	quality assurance
Scientech	Scientech, LLC
STL	Sevren Trent Laboratories, Inc.
uCi/g	microcurie per gram
University	University of Illinois at Champaign-Urbana

1.0 EXECUTIVE SUMMARY

The University of Illinois at Champaign-Urbana (University) discontinued operations of its Advanced TRIGA nuclear research reactor at its Nuclear Research Laboratory (NRL) in 1998. Following shutdown of the reactor and the removal of the reactor fuel elements in 2004, the University decided to decommission and dismantle the NRL and terminate U.S. Nuclear Regulatory Commission (NRC) Facility Operating License No. R-115. Prior to license termination, the NRL facility will be completely dismantled and components will be disposed of according to their radiological condition.

In July 2005, Scientech, LLC (Scientech) performed a detailed characterization of the NRL facility. The characterization was preformed in accordance with the facility Characterization Plan (Scientech 2005a) to identify radiologically impacted systems and structures. The characterization was also designed to determine the extent of activation in the reactor bioshield through concrete sampling and on-site analysis. The information obtained as a result of the site characterization, which is presented in this Characterization Report, will assist in the preparation of the facility Decommissioning Plan. A complete understanding of the site conditions is important in determining the necessary decontamination or dismantlement techniques and designing the radiation safety and material handling procedures required to safely perform the necessary decommissioning tasks. This Characterization Report also presents waste volume estimates that will be used in the preparing the decommissioning cost estimate.

The scope of the characterization survey included the complete NRL facility which encompasses the reactor level, the mezzanine level, and the storage level of the reactor building. The reactor level contains the TRIGA reactor, the Bulk Shielding Tank, the Mechanical Equipment Room, a pipe tunnel, two tank vaults, and several support areas. The mezzanine level contains the office and reactor operations area, the restrooms, the open walkway, and the loading bay. The storage level, located above the mezzanine level contains storage areas and an office. The characterization also examined the soils surrounding and underneath the reactor building.

The characterization of the NRL facility included measurements for total and removable alpha and total beta surface activity and removable tritium surface activity. Samples of concrete from the reactor room floor, the Bulk Shielding Tank floor, the reactor bioshield, and several other areas of the facility were also collected and sampled for tritium, total beta activity, and gamma-emitting isotopes. Soil samples were analyzed for tritium.

The characterization effort identified the reactor bioshield, the primary coolant pipe tunnel, and the concrete floor of the reactor room as primary impacted areas. Large pieces of contaminated and activated equipment identified include the internal reactor components, the nitrogen-16 decay tanks, the primary coolant pipes, and the large glove box. The characterization effort also produced estimates for the activated and contaminated volume of concrete in the bioshield and reactor room floor, estimates for the radionuclide inventory of the internal reactor components (including the rotary specimen rack), and estimates of the radionuclide inventory and volume of activated graphite.

2.0 HISTORICAL SITE ASSESSMENT

A complete Historical Site Assessment (HSA) of the NRL was conducted by Scientech in May 2005. The HSA included document reviews, a site visit, and interviews with current and past reactor personnel. The following sections provide some of the information derived from the HSA. Additional information on the reactor facility can be found in the full HSA report (Scientech 2005b).

2.1 REACTOR OPERATIONS

Construction began on the NRL in the summer of 1959 to house a TRIGA Mark II training and research nuclear reactor manufactured by the General Atomic Division of General Dynamics Corporation. The construction of the building and installation of the reactor was performed by the University. By 1960, the walls of the NRL we complete and the foundation for the reactor and bioshield and the thermal column trench were complete. By the spring of 1960, the reactor and reactor tank were installed along with the beam ports and the forms for bioshield concrete. The NRL was completed in the summer of 1960 and the reactor first went critical on August 16, 1960.

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

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

Additional modifications that were made during the upgrade were the installation of additional beam ports, an underground vault to house the heat exchanger and nitrogen-16 decay tanks, and a tunnel to shield the primary water before it reaches the decay tanks.

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

The Bulk Shielding Tank, located on the south side of the reactor, allowed for neutron beam experiments to be conducted underwater for additional shielding. The Bulk Shielding Tank was also the home of the Low Power Reactor Assembly (LOPRA) which was a subcritical assembly that used TRIGA fuel. The LOPRA operated under its own NRC license (No. R-117) beginning in 1971. The R-117 license governed the use of the LOPRA until 1995 when the fuel and subcritical assembly were transferred to the NRL's current R-115 license.

On August 6, 1998, nearly 30 years since its initial start-up and after 11,566.7 mega-watt-hours (MWhrs) of operation, the NRL TRIGA reactor was shut down permanently. In 1999, the reactor was officially placed in a SAFSTOR condition while waiting for arrangements to be made to remove and ship the reactor fuel. The Bulk Shielding Tank was used for wet storage of the fuel following shutdown. On August 18, 2004, the reactor fuel was removed and shipped to the U.S. Department of Energy's Idaho National Environmental and Engineering Laboratory.

2.2 CONTAMINATION POTENTIAL

The HSA identified various radioactive isotopes at the NRL that may have impacted the facility. These isotopes are primarily activation products and tritium. Activation products will appear in many of the materials used in and around the NRL reactor. Potential activation products include carbon-14 in graphite; iron-55, cobalt-60, and nickel-63 in aluminum and stainless steel; and iron-55, cobalt-60, cesium-134, and europium-152 in the heavy concrete.

Tritium is produced in a small fraction of thermal neutron fissions of uranium-235. The method of tritium generation common to all nuclear reactors has been found to be ternary fission. In this relatively rare mode of fission, the fissile nucleus breaks into two heavy fragments and one light fragment. The resulting light fragment can be any one of several light unstable or stable nuclei. In about 1 out of 10,000 fissions, tritium is formed as a product of ternary fission. Tritium is also produced in the bombardment of deuterium in deuterium oxide (heavy water). While the NRL reactor was not moderated with heavy water, a small amount of heavy water occurs in water naturally. In addition, heavy water was used in at least one experiment at the site.

Tritium is known to have been present in waste water collected and discharged to the sanitary sewer system. While discharges met all appropriate concentration limits, the accumulation of tritium in concrete may present a contamination problem. The HSA identified primary coolant water leaks that occurred in 1971, 1974, 1976, 1978, from either the reactor tank or the primary coolant system. Impacted areas include the reactor room floor, including the thermal column door trench, and the decay tank vault and sump. There were also leaks reported from the Bulk Shielding Tank.

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

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

The HSA identified that there is also a potential for tritium to have impacted the soils under the NRL facility. While no contamination was confirmed prior to the site characterization, the potential exists because there were known leaks of the primary coolant water onto concrete surface. Significant soil contamination may result in shallow groundwater contamination. The groundwater under the NRL is quite shallow, and during periods of heavy rain, groundwater will seep into the tunnel under the reactor room floor and decay tank vault.

2.3 ACTIVATION POTENTIAL

Many of the primary reactor components are constructed of an aluminum alloy. Neutron activation of aluminum may result in to formation of several radioactive isotopes including aluminum-28, sodium-24, and magnesium-27 trough a number of neutron capture processes. Fortunately, each of these isotopes has a very short half-life of less than 15 hours and they are of no concern. However, the aluminum alloy contains impurities in the form of other elements that can become activated by the reactor neutron flux and form radioactive isotopes. Cobalt-60 (half-life = 5.3 years) and iron-55 (half-life = 2.7 years) are the primary isotopes of concern in activated aluminum. Other activation products may include nickel-63 (half-life = 100 years), zinc-65 (half-life = 244 days), manganese-54 (half-life = 313 days), and europium-152 (half-life = 13.6 years). Some known activated aluminum tubes, a flux trap, a removable section of the reactors top grid plate, and other radioactive materials are stored in the Radioactive Materials Storage Cage.

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

Stainless steel and carbon steel are also activated when exposed to neutrons. The primary activation products in reactor components constructed of these materials will be cobalt-60, iron-55, nickel-63, and manganese-54.

Graphite will also become activated and contaminated from surrounding activated materials. A graphite blind element from a German TRIGA reactor became activated and contaminated with carbon-14, cobalt-60, europium-152, and europium-154 (Hampel, et. al., 2001).

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

3.0 CHARACTERIZATION METHODOLOGY

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

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

3.1 RADIONUCLIDES OF CONCERN

The HSA identified the primary radionuclides of concern as tritium and activation products. The tritium sources included concrete activation and contamination from leaks from the reactor tank and Bulk Shielding Tank. The tritium may have also impacted the concrete bioshield, the concrete slab floor, and the soil under the reactor building. The activation products are created in concrete and metal reactor components when exposed to the reactor's neutron flux. The primary activation products of concern include cobalt-60 and europium-152, both gamma emitters; iron-55, which decays by electron capture and emits only weak x-rays; and nickel-63, a low-energy beta-emitter with no gammas. Carbon-14 is also an activation product in graphite. Decay information on the primary radionuclides of concern is provided in Table 3-1.

3.2 SURFACE CONTAMINATION MEASUREMENTS

To assess the general radiological condition of the NRL facility, surface contamination measurements were made throughout the facility using properly calibrated instruments capable of detecting both total alpha and total beta surface activity. To assess removable contamination, a smear sample was taken over a $100 cm^2$ surface area and analyzed for tritium, carbon-14, and total beta activity or for total alpha and total beta activity. The instruments used for these measurements are listed in Table 3-2.

TABLE 3-1
RADIONUCLIDES OF CONCERN

Radionuclide	Decay Mode	Half-Life	Energies	Media
Tritium	Beta only	12.28 years	18.6 keV (maximum) 5.69 (average)	Concrete, Soil
Carbon-14	Beta only	5,730 years	156.5 keV (maximum) 49.5 (average)	Graphite
Iron-55	Electron Capture	2.7 years	-	Concrete Aluminum Stainless Steel
Cobalt-60	Beta/gamma	5.27 years	1173.2 keV gamma 1332.5 keV gamma	Concrete Aluminum Stainless Steel
Nickel-63	Beta only	100.1 years	65.9 keV (maximum) 17.1 (average)	Stainless Steel
Europium-152	Beta/gamma	13.6 years	121.8 keV gamma 344.3 keV gamma 996.3 keV gamma 1112.1 keV gamma 1408.0 keV gamma	Concrete Aluminum Stainless Steel

TABLE 3-2
SURFACE CONTAMINATION MEASUREMENT INSTRUMENTS

	Instrument Model	Probe Model	Radiations Detected	Instrument Ser. No.	Probe Ser. No.	Calibration Date
1	Ludlum 2360	43-93	Total α Total β	184941 202455	200125 200135	7/6/05 7/6/05
2	Ludlum 2221	44-9	Total β	68795 169226	182612 182611	6/8/05 12/6/04
3	Ludlum 12	43-37	Total α Total β	177883	160824	8/5/04
4	Ludlum 2929	43-10-1	Removable α Removable β	167852	174811	6/22/05
5	Beckman 6500LS	NA	Removable: Tritium Carbon-14 Total beta	7068817	NA	7/18/05

Maps of the floor areas of the NRL facility with survey grids are provided in Appendix A-1 and A-2. Direct total alpha and total beta measurements were made at the intersections of the grid lines shown in the figures. Ludlum 2360 alpha/beta phoswich detectors (Instrument #1 in Table 3-2) were used for these measurements. The minimum detectable activity for the direct instruments varied slightly with background, but they were generally about 430 dpm/100cm^2 beta and 35 dpm/100cm^2 alpha. Several beta activity data points were rejected because of the high background radiation level at the survey location. The high background areas were on the mezzanine level and the reactor level near the Radioactive Materials Storage Cage, near the storage area in the southeastern corner of the reactor level, near the graphite blocks removed from the large thermal column, and on top of the reactor.

Scanning measurements were performed on floors and walls of the NRL facility where radiation levels permitted. 100% of the accessible floor areas were scanned using a Ludlum floor monitor (Instrument #3 in Table 3-2). Parts of the reactor level floor could not be scanned due to the high background radiation level caused by the materials in the Radioactive Materials Storage Cage. Wall scans were performed with the Ludlum 2360s. The coverage of the wall scans was as follows:

Room 102 – Control Room	10%
Mezzanine Walkway	10%
Truck Bay	10%
Storage Level	10%
Mechanical Equipment Room	10%
Reactor Level Lower Walls	10%
Vertical surfaces of bioshield	25%

3.3 DOSE RATE AND GAMMA MEASUREMENTS AND DOSE MODELING SOFTWARE

For the purpose of measuring high-dose rate fields, an Eberline RO-7 ion chamber was used with multiple probes. An underwater housing was also used to measure dose rates inside the reactor tank. A Ludlum Model 19 gamma exposure rate meter was used for general area exposure rates in low-dose fields. For surveys of equipment with inaccessible surfaces that were potently contaminated or activated, 2-inch by 2-inch sodium iodide (2x2 NaI) gamma scintillation detectors were used to measure the total gamma activity. Table 3-3 provides information on these two detectors.

The MicroShield® Version 6.01 dose modeling software was used to estimate the activity of radioactive materials based on dose rates measured with the ion chamber.

TABLE 3-3
DOSE/EXPOSURE RATE MEASUREMENT INSTRUMENTS

	Instrument Model	Probe Model	Radiations Detected	Instrument Ser. No.	Probe Ser. No.	Calibration Date
6	Eberline RO-7 (Ion Chamber)	RO7LD RO7BM RO7BH	Low dose γ Mid dose γ High dose γ	480	726968 714855 726531D	6/28/05
7	Ludlum 19 (gamma scint.)	NA	Low dose γ	209728	NA	9/20/04
8	Ludlum 2221 (2x2 NaI)	44-10	Gamma	73700	230163	6/29/05

3.4 SAMPLE COLLECTION

Fifty-two concrete samples were collected at the NRL facility using the TruPro® sampling technology to characterize the reactor room floor and the reactor bioshield. This technology is described in Section 3.4.1. Twelve soil samples were collected using the TruPro® technology. An additional 56 subsurface soil samples were collected using GeoProbe® soil samplers. The GeoProbe® sampling is described in Section 3.4.2.

The TruPro® technology was also used to collect samples of the graphite removed from the large thermal column. To collect metal samples from reactor components, a standard power drill was used to produce metal shavings and the TruPro® vacuum pump was used to collect the sample material.

A breakdown of the material samples collected is provided in Table 3-4. Table 3-4 also provides the analyses that were performed on the samples. The sample analytical methods are described in Section 3.5.

3.4.1 TruPro®

The TruPro® sampling technology was deployed by New Millennium Nuclear Technologies (NMNT) of Lakewood, Colorado, as a subcontractor to Scientech. This technology, which was used in place of traditional core boring, uses a hammer drill with hollow 12-millimeter drill bits to collect concrete powder samples. While drilling, the drill bit is connected to a vacuum pump that removes all of the sample material from the hole as the drill is advanced. The sample material is collected in a filter and then transferred directly to a plastic sample container. Because the hollow drill bits are connected to a vacuum pump, the drilling process does not generate airborne dust that might cause exposure issues.

For samples of the concrete floor, three 1-inch holes were drilled at each of the sample locations to provide enough sample volume for gamma spectroscopy analysis. For samples collected in the bioshield, the drill was first advanced to the sample point, for example, 12 inches from the outside surface of the bioshield. Next, the vacuum was directed to a separate sample filter and the drill was advanced three inches. The material from these three inches made up the sample material for that sample point. The sample locations are provided with the analytical results in Appendix A-4.

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

The TruPro® technology was also used to collect samples of graphite blocks from the thermal column. While the TruPro® technology inherently limits airborne contamination, to further minimize the potential for airborne contaminants, the graphite samples were collected in a sample collection box that was connected to a high-efficiency particulate air (HEPA) filter vacuum.

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

TABLE 3-4
SAMPLE COLLECTED DURING THE NRL CHARACTERIZATION

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

Notes:

a Eleven concrete and two soil samples were sent off-site for iron-55 and nickel-63 analysis (see Section 3.5.4).

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

3.4.2 GeoProbe®

Two different GeoProbe® sampling units were used by Direct Push Analytical Corp. of Batavia, Illinois to collect 60 subsurface soil samples from below and around the NRL facility. The GeoProbe® technology uses hydraulically driven rods to advances sample collection sleeve into the soil to a desired depth. To collect the samples, the sample sleeve is brought to the surface and the soil from the desired depth range is removed. Scientech used clean stainless steel spoons to remove the soil and transfer it to a plastic zipper bag. The samples were then logged into the field logbook. Prior to shipping the samples for analysis, the samples were transferred into 0.5-liter polyethylene sample containers, labeled, seal with electrical tape, and logged on the sample chain-of-custody form.

Samples collected outside the NRL building were collected at 4-to-5, 8-to-9, and 12-to-13-foot depths below the ground surface (surface grade). Samples collected inside the reactor room, through the concrete floor were taken at 0-to-1, 3-to-4, 6-to-7, 9-to-10, and 12-to-13-foot depths below the floor. The reactor room floor is about 6 feet below the ground surface and the floor is 2 feet thick. Therefore, the 0-to-1-foot sample from inside the building corresponds to the same depth below the surface grade as the 8-to-9 foot sample from the outside sample points.

A map of the sampling points is provided with the analytical results in Appendix A-7.

3.5 SAMPLE ANALYSIS

Several analytical techniques were used to examine samples for radionuclides of concern. For concrete, graphite, metal, and some soil samples, gamma spectroscopy was used to identify and quantify the levels of gamma-emitting radionuclides in the samples using an on-site high-purity germanium (HPGe) detector. The concrete, graphite, and soil samples collected by NMNT were also analyzed for tritium and total beta activity using a Beta Scout® portable LSC. The subsurface soil samples collected using the GeoProbe® were sent off site to Severn Trent Laboratories (STL) in Earth City, Missouri for tritium analysis. These samples were not analyzed on-site using gamma spectroscopy because activation products were not considered contaminants of concern for the subsurface soils. STL also performed analysis for iron-55 and nickel-63 on selected concrete samples. Each of these analyses is described in further detail in the following sections.

3.5.1 On-Site Liquid Scintillation Counter

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

The method minimum detectable tritium activity (MDA) was determined by calculating the mean and standard deviation of both portable LSCs using a background data set that included the one "background" sample from bioshield and additional samples from the bioshield that, based on LSC and gamma spectroscopy analysis, could also be considered part of the background population because there was no indication that there were either activated or contaminated. These samples and their tritium and total beta count rates are provided in Table 3-5.

To calculate the MDA, the limit of detection (L_D) was first calculated using the following equation from the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NRC 2000).

$$L_D = 3 + 4.65\sqrt{B}$$

The square root of B is equal to the standard deviation of the background data set provided in Table 3-5.

To convert the L_D (in gross cpm) to the MDA (in pCi/g), the L_D is divided by the LSC tritium efficiency and the sample mass.

$$MDA(pCi/g) = \frac{L_D(grosscpm) \times 27.027(pCi/dps)}{mass \times efficiency \times 60(\sec/\min)}$$

There were no reported efficiencies for either LSC for total beta activity. Therefore, it was assumed that the efficiencies for the total beta activity were equal to the tritium efficiencies. This is a conservative assumption that leads to reporting inflated total beta activities.

As shown in Table 3-5, the MDAs for concrete samples in the two portable LSCs were 7,665 and 19,797 pCi/g. These MDAs are quite high and do not allow for the identification of low levels of tritium activation or contamination. The results of the NMNT LSC analyses are provided in the subsections of Section 4.0.

TABLE 3-5
BACKGROUND DATA SET

	Tritium (cpm)		Total Beta (cpm)	
Sample Number	**LSC #1 (20070060)**	**LSC #2 (20070112)**	**LSC #1 (20070060)**	**LSC #2 (20070112)**
NMNT-1	212	215	9,210	5,950
NMNT-22	137		11,710	
NMNT-23	174		9,679	
NMNT-25	148		7,703	
NMNT-26		172		11,119
NMNT-54	160		8,510	
NMNT-55		171		11,245
NMNT-56	193		11,358	
NMNT-57		213		5320
NMNT-61	130		8,084	
NMNT-62		137		7,807
NMNT-63	158		9,881	
NMNT-64	182		7,831	
Mean Background	166	182	9,330	8,288
Standard Deviation	26.8	32.8	1,472.1	2,795.8
L_D (net cpm)	128	156	6,849	13,004
L_D (gross cpm)	294	338	16,179	21,293
Tritium Efficiency (cpm/dpm)	0.1514	0.07881	0.1514	0.07881
Sample Mass	0.1 g	0.1 g	0.1 g	0.1 g
MDA (pCi/g)	8,729	19,299	481,350	1,217,142

Figure 3-1 shows the distribution of the gross tritium count rates as reported by NMNT. The distribution plot of the total beta count rates is provided in Figure 3-2. The data from both detectors are combined on the plots and average L_Ds for the two detectors are also shown.

The total beta count rates are provided as an indicator of possible other beta-emitting contaminants or activation products such iron-55, cobalt-60, nickel-63, or europium-152. If an elevated beta count rate occurred in a sample when gamma spectroscopy suggested that there were no gamma-emitting activation products present, the total beta is considered to be an indicator of either additional tritium not detected in the tritium window of the LSCs or another non-gamma-emitting contaminant such as iron-55.

FIGURE 3-1
DISTRIBUTION OF NMNT TRITIUM DATA

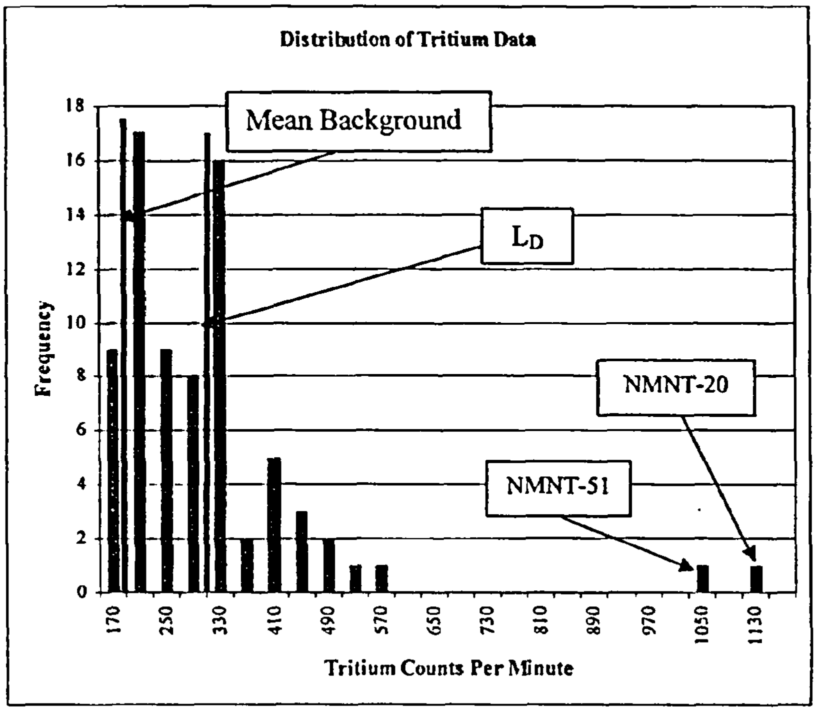
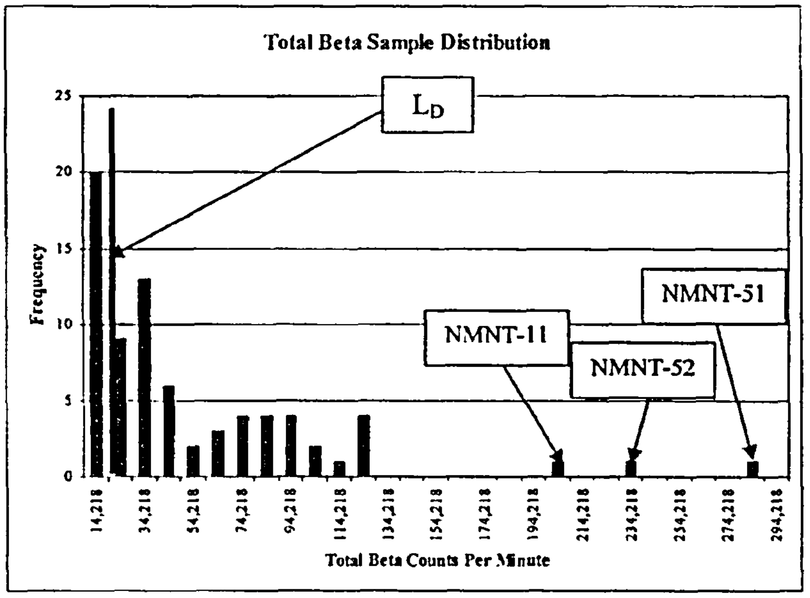


FIGURE 3-2
DISTRIBUTION OF NMNT TOTAL BETA DATA



3.5.2 On-Site Gamma Spectroscopy

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

The Canberra *In-Situ* Object Counting System (ISOCS)® Geometry Composer software was used to model the sample geometries to provide efficiency files. The efficiency files were used by Genie2000® to perform the detector efficiency calibrations. The energy calibrations were performed using a europium-152 multi-peak gamma radiation standard.

The gamma spectra analysis used a limited radionuclide library that contained naturally-occurring radionuclides, the primary radionuclides of concern (cobalt-60, europium-152, and europium-154), and the fission product cesium-137. MDAs for a 20 gram background concrete sample with a 10 minute count time are provided in Table 3-8. The MDAs increase with sample activity.

TABLE 3-8
TYPICAL GAMMA SPECTROSCOPY MDAs FOR A LOW-ACTIVITY SAMPLE

Radionuclide	MDA (pCi/g)
Potassium-40	38.6
Cobalt-60	3.25
Cesium-137	3.75
Europium-152	7.25
Europium-154	10.6
Radium-226+D	6.67
Actinium-228	13.9

The HPGe detector and the ISOCS® geometry software were also used to quantify the activity of several other objects using similar methods. These included a sample of the boral curtain, a graphite block, stacks of graphite blocks, contaminated lead bricks, and a radium-beryllium neutron source.

3.5.3 Off-Site Tritium Analysis

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

3.5.4 Off-Site Iron-55 and Nickel-63 Analysis

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

4.0 CURRENT RADIOLOGICAL CONDITIONS

The following sections provide the current radiological status of the NRL facility and its major components, however, the radiological conditions are expected to change prior to demolition. Because two of the primary isotopes of concern, cobalt-60 and iron-55, are relatively short-lived with half-lives of 5.3 years and 2.7 years respectively, site conditions will change slightly in the time prior to facility demolition. Facility demolition is not expected to start until the fourth quarter of 2007, almost one iron-55 half-life and about half of a cobalt-60 half-life has passed. There are also some radioactive materials stored on-site that will likely be removed before decommissioning operations begin. These materials are not addressed in this Characterization Report but are discussed in the HSA (Scientech 2005b). The presence of these sources did not allow for complete characterization in some areas of the facility because the background radiation levels were too high. Specifically, direct surface contamination measurements could not be made in areas near the radioactive materials storage cage, near the large glove box, near the radioactive materials stored on top of the reactor, and in the southeastern corner of the facility.

The data referenced in the following sections is located in the various sections of Appendix A. Survey data sheets for direct total surface activity measurements provide the gross measurement (in cpm); the calculated surface activity [in disintegrations per minute per 100 square centimeters (dpm/100cm2)]; and the reported activity which is the measured activity if the activity is greater than the minimum detectable concentration (MDC). For activity measurements that are less than the MDC, "<MDC" is the reported value. For removable surface activity, the tritium and carbon-14 activity results from the analysis conducted in the University's Beckman 6500 LSC are provided in dpm/100cm^2. While the total beta count rate is provided by the LSC in cpm, the carbon-14 efficiency is used to estimate the total removable activity in dpm/100cm^2.

4.1 MEZZANINE AND STORAGE LEVEL FLOORS AND WALLS

The survey of the mezzanine level included the following rooms:

- Room 100 – Lobby
- Room 101 – Office (Former Conference Room)
- Room 102 – Office and Reactor Control Room
- Room 103 – Toilet
- Room 104 – Janitor's Closet
- Room 105 – Toilet

The mezzanine area survey also included the walkway adjacent to the control room that overlooks the reactor room and the loading bay south of the mezzanine level.

In these areas, a systematic contamination survey was conducted that included direct fixed-point surface contamination measurements, surface scans, and removable contamination measurements. The direct measurement and removable contamination data are present in Appendix A-1. A map showing the location of the survey points is also provided in Appendix A-1.

Several direct beta activity measurements in the area were rejected (R qualifier) when an anomalous surface beta activity measurement was recorded in an area of known high background radiation. The background was high due to the presence of high-activity gamma sources stored in the Radioactive Materials Storage Cage located below the southern end of the Mezzanine area. Direct measurements were not taken at survey points 10, 11, and 15 due to the high background.

Three removable contamination swipes indicated a significant amount of removable tritium contamination at more than 1,000 dpm/100cm^2, the standard maximum removable beta contamination release limit for decontamination and decommissioning of nuclear facilities (NRC 1993). However, dose-based release limits for removable tritium contamination are typically much higher. The elevated measurements were taken at survey points 10, 11, and 13 in the loading bay area. Additional measurements taken around point 13 were low indicating a localized area of contamination.

Because of the high removable tritium contamination measurement in the loading bay, two concrete samples were taken in the area and analyzed for tritium, gross beta, and gamma-emitting isotopes. Each sample consisted of the top 1 inch of concrete removed from three quarter-inch holes using NMNT's TruPro® sampling technology. The results are summarized in Table 4-1.

TABLE 4-1
ANALYSIS RESULTS – LOADING BAY CONCRETE SAMPLES

Sample No.	Media	H-3 (pCi/g)	Total Beta[a] (pCi/g)	Co-60 (pCi/g)	Eu-152 (pCi/g)
NMNT-68	Concrete	1.93E+04	1.30E+06	82.6[b]	88.2[b]
NMNT-69	Concrete	< MDA	2.80E+05	< MDA	< MDA

Notes: a Estimated based on tritium efficiencies.
b Potential cross contamination.

The gamma spectroscopy analysis of the samples indicated less than detectable levels of cobalt-60 in sample NMNT-69 and no other isotopes of concern were identified in the sample. The analysis of sample NMNT-68 indicated that the sample contained elevated levels of cobalt-60 and europium-152. Based on a review of the sample data, it appears that this sample was partially cross-contaminated from residual material still in the sampling tool from sample NMNT-67. NMNT-67 was a high-activity sample collected from the reactor bioshield just prior to collecting NMNT-68.

Sample NMNT-67, however, was "< MDA" for tritium. Therefore, because the tritium activity increased from NMNT-67 to NMNT-68 and there were detectable levels of removable tritium contamination in the area, the tritium activity in NMNT-68 is not considered to be from cross-contamination.

The results of the total and removable contamination measurements on the walls in these areas are provided in Appendix A-1. Survey points 101 through 109 were from the mezzanine level. Several of the total beta activity measurements were rejected due to high gamma background caused by the materials in the Radioactive Materials Storage Cage located below the southern portion of the mezzanine level. Survey points 110 through 115 were in the loading bay. All but 2 of these measurements were rejected due to high background caused by the proximity of the Radioactive Materials Storage Cage. Survey points 116 through 120 were on the storage level. Other than the rejected data, no anomalies were identified during the direct or removable contamination wall surveys in these areas.

4.2 REACTOR ROOM FLOOR AND WALLS

The reactor room floor is a poured concrete slab about 2 feet thick. There were 75 survey points on the floor of the reactor room floor (numbered 1 through 74 and 101) and an additional 16 survey points in the Mechanical Equipment Room (numbered 85 through 100). There were no survey points numbered 75 through 84. Many of the direct beta measurements had to be rejected due to high gamma background activity. These measurements were primarily in the southwestern area of the reactor room near the Radioactive Materials Storage Cage. No direct measurements indicated the presence of significant detectable surface contamination. The direct measurement and removable contamination data are present in Appendix A-2. A map showing the location of the survey points is also provided in Appendix A-2.

There were no significant removable tritium contamination measurements at the survey points on the reactor room floor. The highest reportable removable tritium concentration was 333 dpm/100cm^2 at point 54, northeast of the reactor bioshield. Removable tritium measurements in the southwestern corner of the Mechanical Equipment Room at points 93 and 97 were higher. The reportable removable tritium concentrations at these points were 430 and 8,474 dpm/100cm^2 respectively. Five additional smears taken from this area because of the other elevated smears had detectable removable tritium activity ranging from 103 to 5,040 dpm/100 cm^2. The contaminated area in the Mechanical Equipment Room appears to be limited to less than 100 square feet (ft^2) of the southwestern corner of the room.

Thirteen concrete samples were taken from the floor of the reactor room. Ten of the samples were taken at points on the surface survey grid and three samples were taken from the thermal column shield door trench. One additional concrete sample was taken in the southwestern corner of the Mechanical Equipment Room. The results of the tritium, gross beta, and gamma spectroscopy analysis are provided summarized in Table 4-2. The complete LSC and gamma spectroscopy results are provided in Appendix B.

TABLE 4-2
ANALYSIS RESULTS – REACTOR ROOM FLOOR CONCRETE SAMPLES

Sample No.	Media	H-3 (pCi/g)	Total Beta[a] (pCi/g)	Co-60 (pCi/g)	Eu-152 (pCi/g)
NMNT-3	Concrete	2.28E+04	9.94E+05	<MDA	<MDA
NMNT-5	Concrete	1.97E+04	1.24E+06	<MDA	<MDA
NMNT-6	Concrete	1.26E+04	4.08E+05	<MDA	<MDA
NMNT-7	Concrete	2.21E+04	1.13E+06	<MDA	<MDA
NMNT-8	Concrete	1.54E+04	4.82E+05	<MDA	<MDA
NMNT-9	Concrete	1.81E+04	7.82E+05	<MDA	<MDA
NMNT-11	Concrete	3.34E+04	2.53E+06	<MDA	<MDA
NMNT-12	Concrete	1.41E+04	4.56E+05	<MDA	<MDA
NMNT-13	Concrete	2.08E+04	1.07E+06	<MDA	<MDA
NMNT-14	Concrete	2.15E+04	6.15E+05	<MDA	<MDA
NMNT-16	Concrete	1.52E+04	3.66E+05	<MDA	<MDA
NMNT-17	Concrete	2.66E+04	1.57E+06	<MDA	<MDA
NMNT-18	Concrete	1.27E+04	4.40E+05	<MDA	<MDA
NMNT-19	Concrete	2.60E+04	1.47E+06	<MDA	<MDA

Note: a Estimated based on tritium efficiencies.

The tritium and total beta count rate from concrete floor and trench samples demonstrate that the concrete is contaminated with tritium in varying concentrations. All fourteen samples had tritium levels greater than the MDA.

While tritium levels in the reactor water were always within discharge limits during reactor operation, it is suspected that the tritium contamination resulted from years of buildup from small leaks in the reactor and Bulk Shielding Tank. The calcite deposits along the southeastern wall of the bioshield were sampled and analyzed on-site using the University's LSC. The tritium levels in the calcite sample were about 800 pCi/g.

The estimated total beta activity in the concrete samples ranged from 0.4 to greater than 2.5 uCi/g. This is expected to be a conservative over-estimate because the counting efficiencies for tritium were used. It is also expected that most of the measured gross beta activity is from tritium. This conclusion is based on the fact that the locations of the elevated gross beta measurements do not correspond with the elevated activation products, which are also beta-emitters. Therefore, the most likely contributor to the high beta count rates is tritium. In the open window setting of the LSC, more tritium betas are available for detection. These higher energy betas also have a higher probability of being detected resulting in higher count rates.

Bulk samples were taken of the concrete from 1 inch to 2 feet at three locations. Due to the mobility of tritium in porous media such as concrete, it is expected that the concrete floor will contain extensive volumetric contamination. As such the entire concrete floor of the reactor room will be disposed of as low-level radioactive waste (LLRW) or released through the dose assessment process. The volume of the concrete floor is estimated to be 267 cubic yards (cy) (80 ft x 45 ft x 2 ft = 267 cy).

The gamma spectroscopy results indicate that the floor is not contaminated with activation products. In all floor samples, gamma-emitting nuclides of concern were not identified; cobalt-60 and europium-152 concentrations were less than their respective MDAs.

The results of the direct measurements on the walls in these areas are provided in Appendix A-2. Survey points 121 through 128 were in the small mechanical/electrical room in the northwestern corner of the reactor room. There were several high total beta activity measurements in this room but no elevated removable activity. Survey points 129 through 135 were in the western side of the reactor room under the mezzanine level. Several of the total beta activity measurements were rejected due to high gamma background caused by the materials in the Radioactive Materials Storage Cage.

Survey points 139 through 161 were located on the reactor room lower walls starting at the northern stairway and moving clockwise around the room to the southern stairway. Several of the total beta activity measurements were rejected due to high gamma background levels in several areas. Survey points 162 through 181 were located around the lower walls of the reactor bioshield beginning on the southern wall and working clockwise around the bioshield. Several of these measurements were also rejected due to high background levels. Survey points 185 through 192 were located in the large Mechanical Equipment Room.

The removable contamination surveys identified two sample points with elevated removable contamination levels. Survey point 147, located on the eastern wall near the northeastern corner had a reported removable tritium activity of 132 dpm/100cm^2 (> MDA) and a total removable beta activity of 105 dpm/100cm^2. Survey point 154, located on the southern wall above the workbench had a reported removable carbon-14 activity of 691 dpm/100cm^2 and a total removable beta activity of 1,077 dpm/100cm^2. There are many contaminated items currently stored in this location.

4.3 PRIMARY COOLANT PIPE TUNNEL

A tunnel containing two 10-inch aluminum pipes is located under the reactor and reactor room floor. One pipe carried primary coolant water to the nitrogen-16 decay tanks and the other returned the primary coolant water to the reactor from the heat exchanger. The tunnel is about 3 feet wide and 4 feet high with filled concrete block walls and a poured concrete floor. This tunnel is known to have received groundwater infiltration during heavy rain events.

4.3.1 Tunnel Surveys

Total beta activity measurements with a GM detector (Instrument #2 in Table 3-2) and removable contamination measurements taken in the tunnel indicated that there was no detectable fixed or removable surface contamination. However, the gamma exposure rates (measured with Instrument #7 in Table 3-3) increased toward the vertical pipes in the back of the tunnel and direct activity measurements were not possible. The total and removable contamination measurements and exposure rates are mapped on a figure provided in Appendix A-3. The survey results are also provided in Appendix A-3.

Gamma surveys with a 2x2 NaI detector (Instrument #8 in Table 3-3) indicated that the primary coolant pipe elbows are either internally contaminated or activated with the outflow pipe having higher activity than the return pipe. It is most likely that the pipes are internally contaminated with a layer of solid materials including metal oxides containing high concentrations of cobalt-60. Since the flow through the pipes has been terminated, settling of the materials in the pipe elbow is possible.

The elbows may also be activated from neutrons streaming down from the reactor core. The center of the core is located about 6 to 7 feet directly above the outflow pipe with only water and a thin layers of aluminum from the bottom of the reactor assembly support acting as a shield. The concrete sample of the tunnel floor taken near the pipe elbows was not, however, contaminated with activation products (see Section 4.3.2 below).

4.3.2 Tunnel Sampling

Four samples of the concrete tunnel floor were taken and analyzed for tritium and activation products. Four concrete samples were also taken of the tunnel ceiling/reactor bottom from the tunnel at four different elevations to an elevation of 15 inches above the tunnel ceiling, about halfway to the reactor tank. Three soil samples were collected from under the reactor through the tunnel walls and four soil samples were collected from below the tunnel floor. Each sample was analyzed for tritium and gross beta activity using a portable LSC and for gamma-emitting isotopes (activation products) using gamma spectroscopy. The sample locations are provided on a figure in Appendix A-3. Table 4-3 provides a summary of the tunnel sampling data. A summary data table is also provided in Appendix A-3.

TABLE 4-3
ANALYSIS RESULTS – TUNNEL CONCRETE SAMPLES

Sample No.	Media	H-3 (pCi/g)	Total Beta[a] (pCi/g)	Co-60 (pCi/g)	Eu-152 (pCi/g)
NMNT-40	Concrete	< MDA	2.59E+05	23.6	18.7
NMNT-42	Concrete	< MDA	2.09E+05	< MDA	< MDA
NMNT-44	Concrete	< MDA	1.68E+05	< MDA	< MDA
NMNT-47	Concrete	< MDA	2.86E+05	< MDA	< MDA
NMNT-50	Concrete	1.76E+04	1.51E+06	< MDA	< MDA
NMNT-51	Concrete	5.59E+04	1.84E+06	< MDA	< MDA
NMNT-52	Concrete	3.69E+04	2.94E+06	< MDA	< MDA
NMNT-53	Concrete	1.80E+04	5.41E+05	< MDA	< MDA

Notes: a Estimated based on tritium efficiencies.

The concrete samples from the tunnel floor showed no tritium above the MDA and total beta activity at two to three times the background count rate. The elevated total beta activity may, however, be from tritium contamination that is below the tritium MDA. The samples from the tunnel ceiling/reactor bottom (NMNT-50 through NMNT-53), however, had much higher levels of tritium and total beta activity than the floor. The estimated total beta activity for the four ceiling samples ranged from 0.54 to 2.9 uCi/g. All four samples also had tritium concentrations above the MDA. The maximum tritium concentration is estimated at about 0.056 uCi/g.

All of the concrete samples from the tunnel were analyzed for activation products using on-site gamma spectroscopy. Cobalt 60 was found to be present above detectable levels in one concrete and three soil samples from the tunnel. The maximum cobalt-60 concentration, 23.6 pCi/g was in sample NMNT-40 from the tunnel sump. The soil concentration at this location was 4.9 pCi/g. The highest cobalt-60 concentration in the soil, 13.9 pCi/g, was in sample NMNT-43 which was taken under the southern end of the tunnel. If the concrete and soil were activated in these locations, one would expect that samples from locations closer to the reactor would also be activated; however, they are not. Therefore, is suspected that the soil may have become contaminated with coolant containing activation products in this area.

Because of the high gross beta activity in the samples form the tunnel ceiling, three of the samples were sent off-site for nickel-63 and iron-55 analysis. In each sample, the nickel-63 concentration was less than the MDC (< 3.5 pCi/g). However, two of the three samples had detectable iron-55 concentrations up to 30 +/- 13 pCi/g. Because there was no measurable cobalt-60 or europium-152 in these samples, the iron-55 is assumed to be a contaminant that was transported to the area along with the tritium by leaking reactor coolant or Bulk Shielding Tank water.

4.4 REACTOR BIOSHIELD

Unlike the reactor room floor, the reactor bioshield is made up of high-density concrete that contains magnetite aggregate. This aggregate was evident in the fine concrete powder samples when a magnet was placed against the sample container. The lower thicker section of the bioshield contains about 200 cubic yards (cy) of high-density concrete (neglecting the volume of the penetrations). The upper section of the bioshield contains about 33 cy of high-density concrete.

The bioshield contains 11 penetrations in the form of 2 thermal columns, 7 beam ports, and 1 through port (counted as two separate penetrations). The bioshield also contains embedded items such as steel rebar, 4 steel shadow shields, and the support structures (angle irons, brackets, bolts, etc.) for the thermal columns, beam tubes, and shadow shields. There are many pipes, filters, railings, and other items mounted on the exterior of the bioshield.

4.4.1 Concrete Activation

Samples of the bioshield concrete were taken at several locations and at various depths on the reactor centerline and also at various elevations to establish a "radius of activation" in the horizontal and vertical planes. The sample points are shown in the figures in Appendix A-4. Each sample was analyzed for tritium and gross beta activity using a portable LSC and for gamma-emitting isotopes (activation products) using gamma spectroscopy. Table 4-4 provides a summary of the analytical results.

TABLE 4-4
ANALYSIS RESULTS – BIOSHIELD CONCRETE SAMPLES

Sample No.	Distance From Tank (cm)	H-3 (pCi/g)	Total Beta[a] (pCi/g)	Co-60 (pCi/g)	Eu-152 (pCi/g)
NMNT-20	< 5	6.21E+04	5.43E+05	8,390	9,010
NMNT-22	115	< MDA	<MDA	30.7[b]	33.1[b]
NMNT-23	85	< MDA	<MDA	< MDA	< MDA
NMNT-24	54	< MDA	1.43E+05	< MDA	< MDA
NMNT-25	60	< MDA	<MDA	< MDA	< MDA
NMNT-26	54	< MDA	<MDA	15.1	17.9
NMNT-27	91	< MDA	3.96E+05	1.96	< MDA
NMNT-28	58	< MDA	<MDA	8.08	7.13
NMNT-29	28	< MDA	1.37E+06	23.1	31.3
NMNT-30	13	< MDA	4.61E+05	66.7	59.1
NMNT-31	5	< MDA	1.53E+06	80.1	79.0
NMNT-32	91	< MDA	3.45E+05	12.6[b]	9.39[b]
NMNT-33	58	< MDA	6.71E+05	12.2	10.2
NMNT-34	28	< MDA	2.17E+05	58.9	52.6
NMNT-35	13	< MDA	3.43E+05	291	262
NMNT-36	5	< MDA	1.83E+05	533	534
NMNT-54	190	< MDA	< MDA	< MDA	< MDA
NMNT-55	160	< MDA	< MDA	< MDA	< MDA
NMNT-56	130	< MDA	< MDA	< MDA	< MDA
NMNT-57	100	< MDA	< MDA	< MDA	< MDA
NMNT-58	70	< MDA	3.13E+05	< MDA	< MDA
NMNT-59	40	< MDA	< MDA	6.40	4.44
NMNT-60	10	< MDA	2.16E+05	388	476
NMNT-61	190	< MDA	< MDA	< MDA	< MDA
NMNT-62	160	< MDA	< MDA	55.5[a]	60.2[a]
NMNT-63	130	< MDA	< MDA	< MDA	< MDA
NMNT-64	100	< MDA	< MDA	< MDA	< MDA
NMNT-65	70	< MDA	< MDA	17.5	10.9
NMNT-66	40	< MDA	2.55E+05	232	206
NMNT-67	10	< MDA	< MDA	3,550	3,890
NMNT-70	15 (62)[c]	< MDA	4.80E+05	203	241
NMNT-71	15 (46)[c]	< MDA	< MDA	265	181
NMNT-72	15 (62)[c]	< MDA	< MDA	26.8	15.9
NMNT-73	15 (76)[c]	< MDA	2.98E+05	47.4	38.7
NMNT-74	15 (91)[c]	< MDA	1.98E+05	6.02	5.16

Notes:
a Estimated based on tritium efficiencies.
b Potential cross contamination.
c (xx) = Vertical distance above the core centerline (cm).

Gamma spectroscopy identified cobalt-60, europium-152, and europium-154 as activation products in the concrete (europium-154 concentrations are not provided in Table 4-4. Complete results are summarized in Appendix B). Cobalt-60 was identified in concrete samples at concentrations less than 1 pCi/g. Typical cobalt-60 MDAs for a low-activity samples were 2 to 3 pCi/g. Europium-152 was identified at concentrations greater than 4.4 pCi/g with a typical low-activity sample MDA of about 3 pCi/g. When europium-152 was identified, it was typically at a concentration on the same order of magnitude as the cobalt-60 concentration. Europium-154 identified in only the highest activity samples at concentrations that were on average about 7% of the europium-152 concentrations (ranged form 5.7% to 13%).

The analytical results indicate that, on average, the concrete is activated or contaminated to a depth of about 36 inches (91.44 cm) from the reactor tank at the reactor centerline. However, the "radius of activation" is not expected to be symmetrical due to the various internal components in the bioshield such as the beam tubes, thermal columns, and shadow shields.

A vertical profile of the "radius of activation" was obtained by taking concrete samples near the tank wall at four different elevations 6 inches apart. These samples were taken from the Bulk Shielding Tank every six inches in the shielding wall that separates the reactor tank from the Bulk Shielding Tank. The concrete here is 36 inches thick. The samples were collected at a depth of 27 to 30 inches. The first sample was taken 1.5 feet (46 cm) up from the core centerline on the tank centerline. Because europium-152 was detected in the sample furthest from the reactor core, the concrete is known to be activated to a depth of 6 inches at an elevation of at least 3 feet from the core centerline. A linear approximation of the data indicates that activation would be near zero at an elevation of 4 feet above the core centerline at a depth of 6 inches.

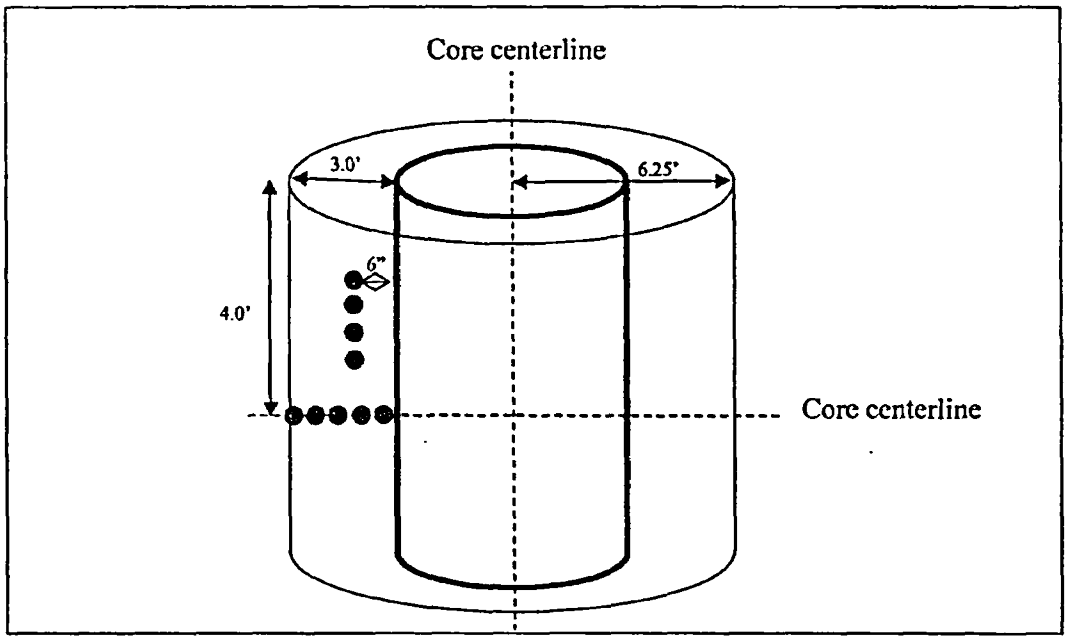
Therefore, assuming a constant "radius of activation" to 4 feet (122 cm) above and below the reactor centerline resulting in an annulus of activated concrete (see Figure 4-1), the volume of the activated concrete would be about 716 cubic feet (26.5 cy or 20.3 m^3) or about 71,700 kg. This volume estimate does not subtract volume for the beam tubes and thermal columns and does not include the concrete volume below the reactor tank. The volume also assumes that the radial extent of activation is constant with elevation; in reality this will not be the case.

The total in the activated bioshield concrete is estimated in Table 4-5. The iron-55 concentrations are based on the average ratio of cobalt-60 to iron-55 concentrations in samples NMNT-30 and NMNT-31. The iron-55 concentrations in these samples were 28.6 ± 7.5 pCi/g and 43.4 ± 8.1 pCi/g respectively.

TABLE 4-5
TOTAL ACTIVITY OF PRIMARY RADIONUCLIDES IN ACTIVATED BIOSHIELD CONCRETE

Isotope	Activity
Cobalt-60	35 mCi
Europium-152	30 mCi
Iron-55	17 mCi
Europium-154	2 mCi
TOTAL	84 mCi

FIGURE 4-1
APPROXIMATION OF ACTIVATED CONCRETE VOLUME



4.4.2 Concrete Contamination

The data in Table 4-4 indicate that the bioshield concrete also contains elevated concentrations of beta activity when gamma-emitting isotopes are less than MDA levels. This total beta activity could be from iron-55, tritium or other beta-emitting isotopes present in the bioshield. However, when both elevated total beta activity and gamma-emitting isotopes are present, they do not always follow the same activity profile with distance from the reactor tank indicating that the total beta activity may be an indicator of contamination rather than activation.

A sample of concrete from the bioshield taken very close to the reactor tank wall (NMNT-20) had high levels of tritium contamination. The tritium concentration in this sample was about 62,100 pCi/g.

The data on the concrete samples taken from the bottom of the bioshield is provided in Table 4-3. The maximum tritium concentration in the samples taken in this area was about 0.056 uCi/g and the maximum total beta concentration was estimated at 2.9 uCi/g. This data indicates that, while the concrete directly under the reactor is not significantly activated, it is significantly contaminated and will be LLRW. The volume of this concrete, not included in the volume calculated in Section 4.1.1, is about 2 m^3 (7,000 kg).

As described in Section 2.1, tritium is also an activation product that should be present at low levels along with other activation products in activated concrete. The water leaks in the reactor may have also mobilized the tritium without mobilizing other less soluble activation products and transported it to the low point of the bioshield. Water would have also likely traveled along the interface between the aluminum tank wall and the concrete. Section 4.4.5 discusses how embedded objects potentially caused contamination of the concrete.

4.4.3 Beam Ports

There are about 80 linear feet of beam ports embedded in the reactor bioshield. All of the beam port tubes, which are made of aluminum, are both contaminated and activated. The maximum activity level will be greater than the activity of the aluminum grid plate (see Section 4.5.2). The activity of the grid plate was 0.12 uCi/g cobalt-60 and 0.006 uCi/g europium-152. A radiological characterization of a TRIGA reactor at the Medical University of Hanover, Germany, reported that the cobalt-60 activity in the central beam tube near the reactor was approximately 6 times the cobalt-60 activity in the top grid plate (Hampel, et. al., 2001). Therefore, using this correlation, the maximum activity of the beam tube may be up to 1 uCi/g.

Scientech surveyed the through port beam tube through the access port on the eastern side of the reactor and the northern beam port using an ion chamber. The results of the surveys are provided in Appendix A-4. The maximum dose rate measured inside the through port beam tube was 10.8 R/hr at the center of the tube adjacent to the reactor core area. The dose rate at this point is a result of the beam tube as well as the adjacent reactor assembly components.

Iron-55 is likely also a major activation product of concern in the beam tubes. However, the reactor has been shut down for about 2.5 half-lives of iron-55 and another half-life will likely pass prior to decommissioning. Based on the isotope ratios of the German reactor beam tube (Hampel, et. al., 2001), following the decay time, the iron-55 concentration at the time of decommissioning will likely be about twice the cobalt-60 concentration.

Because of the presence of unmoderated neutrons in the beam tubes, activated metal likely extends to beyond the concrete activation radius of 30 inches. Removable contamination is present in the beam tubes in the form of metal oxides on the beam tube surfaces.

The contamination survey results for pieces of beam port plugs that were removed are provided in Appendix A-4.

4.4.4 Thermal Columns

The large thermal column, which penetrates the northeastern bioshield and core reflector assembly, is 4 feet square by 5.5 feet in length. The outer thermal column is covered by the 3.5-foot thick heavy concrete door. The graphite in the thermal column is stacked block contained in a welded aluminum can, fabricated of 0.5-inch thick sheets of aluminum and lined with 0.125-inch-thick boral sheets. A 4-inch thick block internal lead shield (stacked bricks) is positioned about 50 inches from the outside surface of the column. During the site characterization, the graphite blocks in the large thermal column were removed back to the lead wall. The graphite was removed to allow sampling of the graphite and the allow access to additional bioshield sampling points. The density of the graphite is about 1.76 g/cm^3.

All of the graphite removed from the thermal column is activated. Four samples were collected and analyzed using an LSC and Scientech's gamma spectroscopy system. Table 4-6 provides the activity levels of cobalt-60, europium-152, and europium-154 in graphite samples.

A smaller thermal column (thermalizing column), 2-feet square and 4.33 feet long, extends from the north wall of the Bulk Shielding Tank to the core reflector assembly. The column contains a welded aluminum can, fabricated of 0.5-inch thick sheets of aluminum and lined with 0.125-inch-thick boral sheets. A 0.675-inch thick aluminum cover plate separates the Bulk Shielding Tank and the thermal column.

There is about 8 cubic feet (0.23 m^3 or 400 kg) of graphite in the small thermal column located outside the wall of lead bricks. Because there is only air between the lead bricks and the graphite, the activation products should be present in this volume of graphite at similar concentrations to the graphite removed from between the boral curtain and the lead bricks in the large thermal column. The total activity of this volume of graphite is estimated in Table 4-8 assuming that the average activity per gram is equivalent to the average activity of the graphite in Stack 2 provided in Table 4-6.

Because nuclear-grade graphite was used in the thermal column, it is expected to have a similar composition as the graphite blind elements samples as part of the Hanover characterization. The Hanover samples had a europium-152 to carbon-14 ratio of approximately 1.3 to 1 (Hampel, et. al., 2001). As such, the estimated the carbon-14 activities are included in Table 4-6. The estimated total activity of the graphite removed from the thermal column is provided in Table 4-7. The total mass of the graphite in the two thermal columns is about 4,490 kg (excluding the graphite inside the lead brick dividing walls). The additional mass of the graphite beyond the lead dividing walls and the reactor core assembly is about 1,000 kg.

TABLE 4-6
GRAPHITE SAMPLE ACTIVITIES

Sample (size)	Distance from outside surface of thermal column	Isotope	Activity (uCi/g)
Graphite powder (18 grams)	48 inches (centerline)	Co-60	1.05E-04
		Eu-152	1.72E-02
		Eu-154	1.20E-03
		C-14	1.28E-02 (J)
Graphite powder (22 grams)	40 inches (centerline)	Co-60	3.56E-05
		Eu-152	5.43E-03
		Eu-154	3.35E-04
		C-14	4.02E-05 (J)
Graphite powder (17 grams)	34 inches (centerline)	Co-60	1.19E-04
		Eu-152	1.09E-04
		C-14	8.09E-05 (J)
Graphite powder (23 grams)	2 inches (centerline)	Co-60	4.91E-06
		Eu-152	1.46E-05
		C-14	1.08E-05 (J)
Graphite block (5.88 kg)	38 to 50 inches (centerline)	Co-60	1.29E-04
		Eu-152	1.68E-02
		Eu-154	1.09E-03
		C-14	1.25E-02 (J)
Graphite Stack 1 (4' x 4' x 4') (3,190 kg)	Volume outside the boral curtain	Co-60	7.64E-07
		Eu-152	2.21E-04
		Eu-154	1.34E-05
		C-14	1.64E-04 (J)
Graphite Stack 2 (3' x 3' x 2') (900 kg)	Volume inside the boral curtain	Co-60	4.1E-07 (UJ)
		Eu-152	3.54E-03
		Eu-154	2.27E-04
		C-14	2.63E-03 (J)

J – Carbon-14 activity estimated from europium-152 activity.
UJ – less than MDA; estimated based on the peak area of the cobalt-60 1332 keV peak.

TABLE 4-7
TOTAL ESTIMATE ACTIVITY OF GRAPHITE REMOVED FROM THE LARGE THERMAL COLUMN

Isotope	Activity
Cobalt-60	0.003 mCi
Europium-152	3.89 mCi
Europium-154	0.25 mCi
Carbon-14	2.89 mCi
TOTAL	**7.03 mCi**

TABLE 4-8
TOTAL ESTIMATED ACTIVITY OF GRAPHITE FROM THE SMALL THERMAL COLUMN POSITIONED OUTSIDE THE LEAD BRICKS

Isotope	Activity
Cobalt-60	0.0002 mCi
Europium-152	1.43 mCi
Europium-154	0.0921mCi
Carbon-14	1.05 mCi
TOTAL	**2.56 mCi**

Approximately 3 feet from the outside surface is a 46-inch by 49-inch boral curtain fabricated from a single 0.25-inch thick boral plate. One sample was taken from the boral curtain and analyzed using gamma spectroscopy. The sample contained 0.322 uCi/g cobalt-60 and 0.013 uCi/g europium-152.

Each thermal column is contained within with 0.5-inch thick aluminum sheets lined with a 0.125-inch boral layer. The entire volume of metal should be considered activated. The total volume of the metal liners and the boral curtain is 6.8 cubic feet or about 526 kg. Using the above activity for the boral curtain sample and applying it to the entire volume of aluminum and boral in the thermal columns, a very conservative estimate of the total activity of the metal is 169 mCi cobalt-60 and 6.8 mCi europium-152.

4.4.5 Embedded Items

Several items of concern that are embedded in the concrete bioshield include:

- Steel shadow shields
- Steel rebar
- Angle irons and supports for shadow shields and thermal columns
- Aluminum pocket for the boral curtain (where it goes when it is raised)
- Off-gas sample lines connected to the thermal columns and the 4 original beam ports

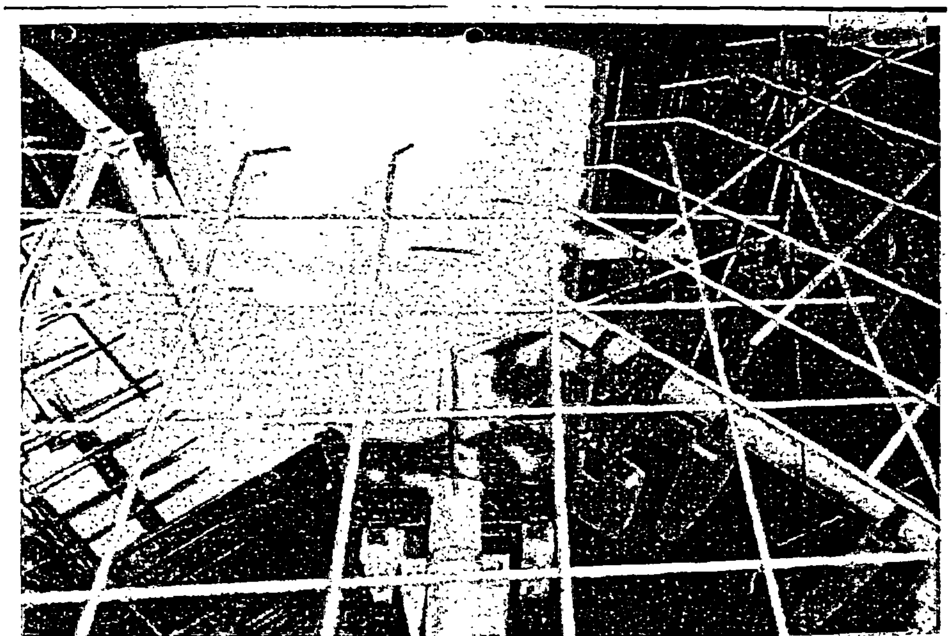
Figure 4-2 provides images of the embedded items in the bioshield before the concrete was poured. In image (a), the large thermal column is on the left with the support system and boral curtain pocket visible. In image (b), the small thermal column and off-gas sample lines are visible on the right side. The steel shadow shields are visible in both photographs.

No samples were collected of the embedded items. However, concrete sample NMNT-26 possibly contained some metal particles from a shadow shield. At this location, the sample was collected until the drill bit came into contact with the shadow shield. Concentrations of cobalt-60 and europium-152 in this sample were greater than 10 pCi/g. This sample point is likely outside of the radius of detectable concrete activation and, therefore, the activity is assumed to have come from the shadow shield.

The shadow shields make up the majority of the mass of the embedded items that are potentially activated because they are located about 2 feet from the reactor tank wall near the extent of the concrete radius of activation. Activated steel will contain mostly iron-55 [about 90% of the total activity (Hampel, et. al., 2001)], cobalt-60, and a small amount of nickel-63. The combined mass of the four shadow shields is about 590 kg.

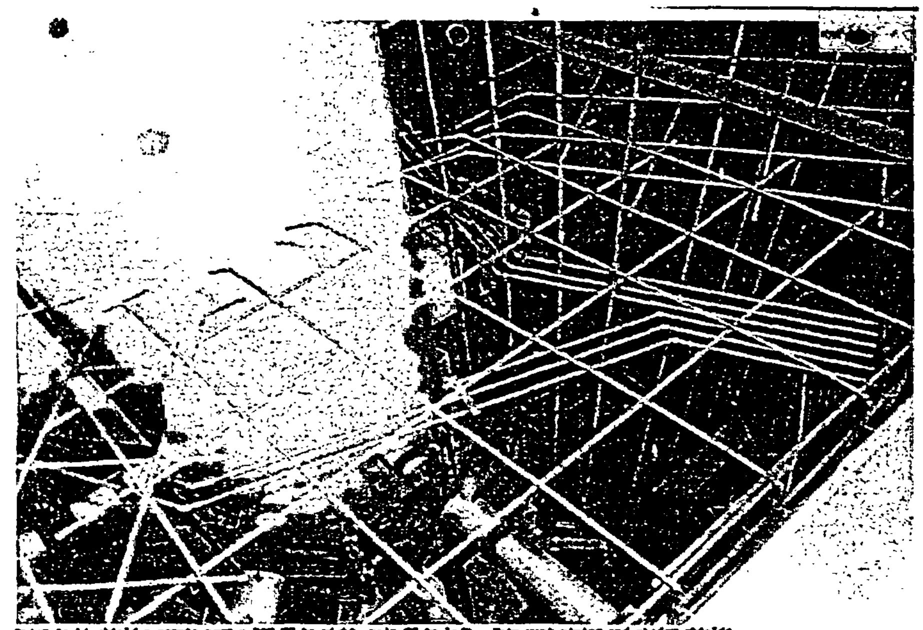
The embedded steel objects, such as the shadow shields and rebar, have oxidized to some degree inside the concrete due to normal moisture content of the concrete as well as to additional water traveling through the concrete due to leaks. The moving water had the potential to mobilize some of the soluble contaminants on the oxidized surfaces resulting in the spread of activation products with the water. For example, sample NMNT-29 taken about 1.5 feet from the reactor tank wall to the right of the small thermal column had an iron-55 concentration of 522 +/- 69 pCi/g. The next two samples taken at locations 6 inches and 1 foot closer to the tank wall along the same sampling line (NMNT-30 and NMNT-31) had iron-55 concentrations of 28.6 +/- 7.5 pCi/g and 43.4 +/- 8.1 pCi/g respectively. The higher activity at a greater distance from the core centerline is likely a result of either nicking a piece of steel rebar or thermal column support with the drill bit and collected a small amount of activated metal in the concrete sample. The elevated iron-55 activity may also be a result of passing the drill bit very near a piece of activated steel that has impacted the surrounding concrete.

FIGURE 4-2
PRE-POUR PHOTOGRAPHS OF UNIVERSITY OF ILLINOIS REACTOR



Reactor tank with beam ports showing - rebar in place for bioshield. Note shadow shields on beam ports, and vent piping

(a)



Prior to bioshield concrete pour - BPT TC to right, main TC to left. Note vent piping and shadow shields.

(b)

The cobalt-60 to iron-55 ratios in the activated concrete samples NMNT-30 and NMNT-31 were 1.5 and 2.8 respectively. Detectable iron-55 concentrations of up to 30 +/- 13 pCi/g were also found in the concrete samples taken from below the reactor from the tunnel. These samples, however, were high in tritium and were void of detectable cobalt-60 and europium-152. This indicates that the concrete is not activated but contaminated.

4.5 INTERNAL COMPONENTS

The bioshield encloses the aluminum reactor tank and the tank contains the reactor core assembly. The reactor core assembly, several other reactor components, and miscellaneous support equipment are still in the reactor tank. Support equipment includes two control rods, five ion chambers, one fission chamber, and submersible lights.

4.5.1 Reactor Tank

The open-top aluminum reactor tank is 0.25-inches thick with an outer diameter of 6.5 feet and 20.5 feet deep. The aluminum has a density of 2.72 g/cm^3. The mass of the tank is about 13,150 kg. In order to obtain aluminum samples that would be representative of the tank wall, a vertical pipe of the emergency spray cooling system located on along the south side of the interior tank wall was removed and sampled. The pipe was disconnected from the spray ring at the first pipe elbow from the top of the tank. Table 4-5 provides the europium-152 results of the sampling and on-site analysis. The cobalt-60 activities are not provided because the samples may have been cross-contaminated with activated stainless steel from sampling a stainless steel screw just prior to sampling the pipe. The positions in Table 4-9 are the approximate locations up the tank wall from the location of the spray ring, which was positioned just above the top grid plate.

TABLE 4-9
PREDICTED TANK WALL ACTIVITY PROFILE

Position	Isotope	Activity (uCi/g)
14 inches	Eu-152	6.85E-03
22 inches	Eu-152	1.53E-04*
30 inches	Eu-152	9.62E-05
38 inches	Eu-152	6.51E-05

Note: * - This is the MDA for Eu-152 for the sample. The analysis of this sample only identified cobalt-60 due to high level of cobalt-60 in the sample from the cross contamination with the activated stainless steel. The cobalt-60 level in the other samples was sufficiently low to allow identification of the europium-152 peaks.

An exponential interpolation of the data in Table 4-9 predicts that the reactor tank will be activated to a height of about 58 inches above the tank bottom where the europium-152 activity will be about 1.0E-6 uCi/g. The mass of the activated portion of the tank is about 212 kg, assuming the entire tank below the top grid plate is activated.

4.5.2 Reactor Core Support and Assembly

The core assembly currently consists of core support (plenum), the control rods safety plate, the reflector, and grid plates. The core support is a cylindrical aluminum structure bolted to a platform on the bottom of the reactor tank. The core support is 31.5 inches high and 30 inches in diameter. The aluminum control rod safety plate is located below the core support. Surrounding the core support is the aluminum primary coolant return spray header.

The reactor core assembly contains about 500 kg of activated aluminum including two aluminum grid plates in the core assembly. The top grid plate is 1 inch thick and the bottom grid plate is 1.25 inches thick. The top grid plate has three removable sections. The grid plates are about 2 feet in diameter. Table 4-10 provides the cobalt-60 and europium-152 activities of two samples from the top grid plate and the estimated iron-55 and nickel-63 activities. The estimated iron-55 and nickel-63 concentrations are based on percent of total activity of the aluminum components from the Hanover reactor with adjustments for radioactive decay (Hampel, et. al., 2001). As described in Section 4.4.3, following the decay time, the iron-55 concentration at the time of decommissioning will likely be about twice the cobalt-60 concentration or about 65% of the total activity. Nickel-63 makes up about 1% of the total activity.

The header spray ring, a perforated 10-inch aluminum pipe is located at the bottom of the reactor tank. The outer diameter of the ring is about 48 inches.

Assuming that the aluminum components of the reactor assembly have an average activity equal to the activity of the center removal portion of the top grid plate as given in Table 4-10, the total activity of the primary isotopes of concern of the reactor assembly would be about 115 mCi of iron-55, 57.5 mCi of cobalt-60, 1.75 mCi nickel-63, and 3.2 mCi of europium-152.

TABLE 4-10
ACTIVITY OF TOP GRID PLATE SAMPLES

Sample Location	Isotope	Activity (uCi/g)
Center removable piece	Co-60	1.10E-01
	Eu-152	6.35E-03
	Fe-55	2.2E-01
	Ni-63	3.4E-03
Side removable piece	Co-60	1.21E-01
	Fe-55	2.4E-01
	Ni-63	3.6E-03

4.5.3 Rotary Specimen Rack

Underwater dose rate surveys were conducted to estimate the cobalt-60 activity of the Rotary Specimen Rack (a.k.a. Lazy Susan). The Lazy Susan is made of aluminum and contains a drive chain with Stellite bearings. Prior to activation, Stellite contains mostly stable cobalt. As such, most of the activity in the Lazy Susan will be cobalt-60 from the Stellite bearings and the aluminum housing. The aluminum housing would also have iron-55, cobalt-60 and europium-152 activities similar to those provided in Table 4-6 for the top grid plate. Assuming the mass of the aluminum in the Lazy Susan to be about 25 kg, the cobalt-60 activity of the aluminum housing would be about 2.75 mCi.

The Lazy Susan geometry was modeled in MicroShield and the cobalt-60 activity was adjusted until the modeled dose rate was equivalent to the measured dose rate. The results indicate that the total cobalt-60 activity of the Lazy Susan is about 4 Ci. An alternate model that summed up forty 0.1-Ci point sources (for a total of 4 Ci) resulted in approximately the same measured dose rate. The MicroShield analysis results are provided in Appendix A-5.

4.5.4 Graphite Reflector

The graphite reflector surrounding the core consists primarily of a circular block of graphite with a hexagonal center cutout. The graphite reflector has an outer diameter of 41.675 inches with a radial wall thickness of 8.6 to 10.3 inches and a height of 22 inches. A 2.25-inch-thick annulus of lead surrounds the graphite. The lead shield and graphite are cut out at the beam port penetrations. The reflector and lead assembly is sealed in a welded aluminum container. The reflector and core support are bolted together with stainless steel bolts.

The graphite was not sampled but is certainly activated with high levels of europium-152, europium-154, carbon-14. and cobalt-60. Due to the close proximity of the reflector to the fuel, the concentrations are expected to be much greater than the concentrations provided in Table 4-6.

4.6 EXTERNAL SYSTEMS

Many of the reactor support systems are located outside the reactor tank. The majority of these systems are associated with the primary and secondary coolant systems. Other external systems include the waste water and air handling systems.

4.6.1 Coolant Pipes

The primary coolant water flows into and out of the reactor tank through two 10-inch Schedule 40 aluminum pipes located in the pipe tunnel. Coolant water passes through the outlet pipe to the nitrogen-16 decay tanks and returns to the reactor from the heat exchanger.

The coolant pipe tunnel was surveyed with an exposure rate meter as described in Section 4.3.1. The survey identified the pipe elbows below the reactor core as having high gamma exposure rates. It is most likely that the pipes are internally contaminated with loose deposits from the reactor tank that have settled into the pipe elbows since the flow was shut off. Since sediments have collected in the elbows, it is possible that loose deposits have collected in other sections of these pipes and they should be treated as internally contaminated. While not measurable with external gamma surveys, contaminated deposits may have also collected in the pipe valves.

4.6.2 Nitrogen-16 Decay Tanks

There are two nitrogen-16 decay tanks located in vault under the Mechanical Equipment Room. The large tank has a 3,000 gallon capacity while the smaller tank has a 1,500 gallon capacity. Water flows from the reactor through the pipe described in section 4.6.1 to the larger decay tank. The large tank overflows into the smaller tank. From the small tank, the water is pumped to the heat exchanger using the primary pump.

Gamma surveys inside the nitrogen tank vault indicated that the tanks were internally contaminated. A NaI gamma spectroscopy system was used to identify cobalt-60 as the primary gamma-emitting contaminant. It is suspected that the tanks are contaminated with deposits from the reactor. The gamma survey indicated that the large tank was contaminated to a greater extent than the smaller tank. Three removable contamination smears were taken on the tanks. The analysis of the smears showed no removable tritium, carbon-14, or total beta activity.

At the time of the facility characterization, there was about 2 to 3 inches of standing water in the nitrogen-16 decay tank vault. Scientech collected two samples of this water and analyzed them for tritium and gross beta in the University's LSC. The results of the analysis, presented in Table 4-11, indicated that the water in the vault was not radiologically contaminated.

TABLE 4-11
WATER SAMPLE RESULTS FROM N-16 TANK VAULT

Sample (10 ml)	Tritium	Carbon-14
N-16-1	< MDC	< MDC
N-16-2	< MDC	< MDC

Tritium MDC = 1.7 pCi/ml
Carbon-14 MDC = 3.6 pCi/ml

4.6.3 Primary Pump and Heat Exchanger

Primary coolant water was pumped from the small nitrogen-16 decay tank to the heat exchanger using the primary pump. The pump and heat exchanger are located in the Mechanical Equipment Room south of the reactor.

A gamma radiation survey conducted on the equipment in the Mechanical Equipment Room is provided in Appendix A-5. The survey indicates that there are recognizable differences in the gamma count rates on the primary pump (6,499 cpm) and secondary pump (2,907 cpm) which may be an indication of some internal contamination in the primary coolant pump.

The gamma count rates decreased from 5,181 cpm to 3,581 cpm while moving down the heat exchanger in the direction of the primary coolant flow, from north to south. This may be an indicator of some internal contamination, but the data are not conclusive, because, moving from north to south, the detector is also moving away from a higher background area outside the Mechanical Equipment Room. Background count rates in the Mechanical Equipment Room were much lower than in other area of the NRL facility.

4.6.4 Secondary Pump and Cooling Towers

The secondary coolant loop is completely isolated from the primary coolant loop and has no contact with contaminated or activated components. The secondary coolant loop is driven by the secondary pump which is also located in the Mechanical Equipment Room. The secondary pump feeds water through the heat exchanger to the cooling towers located outside the reactor building. Water then flows back from the cooling towers to the secondary pump.

The survey discussed in Section 4.6.4 included gamma radiation measurements taken on the secondary pump. The measurements did not indicate that the pump is internally contaminated.

Appendix A-5 contains a survey form for the surveys conducted on the cooling towers. The data indicate that there is no significant beta-gamma contamination on the cooling towers. There were, however, some low levels (less than 1,000 dpm) of detectable removable tritium contamination on the side vents.

The cooling tower manufacturer, The Marly Company, indicates that the cooling towers contain asbestos. The manufacturer's specifications state that corrugated asbestos cement board casings are located on the broad side of the cooling towers (north and south sides) and corrugated asbestos cement board air inlet louvers are located on the narrow sides of the cooling towers (east and west sides). An asbestos survey is described in Section 5.0. This survey failed to identify the asbestos in the cooling towers. A copy of the manufacture's information is provided in Appendix A-5.

4.6.5 Waste Water System and Filters

On the mezzanine floor, the two restrooms and the janitor's closet are the only rooms requiring drainage. Vent pipes are provided for this system which drains directly into an existing 24-inch sanitary sewer line. Prior to May 1996, four floor drains, one trench drain, the clean-up sink, and the decontamination shower on the reactor floor level all drained into a 500-gallon retention tank located below the floor on the western side of the reactor floor. Waste in the tank would be sampled prior to being discharged directly into the existing 24-inch sanitary sewer line. Besides clean-up water, any radioactive solutions from experiments were put into this system.

This 500-gallon tank received an external gamma survey. The record of the survey is provided in Appendix A-5. No elevated gamma radiation levels were detected around the tank, including the bottom of the tank where contaminated particulate would have settled. Gamma radiation count rates did increase as the detector got closer to the vault ceiling which put the detector closer to the Radioactive Materials Storage Cage. Elevated direct beta and alpha measurements are likely due to the presence of elevated radon and radon daughters in the sub-floor level vault. However, because of the years of use and the known fact that the tank received water containing tritium, the tank may be internally contaminated with low levels of radioactive material not detectable from outside the tank.

Once the 500-gallon tank became filled to capacity, the waste water was pumped over to a polyethylene holdup tank located on the reactor room floor. The water passed through a course and a fine filter assembly en route to the holdup tank where it was then sampled. When the soluble activity results were satisfactory and it was verified that no insoluble activity was present, the water was discharged from the holdup tank into the sanitary sewer system. If insoluble activity was detected before the discharge then the contents of the holdup tank were recirculated through a 0.4 micron process filter until the insoluble activity was removed and it was verified that no insoluble activity was present.

Because the polyethylene holdup tank and its associated filters were located such close proximity to the Radioactive Materials Storage Cage, the background radiation level was too high for meaningful direct radiation surveys of these pieces of equipment.

Appendix A-5 provides a survey form providing measurements taken on these pieces of equipment and other water filters. The net gamma count rates reported for measurements taking on filters, however, were calculated using an area-specific background. That is, background measurement was made away from but near the filter and then the gamma measurement was taken in contact with the filter. These measurements indicate that the following filters contain contaminated filter media:

- Filter #6 – large filter located near the top of the Bulk Shielding Tank
- Filter #8 – small filter small filter located north of the Radioactive Materials Storage Cage
- Filter #15 – while no data was recorded for the large filter located under the southern stairway leading down to the reactor floor, gamma surveys indicated that the filter contained radioactive materials.

A volume of filter resins stored in a plastic trash can located under the southern stairway leading down to the reactor floor next to Filter #15 was sampled. A 500 ml sample (NMNT-21A) was analyzed using on-site gamma spectroscopy. The analytical report, which is provided in Appendix B, shows that the resin contains 48 pCi/g cobalt-60. The total volume of contaminated filter resin is less than 55-gallons.

4.6.6 Bulk Shielding Tank

The Bulk Shielding Tank is 12 feet deep, 8 feet wide, and 9 feet long. The south wall of the tank heavy concrete about 1 foot thick. The floor of the tank is about 2 feet thick. The east and west walls are part of the reactor bioshield and have varying thickness from about 1 foot to more than 4 feet. The north wall, which is adjacent to the reactor and is also part of the bioshield, is 3 feet thick.

The Bulk Shielding Tank allowed for neutron beam experiments to be conducted underwater for additional shielding. The Bulk Shielding Tank was also the home of the LOPRA which was a subcritical assembly that used TRIGA fuel. The Bulk Shielding Tank was additionally used for wet fuel storage from 1998 to 2004.

Three concrete samples were collected from the floor of the Bulk Shielding Tank, NMNT-37, NMNT-38, and NMNT-39. The samples were taken from the top inch of concrete. The results of the gamma spectroscopy analysis of the samples show that the concrete floor is at least partially activated. The gross tritium and gross beta count rates measured using the portable LSC also show potential tritium contamination and gross beta concentrations well above background levels. The maximum estimated total beta activity was more than 1.19 uCi/g. The gamma spectroscopy and gross beta data are summarized in Table 4-12.

Appendix A-5 contains a survey form documenting the removable contamination survey conducted in the Bulk Shielding Tank. The survey form also provides the gamma exposure rates which ranged from 320 to 3,200 uR/hr. Because of the high exposure rates in the Bulk Shielding Tank, direct surface contamination measurements were not taken. The removable contamination data indicate that there is some removable beta contamination on the floor of the Bulk Shielding Tank. The maximum removable contamination activities are provided in Table 4-13.

TABLE 4-12
ACTIVITY OF BULK SHIELDING TANK FLOOR CONCRETE SAMPLES

Sample Location	Isotope	Net Activity
NMNT-37 (at the location of the former LOPRA)	Co-60	2.54E-04 uCi/g
	Eu-152	8.15E-05 uCi/g
	Eu-154	1.06E-05 uCi/g
	Total Beta[a]	1.19 uCi/g[a]
	Tritium	< MDA
NMNT-38	Co-60	8.33E-05 uCi/g
	Eu-152	8.91E-06 uCi/g
	Total Beta[a]	0.264 uCi/g[a]
	Tritium	<MDA
NMNT-39	Co-60	2.86E-06 uCi/g[b]
	Total Beta[a]	0.957 uCi/g[a]
	Tritium	0.002 uCi/g

Notes: a Estimated based on tritium efficiencies.
b Less than the MDA of 4.12E-06 uCi/g.

TABLE 4-13
MAXIMUM REMOVABLE ACTIVITY IN BULK SHIELDING TANK

Surface	Tritium ($dpm/100cm^2$)	Carbon-14 ($dpm/100cm^2$)	Gross Beta[a] ($dpm/100cm^2$)
Floor	207	625	1,196
Wall	42	< MDA	48

Note: a Activity based on an efficiency of 0.75, equal to the carbon-14 efficiency.

4.6.7 Air Handling Systems

Air flow in the reactor building was managed by two separate systems. The heating and air conditioning system used outside air to heat and cool the reactor room and the mezzanine area. The air conditioning unit is positioned on an elevated platform in the northeastern corner of the facility.

The entire building is exhausted through the building ventilation system that is located on the storage floor. The system contained a 5-micron pre-filter section and a special 0.3 micron HEPA filter section. An activated charcoal filter is located down stream from these filters in a duct parallel with the main exhaust duct. A damper in the duct containing the charcoal filter is normally closed. In case of a fission product release containing iodine isotopes, the damper in the main exhaust duct was closed automatically and the damper in the duct containing the charcoal filter would be opened, thus forcing the air through the activated charcoal filter.

The blower, which is located on the roof, is capable of changing the building's air in about one hour. The exhaust is discharged through a 15-foot stack at a flow rate of 2,400 cubic feet per minute to a height of about 60 feet above grade.

Both the air conditioning unit and the HEPA filter back were surveyed. The survey reports are provided in Appendix A-5. The high direct alpha and beta measurements are most likely attributable to the presence of radon daughters present on particulate in the air conditioner and HEPA filters. The air handling systems are not believed to have been impacted from reactor operations or on-site experiments.

4.7 OTHER EQUIPMENT

As part of the characterization project, other pieces of equipment were surveyed. The list of equipment surveyed is provided below. Some of the equipment listed below may be removed from the reactor facility prior to the initiation of the facility decommissioning.

- Large glove box
- Reactor top/bridge
- Work benches on reactor room floor (2)
- Retractable shielding door (exterior surface)
- Thermo-shield (front and back surfaces)
- Beam catchers (4)

Survey forms are provided in Appendix A-6 for each of the above. Table 4-14 provides a summary of the most significant findings of these surveys.

Surveys of the large glove box indicate that the glove box is both internally and externally contaminated with alpha and beta-gamma isotopes. It is suspected that the alpha contaminant is uranium based on knowledge of activities conducted in the glove box. The high direct beta measurements should be rejected due to the high gamma exposure rates in the area.

The equipment and supplies on the top of the reactor were surveyed. The background level was too high for meaningful direct measurements. Removable tritium activity was low with a maximum 59 dpm/100cm^2. The highest removable contamination sample for total beta, about 15,000 dpm/100cm^2, was taken under the steel plate of the reactor bridge. The plate covers the opening of an irradiation sample guide tube. The next highest removable contamination measurement of 1,600 dpm/100cm^2 total beta was taken in a drawer on the left side of the workbench used to handle irradiated samples. There were also many contaminated objects located on and around the workbench including plastic trays, sample tubes, and metal buckets.

Surveys were conducted on two work benches on the reactor room floor. One bench is located in the northwestern corner of the room. This workbench was surveyed because radioactive sources are stored and used in this area. Three direct surface contamination measurements were made on the top of the workbench. One of the three points indicated elevated activity but it was determined that the activity was from a sealed radioactive source stored in a drawer. Ten removable contamination smears were taken and sampled for tritium and gross beta with activities less than or close to the MDA.

The second work bench is located in the southeastern corner of the reactor room. This workbench was surveyed because material labeled as contaminated was currently stored on top of the work bench. Ten removable contamination smears were taken and sampled for tritium and total beta with no reportable activities reported in all but two of the smears. One smear, taken on the surface of window of the small glove box, had a reported removable total beta activity of 619 dpm/100cm^2. The second location with measurable removable contamination was inside a drawer in the center of the workbench. The removable total beta activity at this location was 201 dpm/100cm^2.

The survey of the retractable shield door and the movable thermo-shield (located on the same track as the shield door) indicated that the thermo-shield is contaminated or activated. The direct beta measurements on the thermo-shield ranged from less than the survey MDA to more than 11,800 dpm/100cm^2. Measurements on the shield door were close to or less than the survey MDA.

While included in the list above, beam catchers are not actually pieces of equipment. They are hollow concrete pipes cast into the footing of the building across from the four original reactor beam ports. These beam catchers have a 2-foot inner diameter and are set 6 feet into the earth outside the footing. The concrete pipes are 2 to 3 inches thick.

While not free-standing equipment, the four beam catchers are included in this section. Because the purpose of the beam catchers was to catch neutron beams so that they would not be scattered throughout the reactor room by a flat wall, there is a potential for activation of the concrete. The beam catchers were used not only for their designed purpose, but they were also used after the reactor shut down for storing radioactive and contaminated materials.

The survey of beam catcher #1, located across from the southeastern beam port, indicated that the surface of the beam catcher is contaminated. The surveys of beam catchers #2 and #3 showed no significant direct surface activity. The removable tritium levels in beam catcher #2, however, were as high as 425 dpm/100cm^2. During the site characterization, radioactive materials were being stored in beam catcher #4 across from the northern beam port. As a result, the direct measurements should be rejected due to the high background radiation levels in the beam catcher.

TABLE 4-14
SIGNIFICANT FINDINGS OF OTHER EQUIPMENT SURVEYS

Item Surveyed	Survey Type	Radiation	Activity (dpm/100cm^2)
Large Glove Box (exterior)	Direct	Alpha	575
Large Glove Box (interior)	Removable	Alpha Gross Beta	3717 4481
Reactor Bridge	Removable	Gross Beta	11,212
Workbench on reactor top	Removable	Gross Beta	1,205
Southeastern Workbench	Removable	Gross Beta	464
Thermo Shield	Direct	Beta/gamma	11,708
Beam Catcher #1	Direct	Beta/gamma	9,547
Beam Catcher #2	Removable	Tritium Gross beta	425 217

4.8 SOIL

Fifty-six subsurface soil samples from 12 outside locations and 5 interior locations, along with four background soil samples, were collected and analyzed off-site by STL for tritium. The results indicate that nearly all samples are at background levels. The only anomalies are the samples from interior locations 3 and 12. At location 3, southwest of the reactor bioshield, the 0-to-1 and 3-to-4 foot sample results were greater than the detection limits, yet less than 1 pCi/g. At location 12, southeast of the reactor bioshield, the 0-to-1 and 6-to-7 foot sample results were greater than the detection limits, and slightly greater than 1 pCi/g. A summary table of the soil data and a sampling of the STL data reports are provided in Appendix A-7. The complete data packaged is provided as Appendix C.

Table 3 of the "Supplemental Information of the Implementation of the Final Rule on Radiological Criteria for License Termination" (Federal Register 1998) provides a soil screening level of 110 pCi/g for tritium based on a 25 mrem/yr dose. Therefore, when compared to the soil screening value, the concentration of tritium in soil under the reactor is very small and insignificant from a dose standpoint. The location of the only samples that were above the detection limit may indicate that the soil under the reactor may have been slightly impacted from historical water leaks.

5.0 HAZARDOUS MATERIALS ASSESSMENT

As part of the NRL facility characterization, a hazardous materials assessment was also performed. The goal of the assessment was to identify known and potential hazardous materials present at the facility. The assessment was based on site observations and document reviews.

During the on-site characterization, several hazardous materials were identified. These included caustics and acids (water treatment chemicals) and lubricating oils. There appeared to be no evidence of spill or leaks of these materials that would have impacted the facility. Because the NRL facility housed many experiments, there may also be containers of laboratory chemicals stored about the facility that were not observed during the radiological site characterization activities. It is expected that containers of hazardous materials will be disposed of according to current practices and procedures prior to radiological decontamination and demolition.

The site Asbestos Management Plan Report (Industrial Hygiene Services 2000) was reviewed for the presence of known asbestos containing materials (ACM). The findings presented in the plan are summarized in Table 5-1. Table 5-1 also list the ACM identified through other sources.

TABLE 5-1
ASBESTOS INSPECTION FINDINGS

Description	Quantity	ACM		
		Assume	Positive Test	Negative Test
Pipe joints on Magblock pipe insulation	21 each		X	
Pipe joints on cardboard pipe insulation	5 each		X	
Magblock pipe insulation	410 linear feet		X	
Cardboard pipe insulation	95 linear feet		X	
Magblock tank insulation	31 square feet		X	
9x9 mastic floor tiles	700 square feet	X		
Mastic under vinyl baseboards	70 square feet	X		
Built-up roof	3,200 square feet	X		
Gaskets	38 each	X		
2x4 fissured ceiling tiles	175 square feet			X
Cooling tower casing[a]		X		
Cooling tower louvers[a]		X		

Note: a Not identified in the Asbestos Management Plan Report; per manufacture's product specifications.

Lead is present throughout the facility in the form of lead bricks and lead sheets. About 50 lead bricks are also contaminated with cesium-137. These bricks were sorted from the non-contaminated bricks during the site characterization. While a lead paint survey was not conducted, because of the age of the NRL facility, it is expected that lead-based paints were used throughout the facility.

Mercury is present at the facility in several thermometers stored in a desk in the control room. The fluorescent light ballasts and any old thermostats present will also likely contain mercury. There is a significant amount of old experiment equipment present at the NRL facility that, if still on site prior to radiological decontamination and decommissioning, may contain mercury thermometers and switches.

6.0 LIMITATIONS

Because of time and budgetary constraints, not all potential radionuclides of concern could be analyzed for in each segment of the facility characterization. Limitations on the data presented in this report are described in the following sections.

6.1 PORTABLE LSC DATA

There are several limitations to the data obtained from the analysis of concrete and soil sample analyzed on the portable LSCs. First, the MDA for tritium is very high, on the order of 10,000 to 20,000 pCi/g. This is due to the small sample size (0.1 grams) and the low counting efficiencies (8% and 15%) caused by the suspended solid material sample. Typical LSC counting efficiencies for tritium in clear liquids are greater than 50%.

For additional characterization information, the total beta count rate was also recorded from the portable LSCs. For a conservative approach, the same tritium efficiencies were applied to the total beta activities in order to obtain an approximation of the total beta activity in pCi/g. An efficiency for total beta should, in fact, be significantly higher than the tritium efficiency because all detector channels are being counted, not only those in the tritium window.

There was no on-site methodology for identifying the specific radionuclides contributing to the total beta activity. However, much of the activity is likely to be from tritium betas outside the energies of the tritium window and off-site sample analysis also reported the presence of iron-55 in some samples with high total beta activities. This report discusses the potential for tritium and other radionuclides to contribute to the total beta activity in different samples.

6.2 BECKMAN 6500 LSC DATA

The University's Beckman 6500 LSC was used primarily to analyze smear samples from removable contamination surveys. The data reported by the LSC included tritium and carbon-14 data in dpm and total beta activity in cpm. To convert the total beta cpm data to activity values in dpm, the efficiency of the carbon-14 analysis was used. This efficiency was generally about 75%. This provided a conservative estimate for the total removable beta activity.

The University's Beckman 6500 LSC was last serviced and calibrated in January 2004. Therefore, to calibrate the instrument, Scientech ran a set of quenched tritium calibration standards through the LSC. Table 6-1 provides the results of the analysis of the 10-standard set. Each standard has an activity of 0.0825 uCi.

For smear samples, the quench is generally between 150 and 235. When compared to the standard variance in this range, the error is about +/- 4%. As such, it was determined that the LSC, was operating well and provided accurate measurements.

TABLE 6-1
VALIDATION OF BECKMAN LSC

Standard Name	Quench Value	Tritium Counts (cpm)	Tritium Activity (dpm)	Tritium Activity (uCi)	Percent Variance
1UG	30.1	121234	203543.6	0.092	+11%
2UG	52.1	109350	201442.9	0.091	+10%
3UG	72.4	98038	196438.7	0.088	+7%
4UG	96.1	86099	192392.8	0.087	+5%
5UG	123.4	72356	188814.2	0.085	+3%
6UG	163	53390	185608.3	0.084	+1%
7UG	203.4	36896	175290.5	0.079	-4%
8UG	260.5	19296	167585.4	0.075	-8%
9UG	309.2	10254	155378.3	0.070	-15%
10UG[a]	381.4	3575	61495.89	0.028	-66%

Note: a This quenched standard was outside the quench limits of the Beckman LSC.

6.3 IRON-55 AND NICKEL-63 ESTIMATES

In aluminum and stainless steel samples, the iron-55 and nickel-63 activities are estimated based on the cobalt-60 activity measured using the on-site gamma spectroscopy system. The iron-55 and nickel-63 activities were based on the ratios of the total activity reported in a characterization study of a German TRIGA reactor (Hampel, et. al. 2001).

Similar reactor parts were used for these estimates. For example, the iron-55 and nickel-63 activities provided in Table 4-10 for the NRL reactor's top grid plate are based on the activity values for the top grid plate for the German reactor. The iron-55 activities were also adjusted for decay.

6.4 HIGH BACKGROUND AREAS

As noted throughout this report, there were several areas where the background radiation was elevated due to the presence of radioactive materials. In these areas, contamination surveys are not complete. The elevated background resulted in rejected total surface beta activity measurements. The areas with incomplete contamination surveys include the Radioactive Material Storage Cage, the area outside and above the Radioactive Material Storage Cage, the reactor top, the southeastern corner of the reactor room floor, and the interior of the beam tubes.

6.5 INTERNAL CONTAMINTION

The internal surfaces of various components that may be or are internally contaminated were not surveyed. These include the primary coolant pipes in the tunnel under the reactor, the nitrogen-16 decay tanks, the primary coolant pump, the heat exchanger, the 500-gallon wastewater tank, the beam tubes, and the reactor tank.

7.0 REFERENCES

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APPENDIX A

SURVEY PACKAGES

APPENDIX A-1

MEZZANINE SURVEYS AND SAMPLES

Scientech, LLC

Project #:	23566				**Date:**	7/12/2005
Site Name:	University of Illinois				**Location:**	Mezzanine/Loading Bay
Instrument ID:	L-2360 (K)					
Efficiency:	Alpha:	0.21	Beta:	0.12		
MDC:	Alpha:	33	Beta:	430	dpm/100cm2	

Systematic Locations	α Bkgnd (cpm)	β Bkgnd. (cpm)	Gross α (cpm)	Gross β (cpm)	α Activity (dpm/100cm^2)	β Activity (dpm/100cm^2)	Reportable Alpha	Reportable Beta	Survey Technician
FIXED-POINT MEASUREMENTS									
1	1.2	197.9	5	216	18	151	<MDC	<MDC	Streit
2	1.2	197.9	3	3075	9	R	<MDC	R	Streit
3	1.2	197.9	8	213	32	126	<MDC	<MDC	Streit
4	1.2	197.9	2	234	4	301	<MDC	<MDC	Streit
5	1.2	197.9	2	206	4	68	<MDC	<MDC	Streit
6	1.2	197.9	1	220	-1	184	<MDC	<MDC	Streit
7	1.2	197.9	2	258	4	501	<MDC	501	Streit
8	1.2	197.9	4	242	13	368	<MDC	<MDC	Streit
9	1.2	197.9	4	241	13	359	<MDC	<MDC	Streit
12	1.2	197.9	6	224	23	218	<MDC	<MDC	Streit
13	1.2	197.9	5	235	18	309	<MDC	<MDC	Streit
14	1.2	197.9	4	208	13	84	<MDC	<MDC	Streit
16	1.2	197.9	3	418	9	R	<MDC	R	Streit
17	1.2	197.9	1	220	-1	184	<MDC	<MDC	Streit
18	1.2	197.9	1	225	-1	226	<MDC	<MDC	Streit
19	1.2	197.9	2	231	4	276	<MDC	<MDC	Streit
20	1.2	197.9	1	185	-1	-108	<MDC	<MDC	Streit
21	1.2	197.9	1	324	-1	1051	<MDC	1051	Streit
22	1.2	197.9	2	215	4	143	<MDC	<MDC	Streit
23	1.2	197.9	2	165	4	-274	<MDC	<MDC	Streit
24	1.2	197.9	0	183	-6	-124	<MDC	<MDC	Streit
25	1.2	197.9	2	368	4	1418	<MDC	1418	Streit
26	1.2	197.9	3	208	9	84	<MDC	<MDC	Streit
27	1.2	197.9	4	189	13	-74	<MDC	<MDC	Streit
28	1.2	197.9	2	190	4	-66	<MDC	<MDC	Streit
29	1.2	197.9	3	264	9	551	<MDC	551	Streit
30	1.2	197.9	1	209	-1	93	<MDC	<MDC	Streit
31	1.2	197.9	4	243	13	376	<MDC	<MDC	Streit
32	1.2	197.9	1	218	-1	168	<MDC	<MDC	Streit
33	1.2	197.9	2	175	4	-191	<MDC	<MDC	Streit
34	1.2	197.9	3	248	9	418	<MDC	<MDC	Streit
35	1.2	197.9	3	247	9	409	<MDC	<MDC	Streit
36	1.2	197.9	1	477	-1	R	<MDC	R	Streit
37	1.2	197.9	3	184	9	-116	<MDC	<MDC	Streit
38	1.2	197.9	3	237	9	326	<MDC	<MDC	Streit
39	1.2	197.9	0	271	-6	609	<MDC	609	Streit
40	1.2	197.9	3	852	9	R	<MDC	R	Streit
41	1.2	197.9	8	201	32	26	<MDC	<MDC	Streit
42	1.2	197.9	3	219	9	176	<MDC	<MDC	Streit
43	1.2	197.9	3	196	9	-16	<MDC	<MDC	Streit
44	1.2	197.9	3	2224	9	R	<MDC	R	Streit
45	1.2	197.9	2	2095	4	R	<MDC	R	Streit

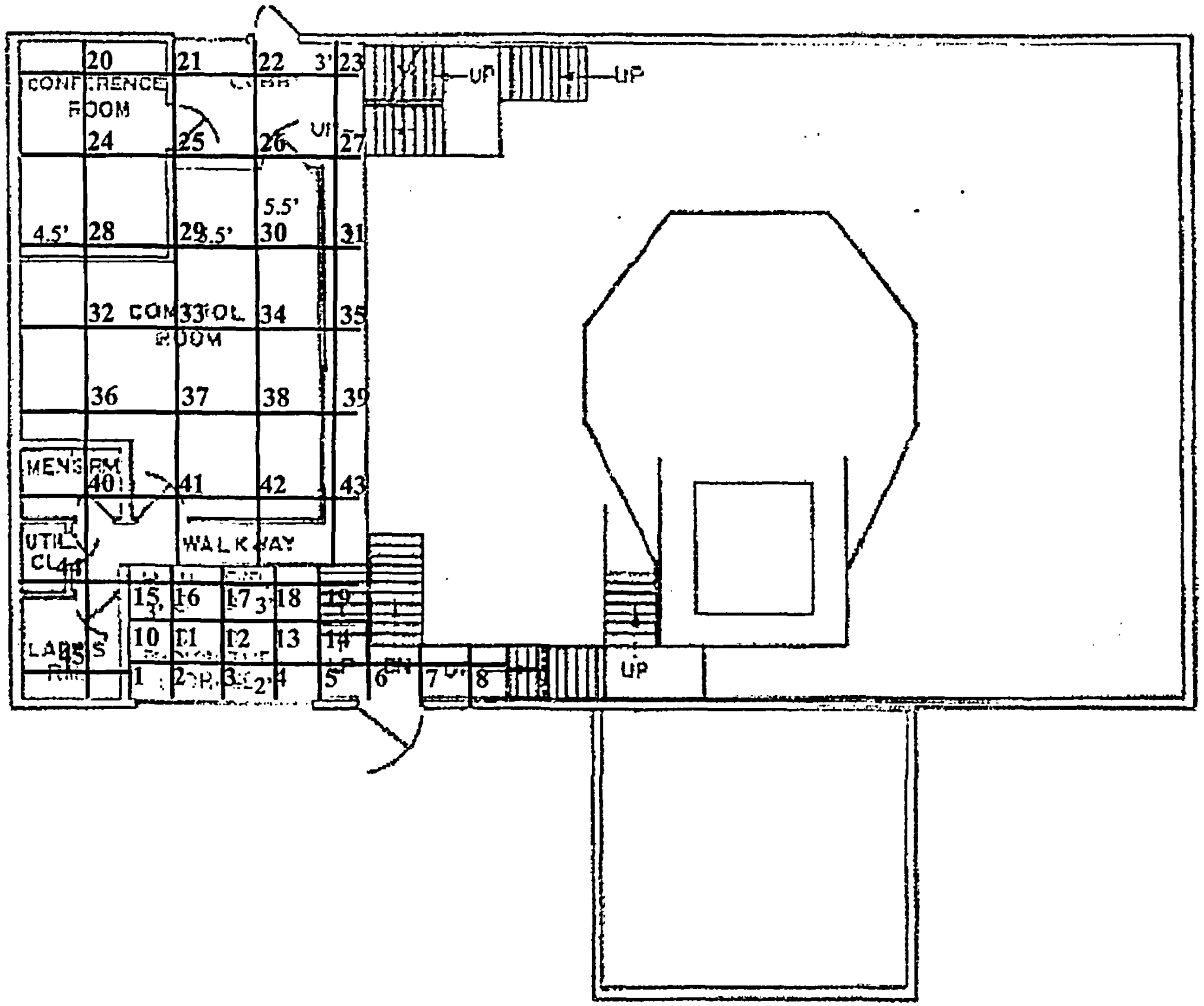
R - rejected due to high background
No measurements at 10, 11, or 15 due to high background

Scientech, LLC

Project #:	23566					**Date:**	7/14/2005	
Site Name:	University of Illinois					**Location:**	Mezzanine/Loading Bay	
Instrument ID:	Beckman-LS6500							
MDC(a):	H-3	42	C-14	27	dpm	Total β	27	dpm
MDC(b):	H-3	32	C-14	29	dpm	Total β	29	dpm

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net total β (dpm)	Reportabe Total β	Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
1	130.85	130.9	1.55	<MDA	28.00	28.0	Streit
1 - Recount	205.07	205.1	14.2	<MDA	-	-	Streit
2	67.2	67.2	209.08	209.1	380.00	380.0	Streit
3	74.57	74.6	2.17	<MDA	41.33	41.3	Streit
4	39.68	39.7	6.79	<MDA	21.33	<MDA	Streit
5	41.07	41.1	5.12	<MDA	37.33	37.3	Streit
6	12.45	<MDA	20.29	<MDA	41.33	41.3	Streit
7	4.65	<MDA	4.61	<MDA	13.33	<MDA	Streit
8	23.27	<MDA	6.65	<MDA	34.67	34.7	Streit
9	3.6	<MDA	169.91	169.9	356.00	356.0	Streit
10	1309.00	1309.0	-19.77	<MDA	-	-	Streit
11	1278.39	1278.4	-17.36	<MDA	-	-	Streit
12	56.83	56.8	7.45	<MDA	8.00	<MDA	Streit
13	12770.26	12770.3	-66.51	<MDA	18.67	<MDA	Streit
14	-7.79	<MDA	16.17	<MDA	28.00	28.0	Streit
15	977.96	978.0	-0.75	<MDA	-	-	Streit
16	38.43	38.4	16.26	<MDA	21.33	<MDA	Streit
17	48.76	48.8	14.12	<MDA	45.33	45.3	Streit
18	76.06	76.1	9.51	<MDA	44.00	44.0	Streit
19	2.79	<MDA	5.27	<MDA	21.33	<MDA	Streit
20	15.86	<MDA	0.38	<MDA	12.00	<MDA	Streit
21	36.97	<MDA	6.38	<MDA	25.33	<MDA	Streit
22	-3.77	<MDA	6.02	<MDA	10.67	<MDA	Streit
23	15.96	<MDA	10.22	<MDA	4.00	<MDA	Streit
24	9.03	<MDA	10.25	<MDA	18.67	<MDA	Streit
25	13.34	<MDA	15.78	<MDA	41.33	41.3	Streit
26	29.7	<MDA	3.11	<MDA	17.33	<MDA	Streit
27	46.63	46.6	9.92	<MDA	22.67	<MDA	Streit
28	16.35	<MDA	-2.22	<MDA	25.33	<MDA	Streit
29	-16.37	<MDA	4.6	<MDA	5.33	<MDA	Streit
29 - Recount	225.28	225.3	-1.41	<MDA	-	-	Streit
30	-11.78	<MDA	6.01	<MDA	17.33	<MDA	Streit
31	52.3	52.3	3.05	<MDA	22.67	<MDA	Streit
32	30.21	<MDA	-3.92	<MDA	1.33	<MDA	Streit
33	26.74	<MDA	9.79	<MDA	4.00	<MDA	Streit
34	4.83	<MDA	1.68	<MDA	-2.67	<MDA	Streit
35	13.36	<MDA	4.6	<MDA	12.00	<MDA	Streit
36	13.3	<MDA	-5.2	<MDA	5.33	<MDA	Streit
37	33.08	<MDA	-3.84	<MDA	9.33	<MDA	Streit
38	7.09	<MDA	-3.84	<MDA	5.33	<MDA	Streit
39	170.96	171.0	8.88	<MDA	56.00	56.0	Streit
39 - Recount	27.25	<MDA	-2.36	<MDA	-	-	Streit
40	16.23	<MDA	8.44	<MDA	-5.33	<MDA	Streit
41	15.84	<MDA	11.54	<MDA	17.33	<MDA	Streit
42	34.43	<MDA	10.38	<MDA	21.33	<MDA	Streit
43	24.88	<MDA	10.31	<MDA	12.00	<MDA	Streit
44	272.55	272.6	17.63	<MDA	-2.67	<MDA	Streit
45	27.52	<MDA	5.59	<MDA	-8.00	<MDA	Streit
Near 13	48.39	48.4	-10.10	<MDA			Streit
Near 13	5.44	<MDA	-7.07	<MDA			Streit
Near 13	33.87	<MDA	-7.49	<MDA			Streit
Near 39	33.0	<MDA	-1.41	<MDA			Streit
Near 39	34.0	34.0	-21.80	<MDA			Streit

FIGURE A-1-1: MEZZANINE LEVEL AND TRUCK BAY SURVEY GRIDS
(Measurements made at grid intersections)



Scientech, LLC

Project #: 23566
Site Name: University of Illinois
Instrument ID: L-2360 (L)
Efficiency: Alpha: 0.20 Beta: 0.13
MDC: Alpha: 37 Beta: 454 dpm/100cm2

Date: 7/13/2005
Location: Wall Surveys
(Sheet 1 of 2)

Systematic Locations	α Bkgnd (cpm)	β Bkgnd. (cpm)	Gross α (cpm)	Gross β (cpm)	α Activity (dpm/100cm²)	β Activity (dpm/100cm²)	Reportable Alpha	Reportable Beta	Survey Technician
FIXED-POINT MEASUREMENTS									
101	1.5	262.4	3	384	8	935	<MDC	935	Higgins
102	1.5	262.4	3	364	8	782	<MDC	782	Higgins
103	1.5	262.4	2	870	3	R	<MDC	R	Higgins
104	1.5	262.4	5	537	18	R	<MDC	R	Higgins
105	1.5	262.4	5	365	18	789	<MDC	789	Higgins
106	1.5	262.4	0	219	-8	-334	<MDC	<MDC	Higgins
107	1.5	262.4	5	376	18	874	<MDC	874	Higgins
108	1.5	262.4	3	252	8	-80	<MDC	<MDC	Higgins
109	1.5	262.4	5	390	18	R	<MDC	R	Higgins
110	1.5	262.4	4	332	13	535	<MDC	535	Higgins
111	1.5	262.4	5	192	18	-542	<MDC	<MDC	Higgins
112	1.5	262.4	2	1067	3	R	<MDC	R	Higgins
113	1.5	262.4	8	560	33	R	<MDC	R	Higgins
114	1.5	262.4	4	375	13	R	<MDC	R	Higgins
115	1.5	262.4	3	344	8	R	<MDC	R	Higgins
116	1.5	262.4	4	253	13	-72	<MDC	<MDC	Higgins
117	1.5	262.4	7	188	28	-572	<MDC	<MDC	Higgins
118	1.5	262.4	9	257	38	-42	38	<MDC	Higgins
119	1.5	262.4	6	285	23	174	<MDC	<MDC	Higgins
120	1.5	262.4	6	323	23	466	<MDC	466	Higgins
121	1.5	262.4	4	627	13	2805	<MDC	2805	Higgins
122	1.5	262.4	4	288	13	197	<MDC	<MDC	Higgins
123	1.5	262.4	9	243	38	-149	38	<MDC	Higgins
124	1.5	262.4	4	327	13	497	<MDC	497	Higgins
125	1.5	262.4	10	690	43	3289	43	3289	Higgins
126	1.5	262.4	2	725	3	3558	<MDC	3558	Higgins
127	1.5	262.4	6	555	23	2251	<MDC	2251	Higgins
128	1.5	262.4	4	591	13	2528	<MDC	2528	Higgins
129	1.5	262.4	5	363	18	774	<MDC	774	Higgins
130	1.5	262.4	5	670	18	3135	<MDC	3135	Higgins
131	1.5	262.4	7	1035	28	5943	<MDC	5943	Higgins
132	1.5	262.4	7	738	28	R	<MDC	R	Higgins
133	1.5	262.4	0	5458	-8	R	<MDC	R	Higgins
134	1.5	262.4	7	803	28	R	<MDC	R	Higgins
135	1.5	262.4	5	327	18	497	<MDC	497	Higgins
136	1.5	262.4	2	303	3	312	<MDC	<MDC	Higgins
137	1.5	262.4	3	316	8	412	<MDC	<MDC	Higgins
138	1.5	262.4	7	312	28	382	<MDC	<MDC	Higgins
139	1.5	262.4	2	302	3	305	<MDC	<MDC	Higgins
140	1.5	262.4	4	276	13	105	<MDC	<MDC	Higgins
141	1.5	262.4	9	237	38	-195	38	<MDC	Higgins
142	1.5	262.4	7	296	28	258	<MDC	<MDC	Higgins
143	1.5	262.4	7	301	28	297	<MDC	<MDC	Higgins
144	1.5	262.4	2	258	3	-34	<MDC	<MDC	Higgins
145	1.5	262.4	2	251	3	-88	<MDC	<MDC	Higgins

R - Rejected due to high background from the Radioactive Material Storage Cage.
Survey points 121 - 128 are in the small mechanical/electrical room in the northwestern corner of the reactor room.
Survey point 130 and 131 are on the reactor room side of the walls of the small mechanical/electrical room.

Scientech, LLC

Project #:	23566	**Date:**	7/13/2005
Site Name:	University of Illinois	**Location:**	Wall Survey
Instrument ID:	L-2360 (L)		(Sheet 2 of 2)

Efficiency: Alpha: 0.2 Beta: 0.13
MDC: Alpha: 37 Beta: 454 dpm/100cm2

Systematic Locations	α Bkgnd (cpm)	β Bkgnd. (cpm)	Gross α (cpm)	Gross β (cpm)	α Activity (dpm/100cm^2)	β Activity (dpm/100cm^2)	Reportable Alpha	Reportable Beta	Survey Technician
FIXED-POINT MEASUREMENTS									
146	1.5	262.4	7	265	28	20	<MDC	<MDC	Higgins
147	1.2	197.9	5	279	19	624	<MDC	624	Higgins
148	1.2	197.9	7	257	29	455	<MDC	455	Higgins
149	1.2	197.9	5	299	19	778	<MDC	778	Higgins
150	1.2	197.9	1	391	-1	R	<MDC	R	Higgins
151	1.2	197.9	4	450	14	R	<MDC	R	Higgins
152	1.2	197.9	2	422	4	R	<MDC	R	Higgins
153	1.2	197.9	7	548	29	R	<MDC	R	Higgins
154	1.2	197.9	5	939	19	R	<MDC	R	Higgins
155	1.2	197.9	4	682	14	R	<MDC	R	Higgins
156	1.2	197.9	4	267	14	532	<MDC	532	Higgins
157	1.2	197.9	6	257	24	455	<MDC	455	Higgins
158	1.2	197.9	7	301	29	793	<MDC	793	Higgins
159	1.2	197.9	9	305	39	824	39	824	Higgins
160	1.2	197.9	5	307	19	839	<MDC	839	Higgins
161	1.2	197.9	3	1385	9	R	<MDC	R	Higgins
162	1.2	197.9	4	392	14	R	<MDC	R	Higgins
163	1.2	197.9	6	407	24	R	<MDC	R	Higgins
164	1.2	197.9	4	352	14	R	<MDC	R	Higgins
165	1.2	197.9	4	279	14	624	<MDC	624	Higgins
166	1.2	197.9	5	261	19	485	<MDC	485	Higgins
167	1.2	197.9	2	303	4	808	<MDC	808	Higgins
168	1.2	197.9	7	294	29	739	<MDC	739	Higgins
169	1.2	197.9	7	342	29	1108	<MDC	1108	Higgins
170	1.2	197.9	6	258	24	462	<MDC	462	Higgins
171	1.2	197.9	6	285	24	670	<MDC	670	Higgins
172	1.2	197.9	5	358	19	1232	<MDC	1232	Higgins
173	1.2	197.9	3	293	9	732	<MDC	732	Higgins
174	1.2	197.9	5	286	19	678	<MDC	678	Higgins
175	1.2	197.9	4	280	14	632	<MDC	632	Higgins
176	1.2	197.9	5	285	19	670	<MDC	670	Higgins
177	1.2	197.9	7	381	29	1408	<MDC	1408	Higgins
178	1.2	197.9	8	264	34	508	<MDC	508	Higgins
179	1.2	197.9	3	347	9	1147	<MDC	1147	Higgins
180	1.2	197.9	5	285	19	670	<MDC	670	Higgins
181	1.2	197.9	7	255	29	439	<MDC	<MDC	Higgins
185	1.2	197.9	9	220	39	170	39	<MDC	Higgins
186	1.2	197.9	6	215	24	132	<MDC	<MDC	Higgins
187	1.2	197.9	6	211	24	101	<MDC	<MDC	Higgins
188	1.2	197.9	6	192	24	-45	<MDC	<MDC	Higgins
189	1.2	197.9	4	228	14	232	<MDC	<MDC	Higgins
190	1.2	197.9	6	258	24	462	<MDC	462	Higgins
191	1.2	197.9	4	268	14	539	<MDC	539	Higgins
192	1.2	197.9	6	203	24	39	<MDC	<MDC	Higgins

R - Rejected due to high background in the area of the survey point.
Survey points 162 - 181 are on the reactor bioshield.

Scientech, LLC

Project #:	23566				Date:	7/15/2005	
Site Name:	University of Illinois				Location:	Wall Survey	
Instrument ID:	Beckman-LS6500					(Sheet 1 of 2)	
MDA(a):	H-3 44	C-14	24	dpm	Total β	24	dpm
MDA(b):	H-3	C-14		dpm	Total β		dpm

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net total β (dpm)	Reportabe Total β	Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
101	-4.61	<MDA	26.79	26.8	46.67	46.7	Higgins
102	-26.56	<MDA	6.94	<MDA	26.67	26.7	Higgins
103	-16.65	<MDA	6.87	<MDA	9.33	<MDA	Higgins
104	-2.51	<MDA	9.39	<MDA	-6.67	<MDA	Higgins
105	-0.54	<MDA	9.38	<MDA	13.33	<MDA	Higgins
106	-22.62	<MDA	23.01	<MDA	12.00	<MDA	Higgins
107	-12.65	<MDA	12.20	<MDA	12.00	<MDA	Higgins
108	-11.63	<MDA	0.20	<MDA	4.00	<MDA	Higgins
109	-9.91	<MDA	8.08	<MDA	10.67	<MDA	Higgins
110	9.68	<MDA	5.75	<MDA	-1.33	<MDA	Higgins
111	2.36	<MDA	11.27	<MDA	17.33	<MDA	Higgins
112	22.42	<MDA	1.37	<MDA	16.00	<MDA	Higgins
113	2.87	<MDA	6.85	<MDA	0.00	<MDA	Higgins
114	-14.41	<MDA	7.16	<MDA	24.00	24.0	Higgins
115	-10.15	<MDA	21.23	<MDA	12.00	<MDA	Higgins
116	-22.67	<MDA	12.24	<MDA	6.67	<MDA	Higgins
117	-13.00	<MDA	8.21	<MDA	21.33	<MDA	Higgins
118	-0.32	<MDA	16.97	<MDA	14.67	<MDA	Higgins
119	6.43	<MDA	14.00	<MDA	21.33	<MDA	Higgins
120	106.13	106.1	9.54	<MDA	77.33	77.3	Higgins
121	-10.42	<MDA	12.49	<MDA	-5.33	<MDA	Higgins
122	-4.78	<MDA	5.57	<MDA	13.33	<MDA	Higgins
123	-3.33	<MDA	-3.75	<MDA	8.00	<MDA	Higgins
124	-7.57	<MDA	9.71	<MDA	-9.33	<MDA	Higgins
125	-4.88	<MDA	4.32	<MDA	25.33	25.3	Higgins
126	-16.37	<MDA	8.33	<MDA	33.33	33.3	Higgins
127	-8.59	<MDA	1.76	<MDA	26.67	26.7	Higgins
128	-5.36	<MDA	6.91	<MDA	49.33	49.3	Higgins
129	-17.17	<MDA	8.30	<MDA	40.00	40.0	Higgins
130	-21.13	<MDA	8.23	<MDA	41.33	41.3	Higgins
131	-0.57	<MDA	0.07	<MDA	18.67	<MDA	Higgins
132	2.63	<MDA	13.42	<MDA	9.33	<MDA	Higgins
133	-11.50	<MDA	-2.54	<MDA	46.67	46.7	Higgins
134	-12.76	<MDA	12.18	<MDA	26.67	26.7	Higgins
135	-21.60	<MDA	14.94	<MDA	9.33	<MDA	Higgins
136	-16.01	<MDA	16.27	<MDA	40.00	40.0	Higgins
137	-17.60	<MDA	8.12	<MDA	20.00	<MDA	Higgins
138	-8.08	<MDA	4.11	<MDA	2.67	<MDA	Higgins
139	18.69	<MDA	9.45	<MDA	46.67	46.7	Higgins
140	-1.94	<MDA	7.13	<MDA	0.00	<MDA	Higgins
141	3.96	<MDA	12.46	<MDA	32.00	32.0	Higgins
142	-2.68	<MDA	11.29	<MDA	33.33	33.3	Higgins
143	-2.89	<MDA	4.44	<MDA	32.00	32.0	Higgins
144	-26.68	<MDA	1.52	<MDA	12.00	<MDA	Higgins
145	-18.79	<MDA	12.20	<MDA	20.00	<MDA	Higgins

Survey points 101 - 120 are from the mezzanine level, loading bay, and storage level.

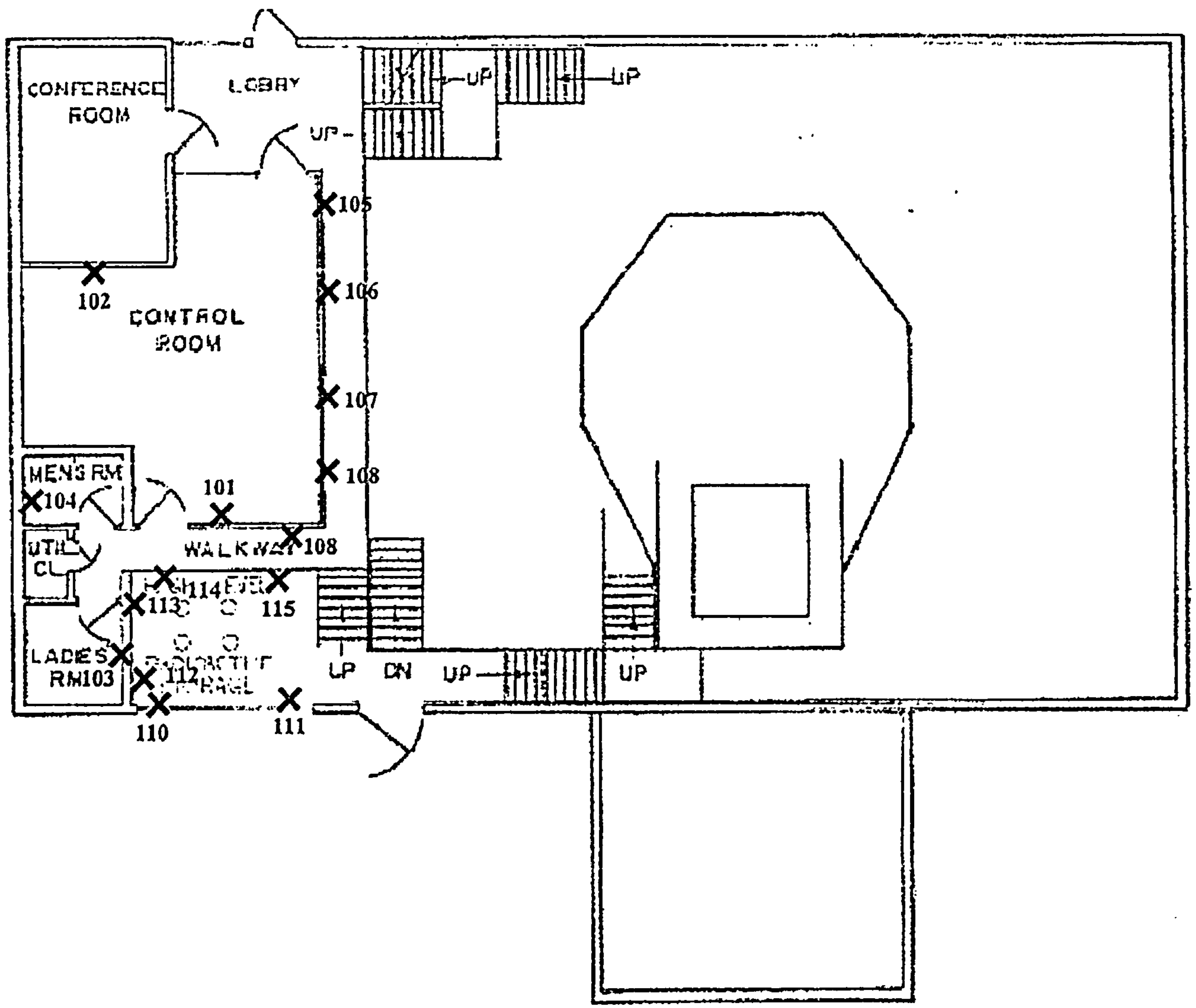
Scientech, LLC

Project #:	23566	Date:	7/15/2005
Site Name:	University of Illinois	Location:	Wall Survey
Instrument ID:	Beckman-LS6500		(Sheet 2 of 2)

MDA(a):	H-3	44	C-14	24	dpm	Total β	24	
MDA(b):	H-3	47	C-14	25	dpm	Total β	25	

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net total β (dpm)	Reportabe Total β	Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
146	4.15	<MDA	0.15	<MDA	14.67	<MDA	Higgins
147	131.85	131.9	5.86	<MDA	105.33	105.3	Higgins
148	-15.00	<MDA	6.81	<MDA	21.33	<MDA	Higgins
149	-7.62	<MDA	9.92	<MDA	21.33	<MDA	Higgins
150	-5.71	<MDA	8.09	<MDA	17.33	<MDA	Higgins
151	-6.78	<MDA	1.45	<MDA	5.33	<MDA	Higgins
152	-21.18	<MDA	8.18	<MDA	25.33	25.3	Higgins
153	-12.77	<MDA	8.20	<MDA	33.33	33.3	Higgins
154	22.91	<MDA	691.26	691.3	1077.33	1077.3	Higgins
155	-11.26	<MDA	23.42	<MDA	32.00	32.0	Higgins
156	-6.96	<MDA	6.77	<MDA	21.33	<MDA	Higgins
157	-5.31	<MDA	1.52	<MDA	37.33	37.3	Higgins
158	-17.44	<MDA	7.00	<MDA	29.33	29.3	Higgins
159	-18.10	<MDA	4.35	<MDA	14.67	<MDA	Higgins
160	-4.78	<MDA	12.86	<MDA	20.00	<MDA	Higgins
161	-13.28	<MDA	13.15	<MDA	18.67	<MDA	Higgins
162	-14.80	<MDA	1.48	<MDA	26.67	26.7	Higgins
163	-10.82	<MDA	12.60	<MDA	13.33	<MDA	Higgins
164	-17.25	<MDA	20.28	<MDA	64.00	64.0	Higgins
165	-17.76	<MDA	10.91	<MDA	17.33	<MDA	Higgins
166	5.95	<MDA	9.50	<MDA	14.67	<MDA	Higgins
167	-8.23	<MDA	5.46	<MDA	38.67	38.7	Higgins
168	-10.67	<MDA	1.61	<MDA	22.67	<MDA	Higgins
169	4.22	<MDA	-2.57	<MDA	18.67	<MDA	Higgins
170	-13.33	<MDA	-3.78	<MDA	53.33	53.3	Higgins
171	-15.96	<MDA	16.22	<MDA	30.67	30.7	Higgins
172	-16.72	<MDA	1.68	<MDA	34.67	34.7	Higgins
173	-24.96	<MDA	11.17	<MDA	17.33	<MDA	Higgins
174	-13.05	<MDA	9.57	<MDA	25.33	25.3	Higgins
175	-13.73	<MDA	-1.06	<MDA	-16.00	<MDA	Higgins
176	-1.41	<MDA	8.37	<MDA	14.67	<MDA	Higgins
177	-19.86	<MDA	20.65	<MDA	-10.67	<MDA	Higgins
178	-24.58	<MDA	9.78	<MDA	-28.00	<MDA	Higgins
179	-10.75	<MDA	3.58	<MDA	-10.67	<MDA	Higgins
180	-23.85	<MDA	16.28	<MDA	-14.67	<MDA	Higgins
181	-6.94	<MDA	-1.18	<MDA	6.67	<MDA	Higgins
185	-4.20	<MDA	4.45	<MDA	-9.33	<MDA	Higgins
186	92.45	92.5	-0.05	<MDA	8.00	<MDA	Higgins
187	-19.73	<MDA	9.83	<MDA	-4.00	<MDA	Higgins
188	-22.75	<MDA	18.07	<MDA	-9.33	<MDA	Higgins
189	-11.99	<MDA	6.05	<MDA	-20.00	<MDA	Higgins
190	-8.51	<MDA	8.55	<MDA	-10.67	<MDA	Higgins
191	-17.60	<MDA	3.20	<MDA	-12.00	<MDA	Higgins
192	-3.54	<MDA	5.62	<MDA	-28.00	<MDA	Higgins
193	-18.79	<MDA	12.20	<MDA	20.00	<MDA	Higgins

FIGURE A-1-2: MEZZANINE LEVEL AND TRUCK BAY WALL SURVEY POINTS



APPENDIX A-2

REACTOR ROOM SURVEYS AND SAMPLES

Scientech, LLC

Project #:	23566				Date:	7/12/2005
Site Name:	University of Illinois				Location:	Reacor Room
Instrument ID:	L-2360 (L)					(Sheet 1 of 2)
Efficiency:	Alpha: 0.20	Beta:	0.14			
MDC:	Alpha: 37	Beta:	433	dpm/100cm2		

Systematic Locations	α Bkgnd (cpm)	β Bkgnd. (cpm)	Gross α (cpm)	Gross β (cpm)	α Activity (dpm/100cm^2)	β Activity	Reportable Alpha	Reportable Beta	Survey Technician
FIXED-POINT MEASUREMENTS									
5	1.6	278.6	2	672	2	R	<MDC	R	Taylor
8	1.6	278.6	1	1171	-3	R	<MDC	R	Taylor
9	1.6	278.6	1	1833	-3	R	<MDC	R	Taylor
10	1.6	278.6	3	598	7	R	<MDC	R	Taylor
11	1.6	278.6	2	619	2	R	<MDC	R	Taylor
12	1.6	278.6	1	1708	-3	R	<MDC	R	Taylor
15	1.6	278.6	1	672	-3	R	<MDC	R	Taylor
16	1.6	278.6	5	911	17	R	<MDC	R	Taylor
17	1.6	278.6	1	591	-3	R	<MDC	R	Taylor
18	1.6	278.6	1	504	-3	R	<MDC	R	Taylor
19	1.6	278.6	1	697	-3	R	<MDC	R	Taylor
20	1.6	278.6	4	372	12	667	<MDC	667	Taylor
21	1.6	278.6	0	331	-8	374	<MDC	<MDC	Taylor
22	1.6	278.6	1	390	-3	R	<MDC	R	Taylor
23	1.6	278.6	5	678	17	R	<MDC	R	Taylor
24	1.6	278.6	3	645	7	R	<MDC	R	Taylor
25	1.6	278.6	4	496	12	1553	<MDC	1553	Taylor
26	1.6	278.6	1	451	-3	1231	<MDC	1231	Taylor
27	1.6	278.6	1	378	-3	710	<MDC	710	Taylor
28	1.6	278.6	1	331	-3	374	<MDC	<MDC	Taylor
29	1.6	278.6	0	546	-8	1910	<MDC	1910	Taylor
30	1.6	278.6	2	521	2	1731	<MDC	1731	Taylor
31	1.6	278.6	1	382	-3	739	<MDC	739	Taylor
32	1.6	278.6	5	307	17	203	<MDC	<MDC	Taylor
33	1.6	278.6	0	295	-8	117	<MDC	<MDC	Taylor
34	1.6	278.6	1	302	-3	167	<MDC	<MDC	Taylor
35	1.6	278.6	7	315	27	260	<MDC	<MDC	Taylor
36	1.6	278.6	7	328	27	353	<MDC	<MDC	Taylor
37	1.6	278.6	5	342	17	453	<MDC	453	Taylor
38	1.6	278.6	4	288	12	67	<MDC	<MDC	Taylor
41	1.6	278.6	0	1978	-8	R	<MDC	R	Taylor
42	1.6	278.6	1	365	-3	617	<MDC	617	Taylor
43	1.6	278.6	2	343	2	460	<MDC	460	Taylor
44	1.6	278.6	0	338	-8	424	<MDC	<MDC	Taylor
45	1.6	278.6	4	342	12	453	<MDC	453	Taylor
46	1.6	278.6	4	303	12	174	<MDC	<MDC	Taylor
47	1.6	278.6	3	290	7	81	<MDC	<MDC	Taylor
48	1.6	278.6	2	284	2	39	<MDC	<MDC	Taylor
49	1.6	278.6	3	313	7	246	<MDC	<MDC	Taylor
50	1.6	278.6	4	287	12	60	<MDC	<MDC	Taylor
51	1.6	278.6	4	299	12	146	<MDC	<MDC	Taylor

R - rejected due to high background

Scientech, LLC

Project #:	23566			Date:	7/12/2005
Site Name:	University of Illinois			Location:	Reacor Room
Instrument ID:	L-2360 (K)				(Sheet 2 of 2)
Efficiency:	Alpha: 0.21	Beta: 0.12			
MDC:	Alpha: 33	Beta: 430	dpm/100cm2		

Systematic Locations	α Bkgnd (cpm)	β Bkgnd. (cpm)	Gross α (cpm)	Gross β (cpm)	α Activity (dpm/100cm^2)	β Activity (dpm/100cm^2)	Reportable Alpha	Reportable Beta	Survey Technician
FIXED-POINT MEASUREMENTS									
5	1.2	197.9	1	517	-1	R	<MDC	R	Streit
6	1.2	197.9	4	258	13	501	<MDC	501	Streit
7	1.2	197.9	2	237	4	326	<MDC	<MDC	Streit
12	1.2	197.9	0	1340	-6	R	<MDC	R	Streit
13	1.2	197.9	4	403	13	1709	<MDC	1709	Streit
14	1.2	197.9	1	372	-1	1451	<MDC	1451	Streit
39	1.2	197.9	1	155	-1	-358	<MDC	<MDC	Streit
40	1.2	197.9	5	231	18	276	<MDC	<MDC	Streit
52	1.2	197.9	2	228	4	251	<MDC	<MDC	Streit
54	1.2	197.9	2	203	4	43	<MDC	<MDC	Streit
55	1.2	197.9	4	223	13	209	<MDC	<MDC	Streit
56	1.2	197.9	4	223	13	209	<MDC	<MDC	Streit
57	1.2	197.9	7	229	28	259	<MDC	<MDC	Streit
58	1.2	197.9	3	235	9	309	<MDC	<MDC	Streit
59	1.2	197.9	4	242	13	368	<MDC	<MDC	Streit
60	1.2	197.9	2	258	4	501	<MDC	501	Streit
61	1.2	197.9	4	218	13	168	<MDC	<MDC	Streit
62	1.2	197.9	0	258	-6	501	<MDC	501	Streit
63	1.2	197.9	3	213	9	126	<MDC	<MDC	Streit
64	1.2	197.9	2	236	4	318	<MDC	<MDC	Streit
65	1.2	197.9	1	277	-1	659	<MDC	659	Streit
66	1.2	197.9	3	2064	9	R	<MDC	R	Streit
67	1.2	197.9	2	373	4	1459	<MDC	1459	Streit
68	1.2	197.9	5	359	18	1343	<MDC	1343	Streit
69	1.2	197.9	3	285	9	734	<MDC	734	Streit
70	1.2	197.9	0	219	-6	176	<MDC	<MDC	Streit
71	1.2	197.9	2	204	4	51	<MDC	<MDC	Streit
72	1.2	197.9	4	316	13	R	<MDC	R	Streit
73	1.2	197.9	2	321	4	R	<MDC	R	Streit
74	1.2	197.9	2	477	4	R	<MDC	R	Streit

R - rejected due to high background

Scientech, LLC

Project #:	23566					**Date:**	7/14/2005
Site Name:	University of Illinois					**Location:**	Reactor Room
Instrument ID:	Beckman-LS6500						(Sheet 1 of 2)
MDC(a):	H-3 41	C-14	29 dpm	Total β	29		
MDC(b):	H-3 40	C-14	29 dpm	Total β	29		

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net total β (dpm)	Reportable Total β	Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
1	36.1	<MDA	7.09	<MDA	18.67	<MDA	Streit
2	37.54	<MDA	30.63	30.6	14.67	<MDA	Streit
3	78.01	78.0	3.23	<MDA	25.33	<MDA	Streit
4	17.99	<MDA	3.36	<MDA	29.33	29.3	Streit
4 - Recount	9.45	<MDA	2.20	<MDA	-	-	Streit
5	61.25	61.3	10.57	<MDA	26.67	<MDA	Streit
8	34.87	<MDA	3.19	<MDA	46.67	46.7	Streit
9	8.18	<MDA	3.17	<MDA	6.67	<MDA	Streit
10	27.16	<MDA	4.67	<MDA	28.00	<MDA	Streit
11	15.78	<MDA	7.52	<MDA	42.67	42.7	Streit
12	18.39	<MDA	10.33	<MDA	13.33	<MDA	Streit
15	10.6	<MDA	13.99	<MDA	36.00	36.0	Streit
15 - Recount	32.53	<MDA	7.13	<MDA	-	-	Streit
16	67.64	67.6	9.92	<MDA	17.33	<MDA	Streit
17	74.43	74.4	7.35	<MDA	22.67	<MDA	Streit
18	29.53	<MDA	10.52	<MDA	6.67	<MDA	Streit
19	19.93	<MDA	9.14	<MDA	29.33	29.3	Streit
20	34.07	<MDA	-0.72	<MDA	2.67	<MDA	Streit
21	41.45	41.5	3.08	<MDA	5.33	<MDA	Streit
22	-12.34	<MDA	0.64	<MDA	25.33	<MDA	Streit
23	0.72	<MDA	6.18	<MDA	14.67	<MDA	Streit
24	27.29	<MDA	8.16	<MDA	28.00	<MDA	Streit
25	-6.35	<MDA	6.16	<MDA	30.67	30.7	Streit
26	30.58	<MDA	11.84	<MDA	21.33	<MDA	Streit
27	76.36	76.4	13.16	<MDA	20.00	<MDA	Streit
28	41.11	41.1	6.04	<MDA	12.00	<MDA	Streit
29	13.98	<MDA	3.48	<MDA	37.33	37.3	Streit
30	89.01	89.0	2.76	<MDA	44.00	44.0	Streit
31	33.25	<MDA	4.83	<MDA	-1.33	<MDA	Streit
32	295.99	296.0	10.13	<MDA	30.67	30.7	Streit
33	53.83	53.8	0.5	<MDA	30.67	30.7	Streit
34	70.55	70.6	5.84	<MDA	29.33	29.3	Streit
34 - Recount	23.78	<MDA	11.98	<MDA	-	-	Streit
35	72.05	72.1	36.78	36.8	20.00	<MDA	Streit
36	54.09	54.1	4.51	<MDA	30.67	30.7	Streit
37	30.57	<MDA	15.66	<MDA	42.67	42.7	Streit
38	4.5	<MDA	15.68	<MDA	5.33	<MDA	Streit
41	34.91	<MDA	7.6	<MDA	4.00	<MDA	Streit
42	71.53	71.5	0.39	<MDA	5.33	<MDA	Streit
43	35.45	<MDA	5.91	<MDA	29.33	29.3	Streit
44	42	42.0	13.16	<MDA	44.00	44.0	Streit
45	133.8	133.8	4.37	<MDA	14.67	<MDA	Streit
46	62.78	62.8	14.19	<MDA	50.67	50.7	Streit
47	39	<MDA	2.94	<MDA	12.00	<MDA	Streit
48	10.27	<MDA	8.76	<MDA	41.33	41.3	Streit
49	16.99	<MDA	16.01	<MDA	5.33	<MDA	Streit
50	78.48	78.5	39.29	39.3	58.67	58.7	Streit
51	49.61	49.6	6.24	<MDA	38.67	38.7	Streit

Scientech, LLC

Project #:	23566					Date:	7/14/2005
Site Name:	University of Illinois					Location:	Reactor Room
Instrument ID:	Beckman-LS6500						(Sheet 2 of 2)
MDC(a):	H-3	37	C-14	28 dpm	Total β	28 dpm	
MDC(b):	H-3	42	C-14	27 dpm	Total β	27 dpm	

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net total β (dpm)	Reportabe Total β	Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
5	-10.85	<MDA	12.73	<MDA	1.33	<MDA	Streit
6	43.78	43.8	13.39	<MDA	17.33	<MDA	Streit
7	62.82	62.8	1.6	<MDA	21.33	<MDA	Streit
12	2.17	<MDA	3.77	<MDA	9.33	<MDA	Streit
13	45.18	45.2	16.13	<MDA	26.67	<MDA	Streit
14	30.75	<MDA	8.95	<MDA	8.00	<MDA	Streit
39	30.08	<MDA	0.83	<MDA	-4.00	<MDA	Streit
40	44.31	44.3	11.84	<MDA	17.33	<MDA	Streit
52	7.26	<MDA	0.64	<MDA	-9.33	<MDA	Streit
54	333.09	333.1	-0.88	<MDA	9.33	<MDA	Streit
55	101.42	101.4	16.3	<MDA	36.00	36.0	Streit
56	51.38	51.4	8.78	<MDA	16.00	<MDA	Streit
57	74.84	74.8	24.8	<MDA	21.33	<MDA	Streit
57 - Recount	38.86	<MDA	12.67	<MDA	-	-	Streit
58	97.18	97.2	26.43	<MDA	64.00	64.0	Streit
59	45.53	45.5	7.42	<MDA	14.67	<MDA	Streit
60	85.71	85.7	24.96	<MDA	68.00	68.0	Streit
61	168.48	168.5	9.31	<MDA	40.00	40.0	Streit
62	28.06	<MDA	0.79	<MDA	21.33	<MDA	Streit
63	57.42	57.4	15.59	<MDA	54.67	54.7	Streit
63 - Recount	44.31	44.3	24.35	<MDA	-	-	Streit
64	40.66	40.7	22.85	<MDA	30.67	30.7	Streit
65	64.21	64.2	6.54	<MDA	29.33	29.3	Streit
66	50.5	50.5	27.66	27.7	106.67	106.7	Streit
67	23.59	<MDA	-0.57	<MDA	28.00	28.0	Streit
67 - Recount	29.44	<MDA	-3.11	<MDA	-	-	Streit
68	77.46	77.5	14.55	<MDA	24.00	<MDA	Streit
69	58.48	58.5	9.13	<MDA	24.00	<MDA	Streit
70	60.82	60.8	1.91	<MDA	29.33	29.3	Streit
71	59.16	59.2	14.14	<MDA	32.00	32.0	Streit
72	50.14	50.1	0.84	<MDA	49.33	49.3	Streit
73	74.2	74.2	14.31	<MDA	54.67	54.7	Streit
74	56.98	57.0	34.69	34.7	50.67	50.7	Streit

Scientech, LLC

Project #:	23566	**Date:**	7/12/2005
Site Name:	University of Illinois	**Location:**	Mech. Equip Room
Instrument ID:	L-2360 (K)		
Efficiency:	Alpha: 0.21	Beta: 0.12	
MDC:	Alpha: 33	Beta: 430	dpm/100cm2

Systematic Locations	α Bkgnd (cpm)	β Bkgnd. (cpm)	Gross α (cpm)	Gross β (cpm)	α Activity (dpm/100cm²)	β Activity (dpm/100cm²)	Reportable Alpha	Reportable Beta	Survey Technician
FIXED-POINT MEASUREMENTS									
85	1.2	197.9	1	199	-1	9	<MDC	<MDC	Streit
86	1.2	197.9	6	176	23	-183	<MDC	<MDC	Streit
87	1.2	197.9	3	214	9	134	<MDC	<MDC	Streit
88	1.2	197.9	4	227	13	243	<MDC	<MDC	Streit
89	1.2	197.9	3	202	9	34	<MDC	<MDC	Streit
90	1.2	197.9	2	144	4	-449	<MDC	<MDC	Streit
91	1.2	197.9	1	185	-1	-108	<MDC	<MDC	Streit
92	1.2	197.9	0	215	-6	143	<MDC	<MDC	Streit
93	1.2	197.9	2	192	4	-49	<MDC	<MDC	Streit
94	1.2	197.9	2	134	4	-533	<MDC	<MDC	Streit
95	1.2	197.9	3	194	9	-33	<MDC	<MDC	Streit
96	1.2	197.9	1	226	-1	234	<MDC	<MDC	Streit
97	1.2	197.9	6	206	23	68	<MDC	<MDC	Streit
98	1.2	197.9	3	230	9	268	<MDC	<MDC	Streit
99	1.2	197.9	2	212	4	118	<MDC	<MDC	Streit
100	1.2	197.9	3	187	9	-91	<MDC	<MDC	Streit
101	1.2	197.9	3	314	9	968	<MDC	968	Streit

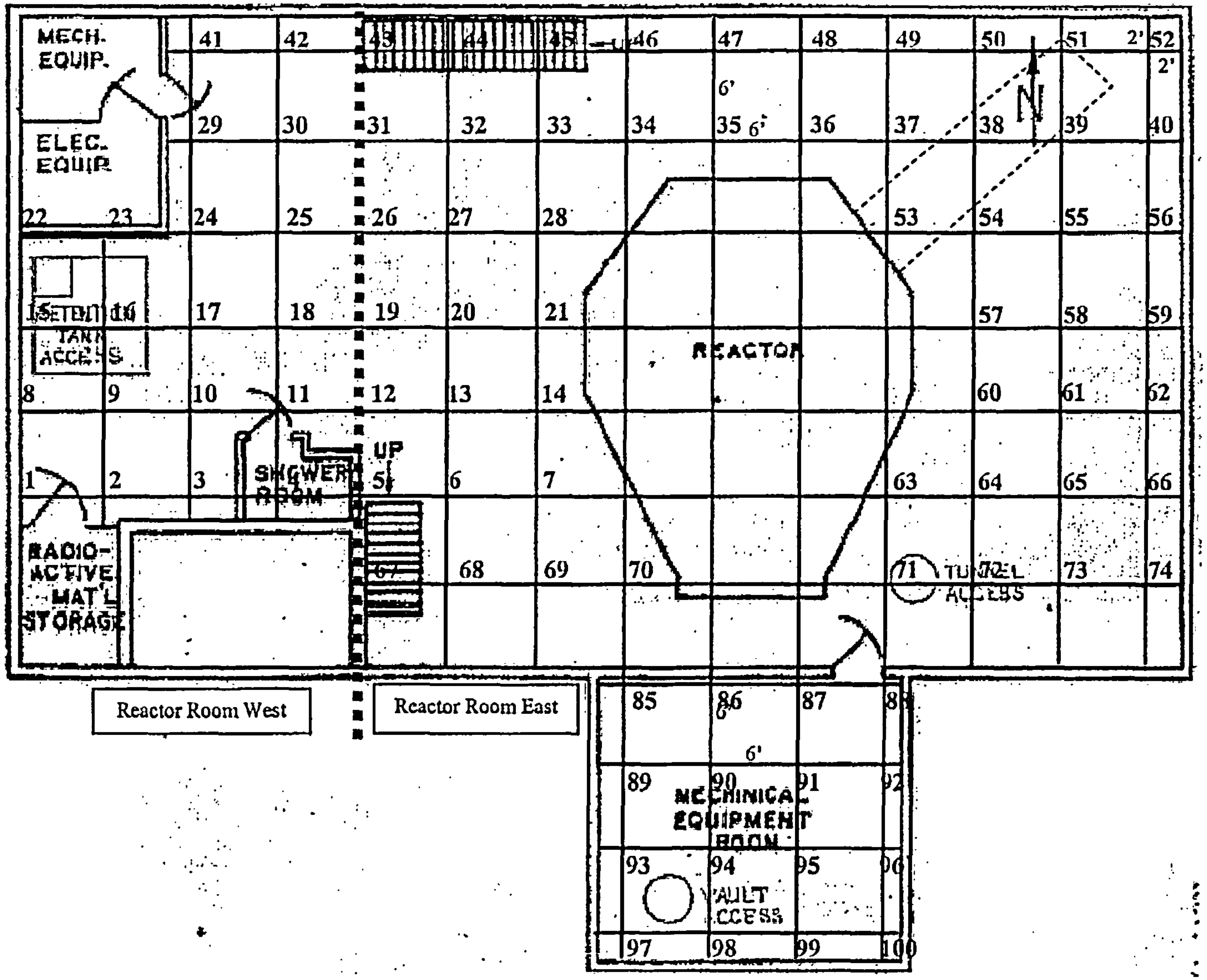
R - rejected due to high background

Scientech, LLC

Project #:	23566				Date:	7/14/2005		
Site Name:	University of Illinois				Location:	Mech. Equip. Rm.		
Instrument ID:	Beckman-LS6500							
MDC(a):	H-3	37	C-14	28 dpm	Total β	28	dpm	
MDC(b):	H-3	42	C-14	27 dpm	Total β	27	dpm	

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net gross β (cpm)		Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
85	17.19	<MDA	5.1	<MDA	30.67	30.7	Streit
86	55.59	55.6	16.74	<MDA	10.67	<MDA	Streit
87	57.66	57.7	12.74	<MDA	34.67	34.7	Streit
88	12.15	<MDA	15	<MDA	28.00	28.0	Streit
89	-10.73	<MDA	-0.55	<MDA	-5.33	<MDA	Streit
90	26.29	<MDA	-0.85	<MDA	25.33	<MDA	Streit
91	44.28	44.3	-0.99	<MDA	18.67	<MDA	Streit
92	66.74	66.7	14.33	<MDA	32.00	32.0	Streit
93	429.92	429.9	-0.86	<MDA	188.00	188.0	Streit
94	34.32	<MDA	7.74	<MDA	4.00	<MDA	Streit
95	56.14	56.1	3.19	<MDA	25.33	<MDA	Streit
96	61.04	61.0	-0.9	<MDA	38.67	38.7	Streit
97	8473.91	8473.9	-90.47	<MDA	405.33	405.3	Streit
98	46.28	46.3	14.77	<MDA	40.00	40.0	Streit
99	31.41	<MDA	7.41	<MDA	28.00	28.0	Streit
100	38.78	<MDA	7.64	<MDA	46.67	46.7	Streit
100 - Recount	25.76	<MDA	7.99	<MDA	-	-	Streit
Additional smears near 93 and 97							
1	1007.77	1007.8	4.13	<MDA			Streit
2	5140.02	5140.0	-34.47	<MDA			Streit
3	103.37	103.4	5.71	<MDA			Streit
4	786.25	786.3	13.99	<MDA			Streit
5	3437.64	3437.6	-29.62	<MDA			Streit
6	26.55	<MDA	11.72	<MDA			Streit

FIGURE A-2-1: REACTOR AND MECHANICAL ROOM SURVEY AND SAMPLING POINTS
(Measurements made at grid intersections)



Scientech, LLC

Project #: 23566
Site Name: University of Illinois
Instrument ID: L-2360 (L)
Efficiency: Alpha: 0.20 Beta: 0.13
MDC: Alpha: 37 Beta: 454 dpm/100cm2

Date: 7/13/2005
Location: Wall Surveys
(Sheet 1 of 2)

Systematic Locations	α Bkgnd (cpm)	β Bkgnd. (cpm)	Gross α (cpm)	Gross β (cpm)	α Activity (dpm/100cm²)	β Activity (dpm/100cm²)	Reportable Alpha	Reportable Beta	Survey Technician
FIXED-POINT MEASUREMENTS									
101	1.5	262.4	3	384	8	935	<MDC	935	Higgins
102	1.5	262.4	3	364	8	782	<MDC	782	Higgins
103	1.5	262.4	2	870	3	R	<MDC	R	Higgins
104	1.5	262.4	5	537	18	R	<MDC	R	Higgins
105	1.5	262.4	5	365	18	789	<MDC	789	Higgins
106	1.5	262.4	0	219	-8	-334	<MDC	<MDC	Higgins
107	1.5	262.4	5	376	18	874	<MDC	874	Higgins
108	1.5	262.4	3	252	8	-80	<MDC	<MDC	Higgins
109	1.5	262.4	5	390	18	R	<MDC	R	Higgins
110	1.5	262.4	4	332	13	535	<MDC	535	Higgins
111	1.5	262.4	5	192	18	-542	<MDC	<MDC	Higgins
112	1.5	262.4	2	1067	3	R	<MDC	R	Higgins
113	1.5	262.4	8	560	33	R	<MDC	R	Higgins
114	1.5	262.4	4	375	13	R	<MDC	R	Higgins
115	1.5	262.4	3	344	8	R	<MDC	R	Higgins
116	1.5	262.4	4	253	13	-72	<MDC	<MDC	Higgins
117	1.5	262.4	7	188	28	-572	<MDC	<MDC	Higgins
118	1.5	262.4	9	257	38	-42	38	<MDC	Higgins
119	1.5	262.4	6	285	23	174	<MDC	<MDC	Higgins
120	1.5	262.4	6	323	23	466	<MDC	466	Higgins
121	1.5	262.4	4	627	13	2805	<MDC	2805	Higgins
122	1.5	262.4	4	288	13	197	<MDC	<MDC	Higgins
123	1.5	262.4	9	243	38	-149	38	<MDC	Higgins
124	1.5	262.4	4	327	13	497	<MDC	497	Higgins
125	1.5	262.4	10	690	43	3289	43	3289	Higgins
126	1.5	262.4	2	725	3	3558	<MDC	3558	Higgins
127	1.5	262.4	6	555	23	2251	<MDC	2251	Higgins
128	1.5	262.4	4	591	13	2528	<MDC	2528	Higgins
129	1.5	262.4	5	363	18	774	<MDC	774	Higgins
130	1.5	262.4	5	670	18	3135	<MDC	3135	Higgins
131	1.5	262.4	7	1035	28	5943	<MDC	5943	Higgins
132	1.5	262.4	7	738	28	R	<MDC	R	Higgins
133	1.5	262.4	0	5458	-8	R	<MDC	R	Higgins
134	1.5	262.4	7	803	28	R	<MDC	R	Higgins
135	1.5	262.4	5	327	18	497	<MDC	497	Higgins
136	1.5	262.4	2	303	3	312	<MDC	<MDC	Higgins
137	1.5	262.4	3	316	8	412	<MDC	<MDC	Higgins
138	1.5	262.4	7	312	28	382	<MDC	<MDC	Higgins
139	1.5	262.4	2	302	3	305	<MDC	<MDC	Higgins
140	1.5	262.4	4	276	13	105	<MDC	<MDC	Higgins
141	1.5	262.4	9	237	38	-195	38	<MDC	Higgins
142	1.5	262.4	7	296	28	258	<MDC	<MDC	Higgins
143	1.5	262.4	7	301	28	297	<MDC	<MDC	Higgins
144	1.5	262.4	2	258	3	-34	<MDC	<MDC	Higgins
145	1.5	262.4	2	251	3	-88	<MDC	<MDC	Higgins

R - Rejected due to high background from the Radioactive Material Storage Cage.
Survey points 121 - 128 are in the small mechanical/electrical room in the northwestern corner of the reactor room.
Survey point 130 and 131 are on the reactor room side of the walls of the small mechanical/electrical room.

Scientech, LLC

Project #:	23566	**Date:**	7/13/2005
Site Name:	University of Illinois	**Location:**	Wall Survey (Sheet 2 of 2)
Instrument ID:	L-2360 (L)		

Efficiency: Alpha: 0.2 Beta: 0.13

MDC: Alpha: 37 Beta: 454 dpm/100cm2

Systematic Locations	α Bkgnd (cpm)	β Bkgnd. (cpm)	Gross α (cpm)	Gross β (cpm)	α Activity (dpm/100cm^2)	β Activity (dpm/100cm^2)	Reportable Alpha	Reportable Beta	Survey Technician
FIXED-POINT MEASUREMENTS									
146	1.5	262.4	7	265	28	20	<MDC	<MDC	Higgins
147	1.2	197.9	5	279	19	624	<MDC	624	Higgins
148	1.2	197.9	7	257	29	455	<MDC	455	Higgins
149	1.2	197.9	5	299	19	778	<MDC	778	Higgins
150	1.2	197.9	1	391	-1	R	<MDC	R	Higgins
151	1.2	197.9	4	450	14	R	<MDC	R	Higgins
152	1.2	197.9	2	422	4	R	<MDC	R	Higgins
153	1.2	197.9	7	548	29	R	<MDC	R	Higgins
154	1.2	197.9	5	939	19	R	<MDC	R	Higgins
155	1.2	197.9	4	682	14	R	<MDC	R	Higgins
156	1.2	197.9	4	267	14	532	<MDC	532	Higgins
157	1.2	197.9	6	257	24	455	<MDC	455	Higgins
158	1.2	197.9	7	301	29	793	<MDC	793	Higgins
159	1.2	197.9	9	305	39	824	39	824	Higgins
160	1.2	197.9	5	307	19	839	<MDC	839	Higgins
161	1.2	197.9	3	1385	9	R	<MDC	R	Higgins
162	1.2	197.9	4	392	14	R	<MDC	R	Higgins
163	1.2	197.9	6	407	24	R	<MDC	R	Higgins
164	1.2	197.9	4	352	14	R	<MDC	R	Higgins
165	1.2	197.9	4	279	14	624	<MDC	624	Higgins
166	1.2	197.9	5	261	19	485	<MDC	485	Higgins
167	1.2	197.9	2	303	4	808	<MDC	808	Higgins
168	1.2	197.9	7	294	29	739	<MDC	739	Higgins
169	1.2	197.9	7	342	29	1108	<MDC	1108	Higgins
170	1.2	197.9	6	258	24	462	<MDC	462	Higgins
171	1.2	197.9	6	285	24	670	<MDC	670	Higgins
172	1.2	197.9	5	358	19	1232	<MDC	1232	Higgins
173	1.2	197.9	3	293	9	732	<MDC	732	Higgins
174	1.2	197.9	5	286	19	678	<MDC	678	Higgins
175	1.2	197.9	4	280	14	632	<MDC	632	Higgins
176	1.2	197.9	5	285	19	670	<MDC	670	Higgins
177	1.2	197.9	7	381	29	1408	<MDC	1408	Higgins
178	1.2	197.9	8	264	34	508	<MDC	508	Higgins
179	1.2	197.9	3	347	9	1147	<MDC	1147	Higgins
180	1.2	197.9	5	285	19	670	<MDC	670	Higgins
181	1.2	197.9	7	255	29	439	<MDC	<MDC	Higgins
185	1.2	197.9	9	220	39	170	39	<MDC	Higgins
186	1.2	197.9	6	215	24	132	<MDC	<MDC	Higgins
187	1.2	197.9	6	211	24	101	<MDC	<MDC	Higgins
188	1.2	197.9	6	192	24	-45	<MDC	<MDC	Higgins
189	1.2	197.9	4	228	14	232	<MDC	<MDC	Higgins
190	1.2	197.9	6	258	24	462	<MDC	462	Higgins
191	1.2	197.9	4	268	14	539	<MDC	539	Higgins
192	1.2	197.9	6	203	24	39	<MDC	<MDC	Higgins

R - Rejected due to high background in the area of the survey point.
Survey points 162 - 181 are on the reactor bioshield.

Scientech, LLC

Project #:	23566	**Date:**	7/15/2005
Site Name:	University of Illinois	**Location:**	Wall Survey
Instrument ID:	Beckman-LS6500		(Sheet 1 of 2)

MDA(a):	H-3	44	C-14	24 dpm	**Total β**	24 dpm
MDA(b):	H-3		C-14	dpm	**Total β**	dpm

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net total β (dpm)	Reportabe Total β	Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
101	-4.61	<MDA	26.79	26.8	46.67	46.7	Higgins
102	-26.56	<MDA	6.94	<MDA	26.67	26.7	Higgins
103	-16.65	<MDA	6.87	<MDA	9.33	<MDA	Higgins
104	-2.51	<MDA	9.39	<MDA	-6.67	<MDA	Higgins
105	-0.54	<MDA	9.38	<MDA	13.33	<MDA	Higgins
106	-22.62	<MDA	23.01	<MDA	12.00	<MDA	Higgins
107	-12.65	<MDA	12.20	<MDA	12.00	<MDA	Higgins
108	-11.63	<MDA	0.20	<MDA	4.00	<MDA	Higgins
109	-9.91	<MDA	8.08	<MDA	10.67	<MDA	Higgins
110	9.68	<MDA	5.75	<MDA	-1.33	<MDA	Higgins
111	2.36	<MDA	11.27	<MDA	17.33	<MDA	Higgins
112	22.42	<MDA	1.37	<MDA	16.00	<MDA	Higgins
113	2.87	<MDA	6.85	<MDA	0.00	<MDA	Higgins
114	-14.41	<MDA	7.16	<MDA	24.00	24.0	Higgins
115	-10.15	<MDA	21.23	<MDA	12.00	<MDA	Higgins
116	-22.67	<MDA	12.24	<MDA	6.67	<MDA	Higgins
117	-13.00	<MDA	8.21	<MDA	21.33	<MDA	Higgins
118	-0.32	<MDA	16.97	<MDA	14.67	<MDA	Higgins
119	6.43	<MDA	14.00	<MDA	21.33	<MDA	Higgins
120	106.13	106.1	9.54	<MDA	77.33	77.3	Higgins
121	-10.42	<MDA	12.49	<MDA	-5.33	<MDA	Higgins
122	-4.78	<MDA	5.57	<MDA	13.33	<MDA	Higgins
123	-3.33	<MDA	-3.75	<MDA	8.00	<MDA	Higgins
124	-7.57	<MDA	9.71	<MDA	-9.33	<MDA	Higgins
125	-4.88	<MDA	4.32	<MDA	25.33	25.3	Higgins
126	-16.37	<MDA	8.33	<MDA	33.33	33.3	Higgins
127	-8.59	<MDA	1.76	<MDA	26.67	26.7	Higgins
128	-5.36	<MDA	6.91	<MDA	49.33	49.3	Higgins
129	-17.17	<MDA	8.30	<MDA	40.00	40.0	Higgins
130	-21.13	<MDA	8.23	<MDA	41.33	41.3	Higgins
131	-0.57	<MDA	0.07	<MDA	18.67	<MDA	Higgins
132	2.63	<MDA	13.42	<MDA	9.33	<MDA	Higgins
133	-11.50	<MDA	-2.54	<MDA	46.67	46.7	Higgins
134	-12.76	<MDA	12.18	<MDA	26.67	26.7	Higgins
135	-21.60	<MDA	14.94	<MDA	9.33	<MDA	Higgins
136	-16.01	<MDA	16.27	<MDA	40.00	40.0	Higgins
137	-17.60	<MDA	8.12	<MDA	20.00	<MDA	Higgins
138	-8.08	<MDA	4.11	<MDA	2.67	<MDA	Higgins
139	18.69	<MDA	9.45	<MDA	46.67	46.7	Higgins
140	-1.94	<MDA	7.13	<MDA	0.00	<MDA	Higgins
141	3.96	<MDA	12.46	<MDA	32.00	32.0	Higgins
142	-2.68	<MDA	11.29	<MDA	33.33	33.3	Higgins
143	-2.89	<MDA	4.44	<MDA	32.00	32.0	Higgins
144	-26.68	<MDA	1.52	<MDA	12.00	<MDA	Higgins
145	-18.79	<MDA	12.20	<MDA	20.00	<MDA	Higgins

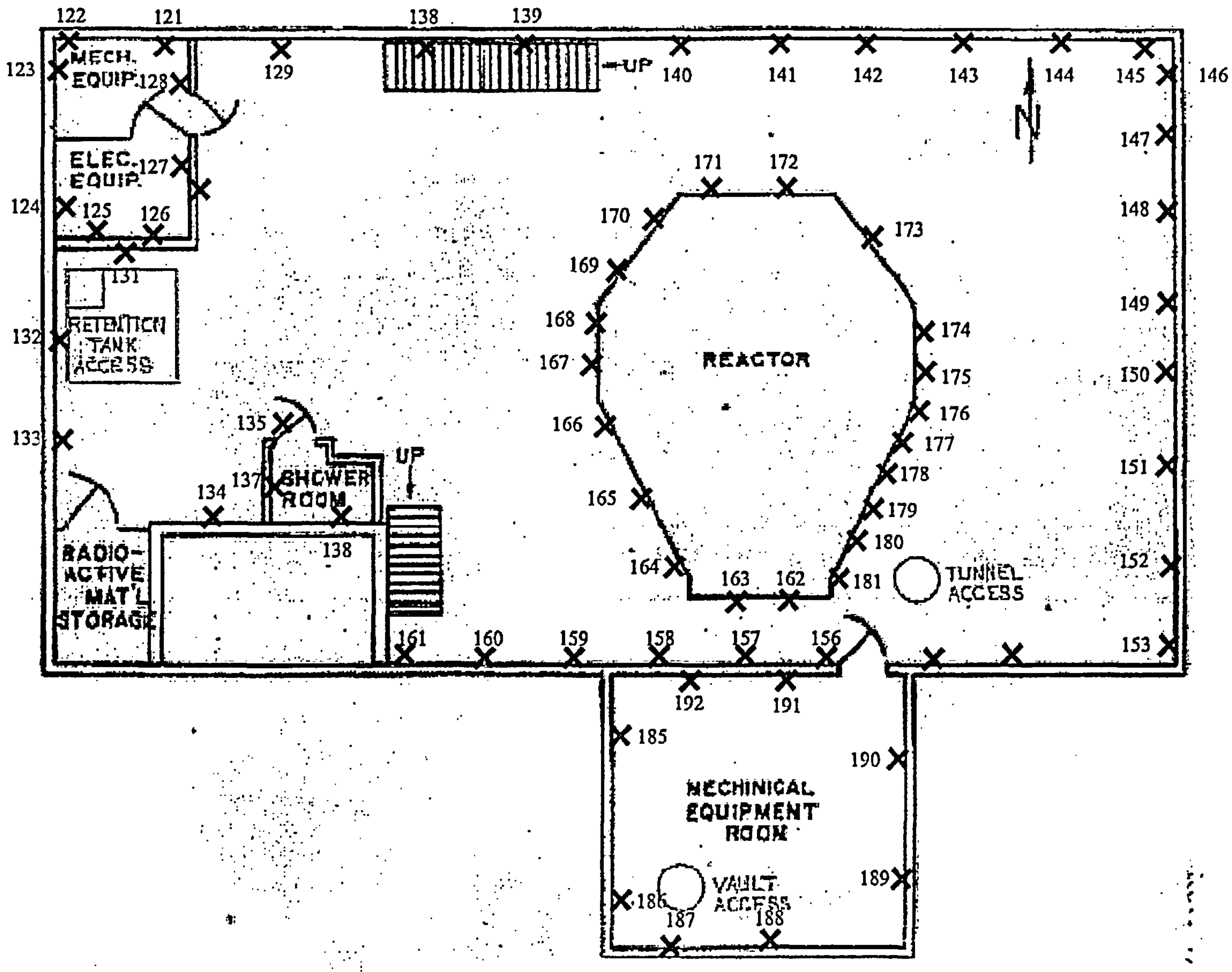
Survey points 101 - 120 are from the mezzanine level, loading bay, and storage level.

Scientech, LLC

Project #:	23566					Date:	7/15/2005	
Site Name:	University of Illinois					Location:	Wall Survey	
Instrument ID:	Beckman-LS6500						(Sheet 2 of 2)	
MDA(a):	H-3	44	C-14	24	dpm	Total β	24	
MDA(b):	H-3	47	C-14	25	dpm	Total β	25	

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net total β (dpm)	Reportabe Total β	Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
146	4.15	<MDA	0.15	<MDA	14.67	<MDA	Higgins
147	131.85	131.9	5.86	<MDA	105.33	105.3	Higgins
148	-15.00	<MDA	6.81	<MDA	21.33	<MDA	Higgins
149	-7.62	<MDA	9.92	<MDA	21.33	<MDA	Higgins
150	-5.71	<MDA	8.09	<MDA	17.33	<MDA	Higgins
151	-6.78	<MDA	1.45	<MDA	5.33	<MDA	Higgins
152	-21.18	<MDA	8.18	<MDA	25.33	25.3	Higgins
153	-12.77	<MDA	8.20	<MDA	33.33	33.3	Higgins
154	22.91	<MDA	691.26	691.3	1077.33	1077.3	Higgins
155	-11.26	<MDA	23.42	<MDA	32.00	32.0	Higgins
156	-6.96	<MDA	6.77	<MDA	21.33	<MDA	Higgins
157	-5.31	<MDA	1.52	<MDA	37.33	37.3	Higgins
158	-17.44	<MDA	7.00	<MDA	29.33	29.3	Higgins
159	-18.10	<MDA	4.35	<MDA	14.67	<MDA	Higgins
160	-4.78	<MDA	12.86	<MDA	20.00	<MDA	Higgins
161	-13.28	<MDA	13.15	<MDA	18.67	<MDA	Higgins
162	-14.80	<MDA	1.48	<MDA	26.67	26.7	Higgins
163	-10.82	<MDA	12.60	<MDA	13.33	<MDA	Higgins
164	-17.25	<MDA	20.28	<MDA	64.00	64.0	Higgins
165	-17.76	<MDA	10.91	<MDA	17.33	<MDA	Higgins
166	5.95	<MDA	9.50	<MDA	14.67	<MDA	Higgins
167	-8.23	<MDA	5.46	<MDA	38.67	38.7	Higgins
168	-10.67	<MDA	1.61	<MDA	22.67	<MDA	Higgins
169	4.22	<MDA	-2.57	<MDA	18.67	<MDA	Higgins
170	-13.33	<MDA	-3.78	<MDA	53.33	53.3	Higgins
171	-15.96	<MDA	16.22	<MDA	30.67	30.7	Higgins
172	-16.72	<MDA	1.68	<MDA	34.67	34.7	Higgins
173	-24.96	<MDA	11.17	<MDA	17.33	<MDA	Higgins
174	-13.05	<MDA	9.57	<MDA	25.33	25.3	Higgins
175	-13.73	<MDA	-1.06	<MDA	-16.00	<MDA	Higgins
176	-1.41	<MDA	8.37	<MDA	14.67	<MDA	Higgins
177	-19.86	<MDA	20.65	<MDA	-10.67	<MDA	Higgins
178	-24.58	<MDA	9.78	<MDA	-28.00	<MDA	Higgins
179	-10.75	<MDA	3.58	<MDA	-10.67	<MDA	Higgins
180	-23.85	<MDA	16.28	<MDA	-14.67	<MDA	Higgins
181	-6.94	<MDA	-1.18	<MDA	6.67	<MDA	Higgins
185	-4.20	<MDA	4.45	<MDA	-9.33	<MDA	Higgins
186	92.45	92.5	-0.05	<MDA	8.00	<MDA	Higgins
187	-19.73	<MDA	9.83	<MDA	-4.00	<MDA	Higgins
188	-22.75	<MDA	18.07	<MDA	-9.33	<MDA	Higgins
189	-11.99	<MDA	6.05	<MDA	-20.00	<MDA	Higgins
190	-8.51	<MDA	8.55	<MDA	-10.67	<MDA	Higgins
191	-17.60	<MDA	3.20	<MDA	-12.00	<MDA	Higgins
192	-3.54	<MDA	5.62	<MDA	-28.00	<MDA	Higgins
193	-18.79	<MDA	12.20	<MDA	20.00	<MDA	Higgins

FIGURE A-2-2: REACTOR AND MECHANICAL ROOM WALL SURVEY POINTS



APPENDIX A-3

TUNNEL SURVEYS AND SAMPLES

Scientech, LLC

Project #:	23566				**Date:**	7/13/2015
Site Name:	University of Illinois				**Location:**	Pipe Chase Tunnel
Instrument ID:	L-44-9 (B)					
Efficiency:	Alpha:	-	Beta:	0.15		
MDC:	Alpha:	-	Beta:	175	dpm/100cm2	

Systematic Locations	α Bkgnd (cpm)	β Bkgnd. (cpm)	Gross α (cpm)	Gross β (cpm)	α Activity (dpm/100cm^2)	β Activity (dpm/100cm^2)	Reportable Alpha	Reportable Beta	Survey Technician
FIXED-POINT MEASUREMENTS									
1	-	45.2	-	62	-	112	-	<MDC	Taylor
2	-	45.2	-	43	-	-15	-	<MDC	Taylor
3	-	45.2	-	55	-	65	-	<MDC	Taylor
4	-	45.2	-	79	-	225	-	225	Taylor
5	-	45.2	-	94	-	325	-	325	Taylor
6	-	45.2	-	2141	-	R	-	R	Taylor
	Ex. Rate (uR/hr)								
1	7								Taylor
2	7								Taylor
3	6								Taylor
4	6								Taylor
5	40								Taylor
6	200								Taylor

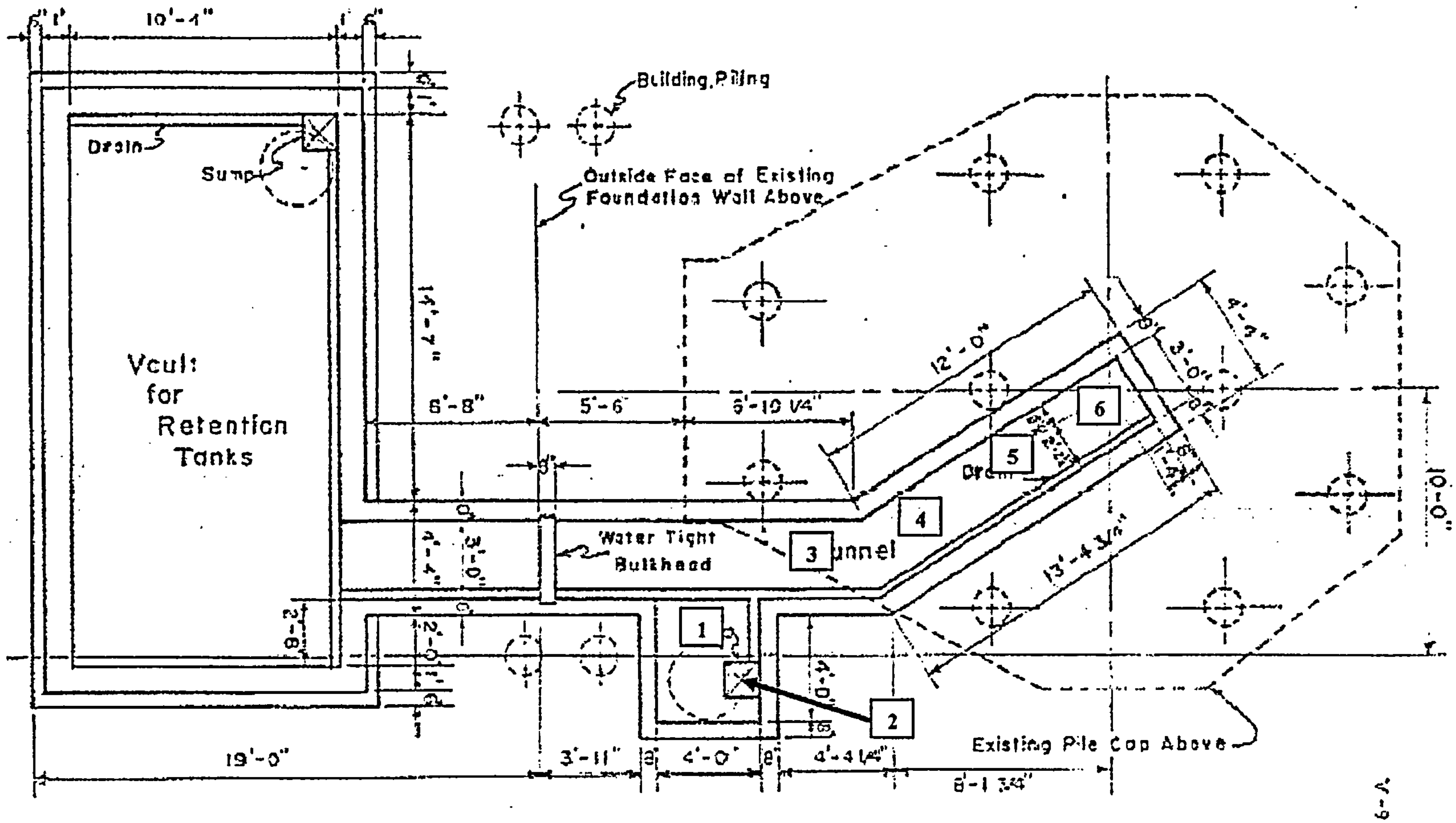
R - rejected due to high background

Scientech, LLC

Project #:	23566				Date:	7/14/2005	
Site Name:	University of Illinois				Location:	Pipe Chase Tunnel	
Instrument ID:	Beckman-LS6500						
MDA(a):	H-3	32	C-14	29 dpm	Total β	29	dpm
MDA(b):	H-3		C-14	dpm	Total β		dpm

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net total β (dpm)	Reportabe Total β	Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
1	-4.46	<MDA	26.97	<MDA	5.33	<MDA	Taylor
2	10.05	<MDA	8.22	<MDA	18.67	<MDA	Taylor
3	-19.97	<MDA	-2.92	<MDA	-13.33	<MDA	Taylor
4	23.51	<MDA	4.43	<MDA	22.67	<MDA	Taylor
5	-10.61	<MDA	8.85	<MDA	6.67	<MDA	Taylor
6	11.56	<MDA	2.11	<MDA	10.67	<MDA	Taylor

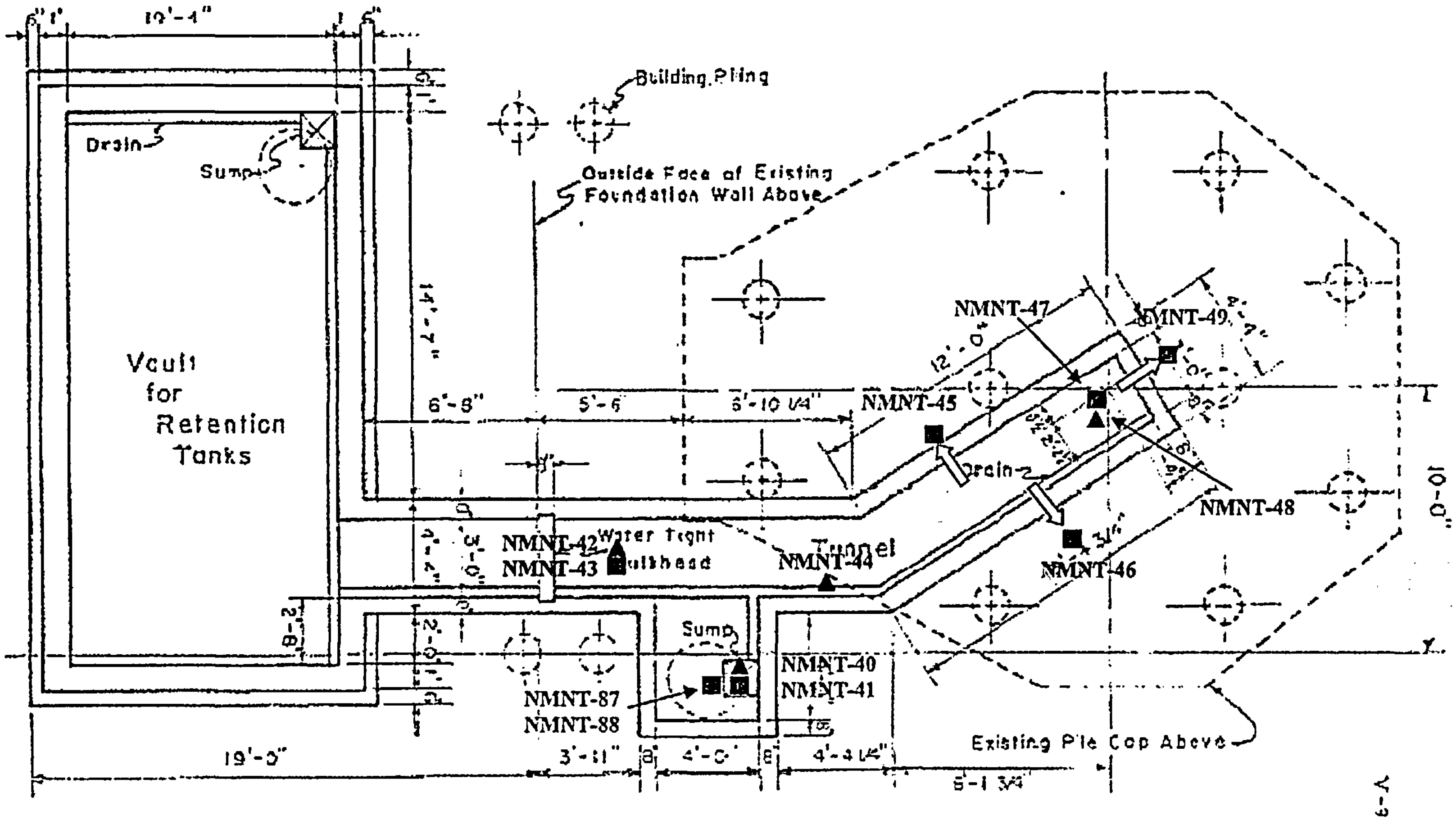
FIGURE A-3-2: TUNNEL SURVEY POINTS



TUNNEL CONCRETE AND SOIL SAMPLE ANALYSIS RESULTS

Sample Number	Description	Gross Tritium (cpm)	Estimated Tritium (pCi/g)	Total Beta (net cpm)	Cobalt-60 (pCi/g)	Europium-152 (pCi/g)	Nickel-63 (pCi/g)	Iron-55 (pCi/g)
CONCRETE								
NMNT-40	Sump	306	< MDA	31,005	23.6 +/- 1.8	18.7 +/- 1.7		
NMNT-42	Floor	317	< MDA	23,399	< MDA	< MDA		
NMNT-44	Floor	278	< MDA	17,192	< MDA	< MDA		
NMNT-47	Floor	288	< MDA	35,142	< MDA	< MDA		
NMNT-50	Ceiling	305	17600	109,439	< MDA	< MDA		
NMNT-51	Ceiling	1,029	55,900	270,080	< MDA	< MDA	< MDA	30 +/- 13
NMNT-52	Ceiling	457	36,900	222,702	< MDA	< MDA	< MDA	< MDA
NMNT-53	Ceiling	455	18,000	73,742	< MDA	< MDA	< MDA	10 +/- 21
SOIL								
NMNT-41	Sump	317	< MDA	< MDA	4.85	< MDA	< MDA	< MDA
NMNT-43	Floor	256	< MDA	< MDA	13.9	< MDA		
NMNT-45	Wall	222	< MDA	< MDA	< MDA	< MDA		
NMNT-46	Wall	279	< MDA	20,961	< MDA	< MDA		
NMNT-48	Floor	236	< MDA	26,201	< MDA	< MDA	< MDA	22.5 +/- 8.9
NMNT-49	Wall	191	< MDA	16,197	< MDA	< MDA		
NMNT-87	Floor	288	< MDA	< MDA	< MDA	< MDA		
NMNT-88	Floor	183	< MDA	< MDA	1.14 +/- 0.3	< MDA		

FIGURE A-3-1: TUNNEL SAMPLING POINTS



■ Soil Sample Locations

▲ Concrete Samples

⇨ Trench Wall Penetration

APPENDIX A-4

BIOSHIELD SURVEYS AND SAMPLES

GAMMA SPECTROSCOPY RESULTS OF BIOSHIELD CONCRETE SAMPLES

Sample Number	Media	Location	Isotope	Weighted Mean Concentration (uCi/g)	+/- Error (1 sigma)	MDA	
NMNT-01	Concrete	Bkgd Bioshield	K-40	3.59E-06	1.07E-05	3.86E-05	<MDA
			Co-60	3.52E-06	8.24E-07	3.25E-06	-
NMNT-20	Concrete	Bioshield Loc. 17	K-40	4.89E-04	4.96E-05	1.39E-04	-
			Co-60	8.39E-03	1.83E-04	6.07E-05	-
			Eu-152	9.01E-03	1.42E-04	8.79E-05	-
			Eu-154	5.78E-04	2.28E-05	8.01E-05	-
NMNT-22*	**Concrete**	**Bioshield Loc. 18-1**	**K-40**	**-9.41E-08**	**9.34E-06**	**3.46E-05**	**<MDA**
			Co-60	**3.07E-05**	**1.72E-06**	**3.42E-06**	**-**
			Eu-152	**3.31E-05**	**1.89E-06**	**5.13E-06**	**-**
NMNT-23	Concrete	Bioshield Loc. 18-2	K-40	1.31E-05	9.53E-06	3.17E-05	<MDA
NMNT-24	Concrete	Bioshield Loc. 18-3	Co-60	9.78E-07	6.10E-07	2.82E-06	<MDA
			Ra-226+D	1.29E-06	4.47E-06	1.82E-06	<MDA
NMNT-25	Concrete	Bioshield Loc. 19-1	K-40	1.38E-07	3.67E-06	1.35E-05	<MDA
			Co-60	3.65E-07	2.61E-07	1.22E-06	<MDA
NMNT-26	Concrete	Bioshield Loc. 19-2	K-40	5.75E-06	4.31E-06	1.44E-05	<MDA
			Co-60	1.51E-05	7.59E-07	1.39E-06	-
			Eu-152	1.79E-05	9.01E-07	2.22E-06	-
NMNT-27	Concrete	Bioshield Loc. 20-1	K-40	0.00E+00	5.88E-06	2.19E-05	<MDA
			Co-60	1.96E-06	4.65E-07	1.77E-06	-
			Cs-137	2.97E-07	2.46E-07	8.43E-07	<MDA
NMNT-28	Concrete	Bioshield Loc. 20-2	Co-60	8.08E-06	8.94E-07	2.67E-06	-
			Eu-152	7.13E-06	8.77E-07	2.93E-06	-
NMNT-29	Concrete	Bioshield Loc. 20-3	K-40	-1.47E-06	1.09E-05	4.08E-05	<MDA
			Co-60	2.31E-05	1.62E-06	3.85E-06	-
			Eu-152	3.13E-05	1.95E-06	5.75E-06	-
NMNT-30	Concrete	Bioshield Loc. 20-4	K-40	6.32E-06	1.10E-05	3.88E-05	<MDA
			Co-60	6.67E-05	2.83E-06	4.37E-06	-
			Eu-152	5.91E-05	2.85E-06	7.20E-06	-
NMNT-31	Concrete	Bioshield Loc. 20-5	Co-60	8.01E-05	2.92E-06	3.10E-06	-
			Eu-152	7.90E-05	3.22E-06	6.69E-06	-
NMNT-32*	**Concrete**	**Bioshield Loc. 21-1**	**K-40**	**-4.60E-07**	**5.89E-06**	**2.17E-05**	**<MDA**
			Co-60	**1.26E-05**	**8.51E-07**	**2.02E-06**	**-**
			Eu-152	**9.39E-06**	**8.08E-07**	**2.55E-06**	**-**
NMNT-33	Concrete	Bioshield Loc. 21-2	K-40	4.92E-06	5.85E-06	2.03E-05	<MDA
			Co-60	1.22E-05	8.17E-07	1.78E-06	-
			Cs-137	6.51E-07	3.43E-07	1.09E-06	<MDA
			Eu-152	1.02E-05	8.19E-07	2.50E-06	-

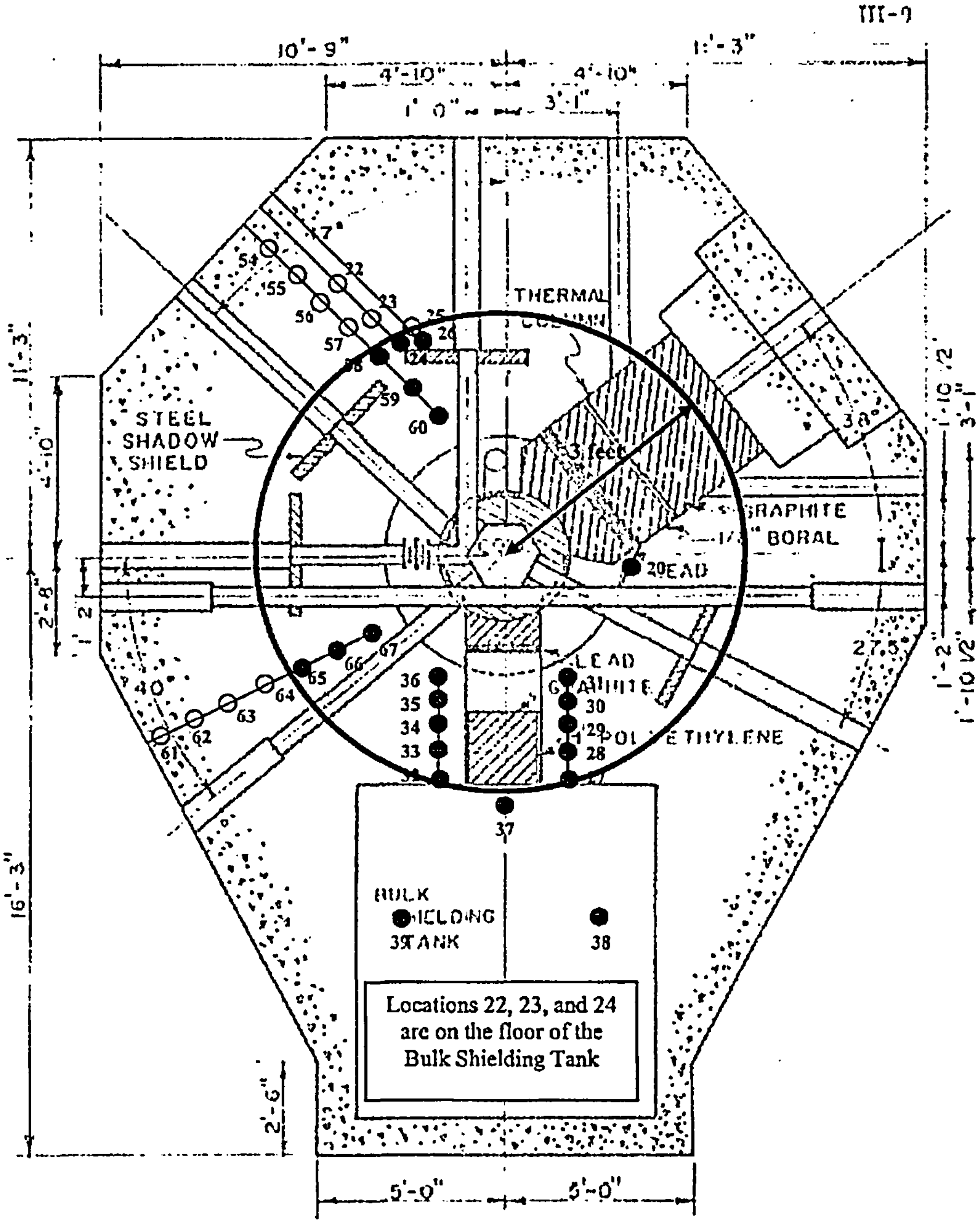
GAMMA SPECTROSCOPY RESULTS OF BIOSHIELD CONCRETE SAMPLES

Sample Number	Media	Location	Isotope	Weighted Mean Concentration (uCi/g)	+/- Error (1 sigma)	MDA	
NMNT-34	Concrete	Bioshield Loc. 21-3	K-40	1.38E-05	1.01E-05	3.37E-05	<MDA
			Co-60	5.89E-05	2.46E-06	3.57E-06	-
			Eu-152	5.26E-05	2.55E-06	6.53E-06	-
NMNT-35	Concrete	Bioshield Loc. 21-4	Co-60	2.91E-04	7.80E-06	6.36E-06	-
			Eu-152	2.62E-04	7.62E-06	1.23E-05	-
NMNT-36	Concrete	Bioshield Loc. 21-5	Co-60	5.33E-04	1.27E-05	6.30E-06	-
			Eu-152	5.34E-04	1.21E-05	1.56E-05	-
			Eu-154	3.26E-05	4.28E-06	1.35E-05	-
NMNT-37	Conctrete	BST Floor Loc. 22	K-40	6.51E-06	1.23E-05	4.34E-05	<MDA
			Co-60	2.54E-04	7.20E-06	5.01E-06	-
			Cs-137	3.01E-06	1.35E-06	4.32E-06	<MDA
			Eu-152	8.15E-05	3.99E-06	1.09E-05	-
			Eu-154	1.06E-05	2.45E-06	7.19E-06	-
NMNT-38	Concrete	BST Floor Loc. 23	K-40	-5.93E-06	1.27E-05	4.85E-05	<MDA
			Co-60	8.33E-05	3.50E-06	5.54E-06	-
			Cs-137	2.13E-06	8.93E-07	2.72E-06	<MDA
			Eu-152	8.91E-06	1.49E-06	6.13E-06	-
NMNT-39	Concrete	BST Floor Loc. 24	K-40	-1.29E-07	1.34E-05	4.96E-05	<MDA
			Co-60	2.86E-06	9.62E-07	4.12E-06	<MDA
NMNT-54	Concrete	Bioshield Loc. 34-1	K-40	-3.79E-06	5.77E-06	2.24E-05	<MDA
			Co-60	1.90E-06	4.91E-07	1.95E-06	<MDA
NMNT-55	Concrete	Bioshield Loc. 34-2	K-40	8.90E-06	6.93E-06	2.32E-05	<MDA
NMNT-56	Concrete	Bioshield Loc. 34-3	Co-60	1.27E-06	6.38E-07	2.76E-06	<MDA
NMNT-57	Concrete	Bioshield Loc. 34-4	K-40	-1.66E-07	8.12E-06	3.00E-05	<MDA
NMNT-58	Concrete	Bioshield Loc. 34-5	K-40	3.35E-06	7.49E-06	2.68E-05	<MDA
			Co-60	8.88E-07	5.04E-07	2.23E-06	<MDA
NMNT-59	Concrete	Bioshield Loc. 34-6	K-40	5.50E-06	9.13E-06	3.24E-05	<MDA
			Co-60	6.40E-06	8.77E-07	2.99E-06	-
			Eu-152	4.44E-06	5.54E-07	3.72E-06	-
NMNT-60	Concrete	Bioshield Loc. 34-7	K-40	-1.60E-05	1.17E-05	4.51E-05	<MDA
			Co-60	3.88E-04	9.93E-06	6.63E-06	-
			Eu-152	4.76E-04	1.15E-05	1.55E-05	-
			Eu-154	2.69E-05	3.81E-06	1.02E-05	-
NMNT-61	Concrete	Bioshield Loc. 35-1	K-40	5.68E-06	6.20E-06	2.14E-05	<MDA
			Co-60	1.59E-06	4.36E-07	1.74E-06	<MDA
NMNT-62*	**Concrete**	**Bioshield Loc. 35-2**	**K-40**	**-1.61E-06**	**1.13E-05**	**4.19E-05**	**<MDA**
			Co-60	**5.55E-05**	**2.55E-06**	**4.72E-06**	**-**
			Eu-152	**6.02E-05**	**2.86E-06**	**7.28E-06**	**-**

GAMMA SPECTROSCOPY RESULTS OF BIOSHIELD CONCRETE SAMPLES

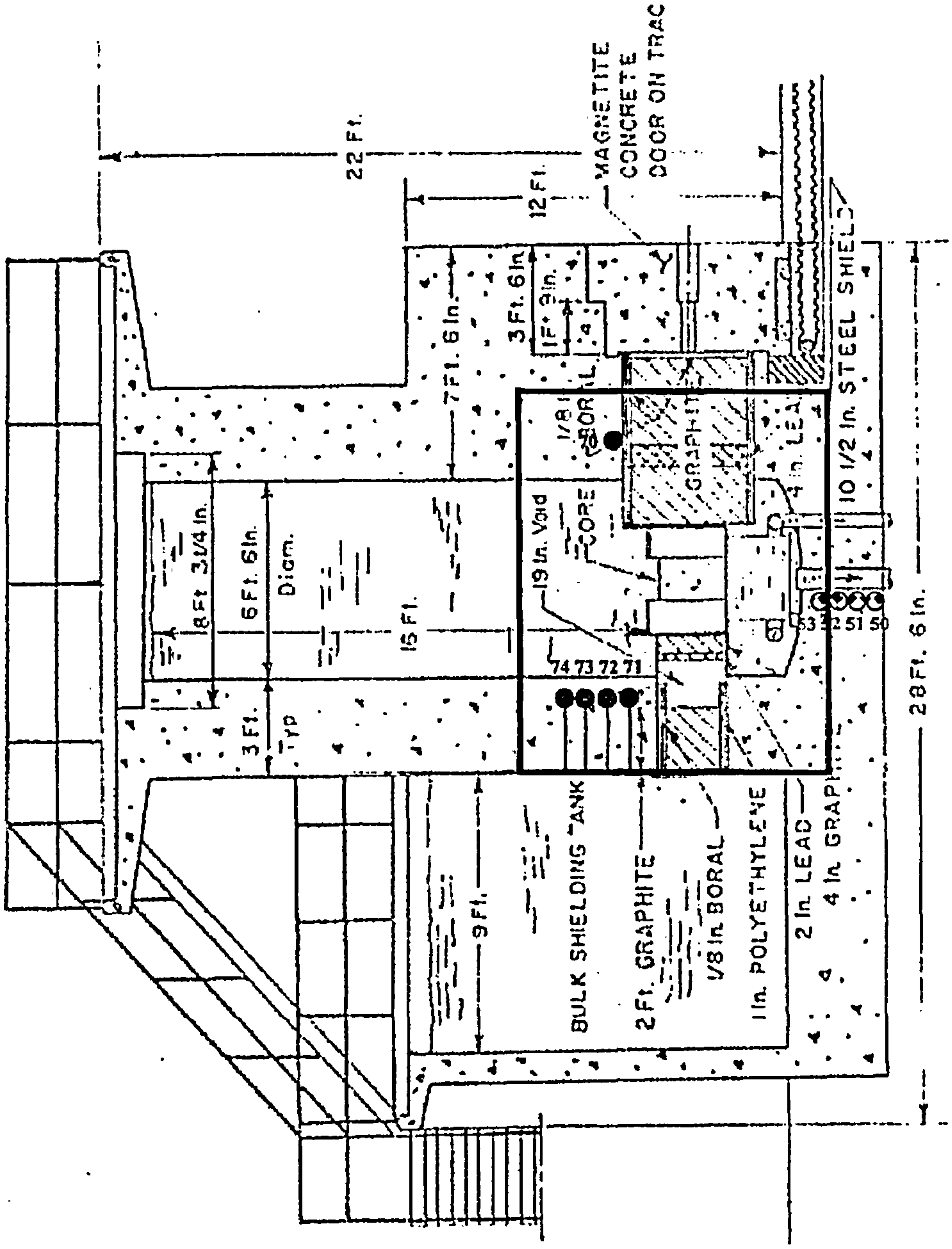
Sample Number	Media	Location	Isotope	Weighted Mean Concentration (uCi/g)	+/- Error (1 sigma)	MDA	
NMNT-63	Concrete	Bioshield Loc. 35-3	K-40	1.82E-06	5.49E-06	1.98E-05	<MDA
			Co-60	1.02E-06	3.92E-07	1.74E-06	<MDA
NMNT-64	Concrete	Bioshield Loc. 35-4	K-40	-3.43E-07	9.62E-06	3.56E-05	<MDA
			Co-60	2.13E-06	6.89E-07	2.80E-06	<MDA
NMNT-65	Concrete	Bioshield Loc. 35-5	K-40	-5.30E-06	1.21E-05	4.57E-05	<MDA
			Co-60	1.75E-05	1.50E-06	3.94E-06	-
			Eu-152	1.09E-05	1.40E-06	4.87E-06	-
NMNT-66	Concrete	Bioshield Loc. 35-6	Co-60	2.32E-04	6.29E-06	4.51E-06	-
			Eu-152	2.06E-04	5.76E-06	9.70E-06	-
			Eu-154	1.29E-05	2.49E-06	7.35E-06	-
NMNT-67	Concrete	Bioshield Loc. 35-7	Co-60	3.55E-03	7.83E-05	2.89E-05	-
			Eu-152	3.89E-03	6.54E-05	5.03E-05	-
			Eu-154	2.29E-04	1.39E-05	4.18E-05	-
NMNT-70	Concrete	Therm. Col. Loc. 38	K-40	8.21E-06	9.28E-06	3.20E-05	<MDA
			Co-60	2.03E-04	5.67E-06	4.59E-06	-
			Eu-152	2.41E-04	6.17E-06	9.44E-06	-
			Eu-154	2.37E-05	2.90E-06	6.78E-06	-
NMNT-71	Concrete	Bioshield Loc. 39	K-40	0.00E+00	1.29E-05	4.75E-05	<MDA
			Co-60	2.65E-04	7.63E-06	6.38E-06	-
			Eu-152	1.81E-04	6.43E-06	1.13E-05	-
			Eu-154	1.48E-05	3.45E-06	1.00E-05	-
NMNT-72	Concrete	Bioshield Loc. 40	K-40	-5.94E-06	7.80E-06	3.10E-05	<MDA
			Co-60	2.68E-05	1.49E-06	2.98E-06	-
			Eu-152	1.59E-05	1.24E-06	4.44E-06	-
NMNT-73	Concrete	Bioshield Loc. 41	K-40	2.89E-06	5.09E-06	1.80E-05	<MDA
			Co-60	4.74E-05	1.69E-06	1.89E-06	-
			Eu-152	3.87E-05	1.66E-06	4.11E-06	-
NMNT-74	Concrete	Bioshield Loc. 42	K-40	2.44E-07	8.10E-06	2.98E-05	<MDA
			Co-60	6.02E-06	7.67E-07	2.48E-06	-
			Eu-152	5.15E-06	7.65E-07	2.97E-06	-

CONCRETE SAMPLES AT REACTOR CORE CENTERLINE LEVEL
(Figure 1 of 2)



- ● Potentially activated or contaminated
- ○ Likely not impacted

ADDITIONAL BIOSHIELD CONCRETE SAMPLES
(Figure 2 of 2)



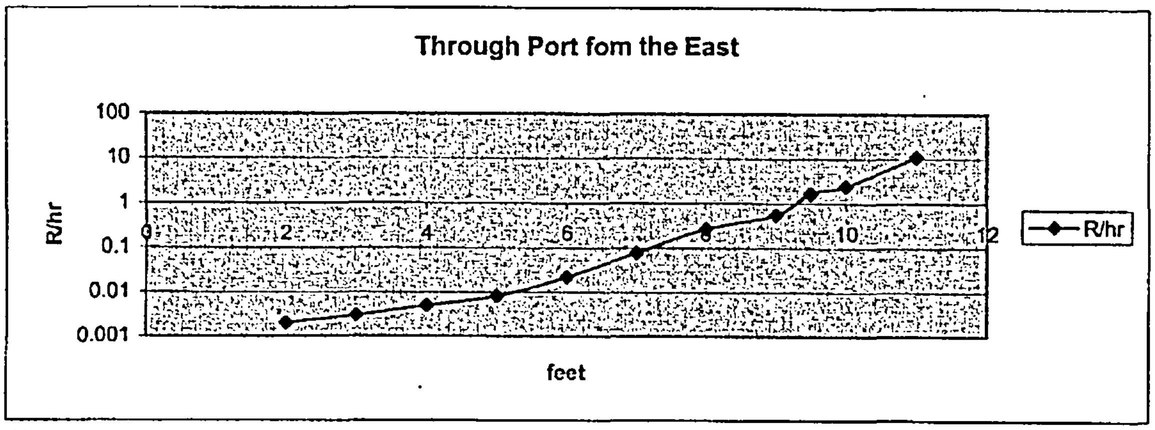
- ● Potentially activated or contaminated
- ◐ Contaminated

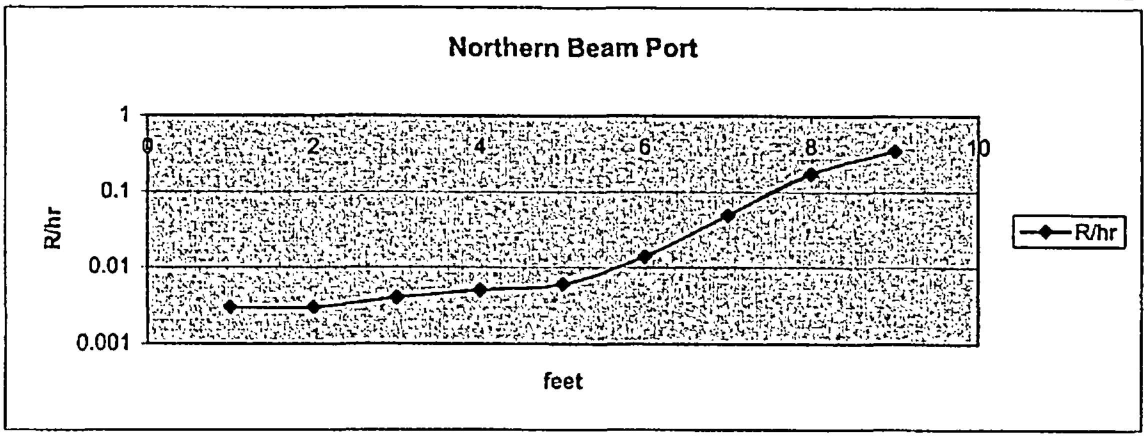
Beam Port Surveys - University of Illinois
22-Jul-05

Through Port From the East	
feet	R/hr
1	0
2	0.002
3	0.003
4	0.005
5	0.008
6	0.022
7	0.076
8	0.266
9	0.548
9.5	1.66
10	2.4
11	10.8

Northern Beam Port	
feet	R/hr
1	0.003
2	0.003
3	0.004
4	0.005
5	0.006
6	0.014
7	0.049
8	0.174
9	0.35

Instrument:
Eberline RO-7 # 480
Probes:
RO7LD # 726968
(low range)
RO7BM # 714855
(medium range)





Survey Cover Sheet

Surveyor: Katie Streit

Date: 7/22/2005

Reviewer: Kevin Taylor

Date: 9/27/2005

Survey Meters

Meter Model #	44-9	Meter Model #	Beckman 6500	Meter Model #	Beckman 6500
Meter / Probe Serial #	169226	Meter / Probe Serial #		Meter / Probe Serial #	
Cal Due	12/6/2005	Cal Due		Cal Due	
Efficiency	15%	Efficiency	54%	Efficiency	75%
Type α, β γ	β	Type	H-3	Type	Total β
Sample Time	1	Sample Time	1	Sample Time	1
Background Time	10	Background Time	1	Background Time	1
BKG	63.9	BKG	6	BKG	59
MDA	204	MDA	27	MDA	52
Guideline		Guideline		Guideline	
Action Level		Action Level		Action Level	

LOCATION: Parts from beam port

PURPOSE:

Location	Direct cpm	Removable dpm	
#	β	H-3	Total β
1	200	540.8	570.7
2	130	23.81	9.3
3	100	165.8	110.7
4	1450	55.8	9.3

4 is inside stainless steel plug

APPENDIX A-5

EXTERNAL REACTOR SYSTEMS SURVEYS AND SAMPLES

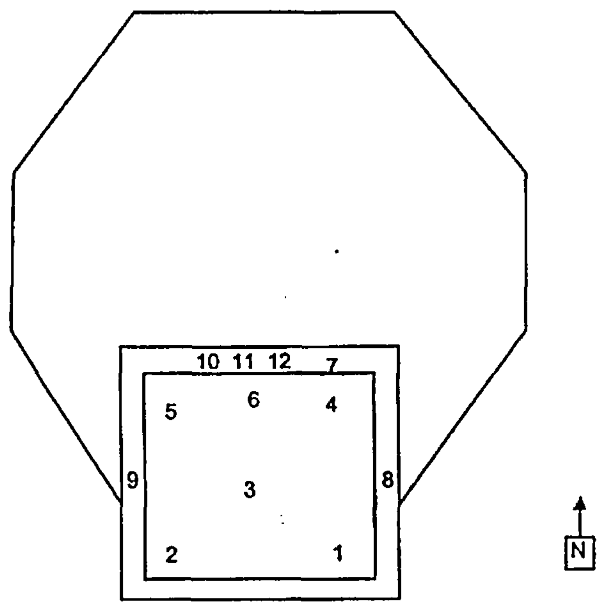
Scientech, LLC

Project #: 23566 **Date:** 7/14/2005
Site Name: University of Illinois **Location:** Bulk Shielding Tank
Instrument ID: Beckman-LS6500
MDA(a): H-3 32 C-14 29 dpm Total β 29 dpm
MDA(b): H-3 ____ C-14 ____ dpm Total β ____ dpm

Systematic Locations	H-3 (dpm)	Reportable H-3	C-14 (dpm)	Reportable C-14	Net total β (dpm)	Reportabe Total β	Survey Technician
REMOVABLE CONTAMINATION MEASUREMENTS							
1	139.55	139.6	382.36	382.4	673.3	673.3	Taylor
2	116.71	116.7	384.91	384.9	698.7	698.7	Taylor
3	72.19	72.2	269.63	269.6	429.3	429.3	Taylor
4	193.20	193.2	328.27	328.3	605.3	605.3	Taylor
5	162.96	163.0	625.09	625.1	1196.0	1196.0	Taylor
6	206.76	206.8	463.66	463.7	849.3	849.3	Taylor
7	24.18	<MDA	18.57	<MDA	64.0	64.0	Taylor
8	13.00	<MDA	11.95	<MDA	25.3	<MDA	Taylor
9	22.91	<MDA	21.19	<MDA	34.7	34.7	Taylor
10	41.71	41.7	3.97	<MDA	12.0	<MDA	Taylor
11	26.53	<MDA	6.57	<MDA	-28.0	<MDA	Taylor
12	37.25	37.3	1.13	<MDA	48.0	48.0	Taylor

1 - 6: Floors
7 - 12: Walls
10, 11,12: Under pealing paint

Exposure Rate uR/hr	
1	360
2	320
3	800
4	3200
5	600
6	2400



Survey Cover Sheet

Surveyor: Chris Higgins Date: 7/26/2005

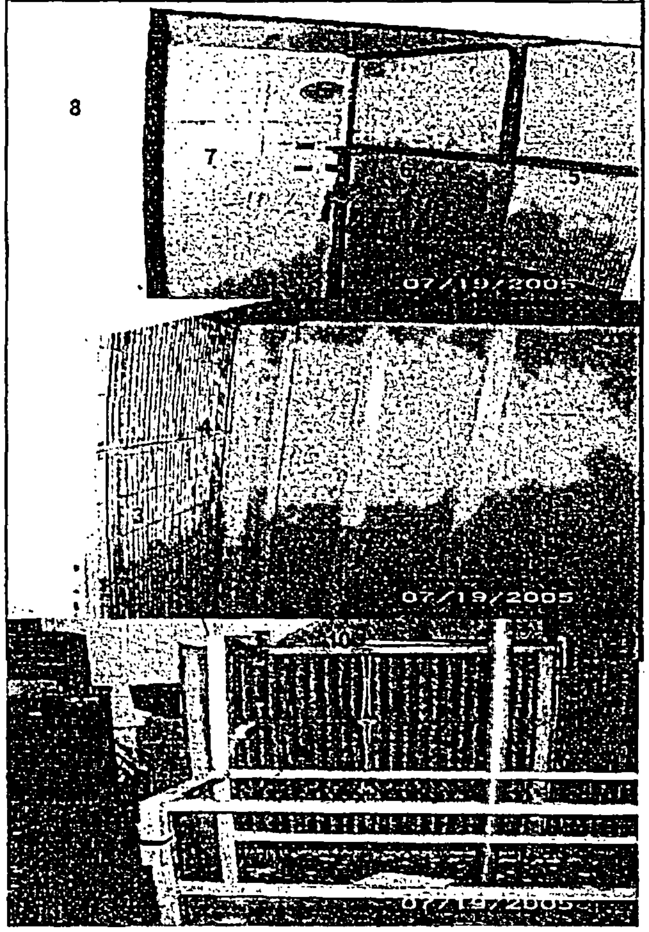
Reviewer: Kevin Taylor Date: 9/27/2005

Survey Meters

Meter Model #	2360	Meter Model #	2360	Meter Model #	Beckman 6500
Meter / Probe Serial #	202455	Meter / Probe Serial #	202455	Meter / Probe Serial #	
Cal Due	7/6/2006	Cal Due	7/6/2006	Cal Due	
Efficiency	14%	Efficiency	20%	Efficiency	75%
Type α, β γ	β γ	Type α, β γ	α	Type α, β γ	Total β
Sample Time	1	Sample Time	1	Sample Time	1
Background Time	10	Background Time	10	Background Time	1
BKG	329.1	BKG	1.7	BKG	59
MDA		MDA		MDA	52
Guideline		Guideline		Guideline	
Action Level		Action Level		Action Level	

LOCATION: Air Filter

PURPOSE:



Location #	Direct dpm/100cm2 α	Direct dpm/100cm2 β	Removable dpm H-3	Removable dpm Total β
1	6.5	692.1	-28.9	5.3
2	11.5	756.4	-14.7	14.7
3	11.5	1635.0	-6.1	-4.0
4	-8.5	2506.4	-16.5	-14.7
5	111.5	2985.0	-11.2	13.3
6	106.5	2985.0	-7.6	-12.0
7	1.5	-472.1	-5.4	2.7
8	6.5	-93.6	-13.7	-22.7
9	1.5	2570.7	-8.5	9.3
10	1.5	-36.4	-10.7	-7.9

Survey Cover Sheet

Surveyor: Chris Higgins — 7/14/2005 (Date)

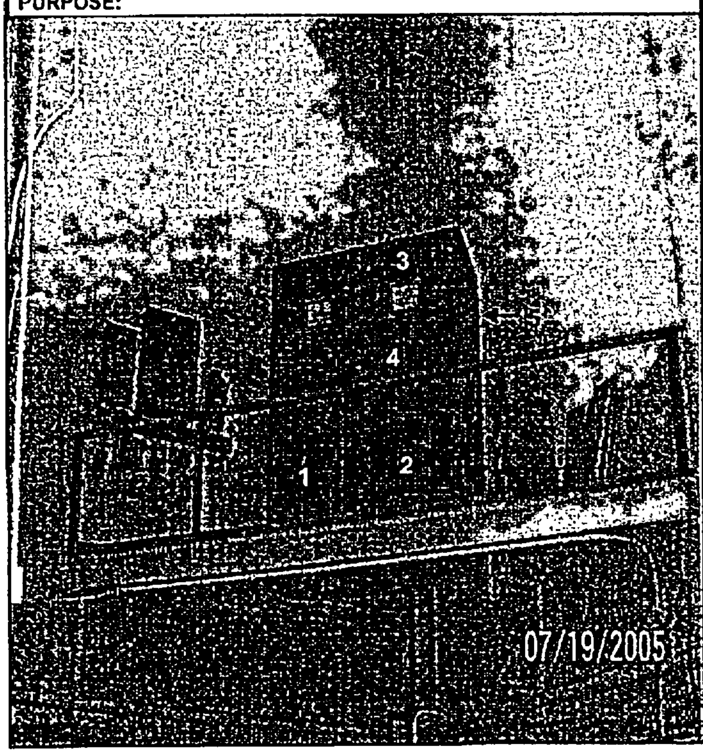
Reviewer: Kevin Taylor — 9/27/2005 (Date)

Survey Meters

Meter Model #	2360	Meter Model #	Beckman 6500	Meter Model #	Beckman 6500
Meter / Probe Serial #	202455	Meter / Probe Serial #		Meter / Probe Serial #	
Cal Due	7/6/2006	Cal Due		Cal Due	
Efficiency	13%	Efficiency	54%	Efficiency	75%
Type $\alpha, \beta\gamma$	$\beta\gamma$	Type $\alpha, \beta\gamma$	H-3	Type $\alpha, \beta\gamma$	Total β
Sample Time	1	Sample Time	1	Sample Time	1
Background Time	10	Background Time	1	Background Time	1
BKG	285.9	BKG	6	BKG	59
MDA	455.90	MDA	27	MDA	52
Guideline		Guideline		Guideline	
Action Level		Action Level		Action Level	

LOCATION: Air Conditioning Unit and Platform

PURPOSE:



Location #	Direct dpm/100 cm2 $\beta\gamma$	Removable dpm H-3	Removable dpm Total β
1	36416	-9.8	-1.3
2	36416	4.6	16.0
3	5647	-20.3	10.5
4	5647	-9.0	16
5	5647	-2.7	0

Survey Cover Sheet

Surveyor: Katie Streit — 7/19/2005 Date

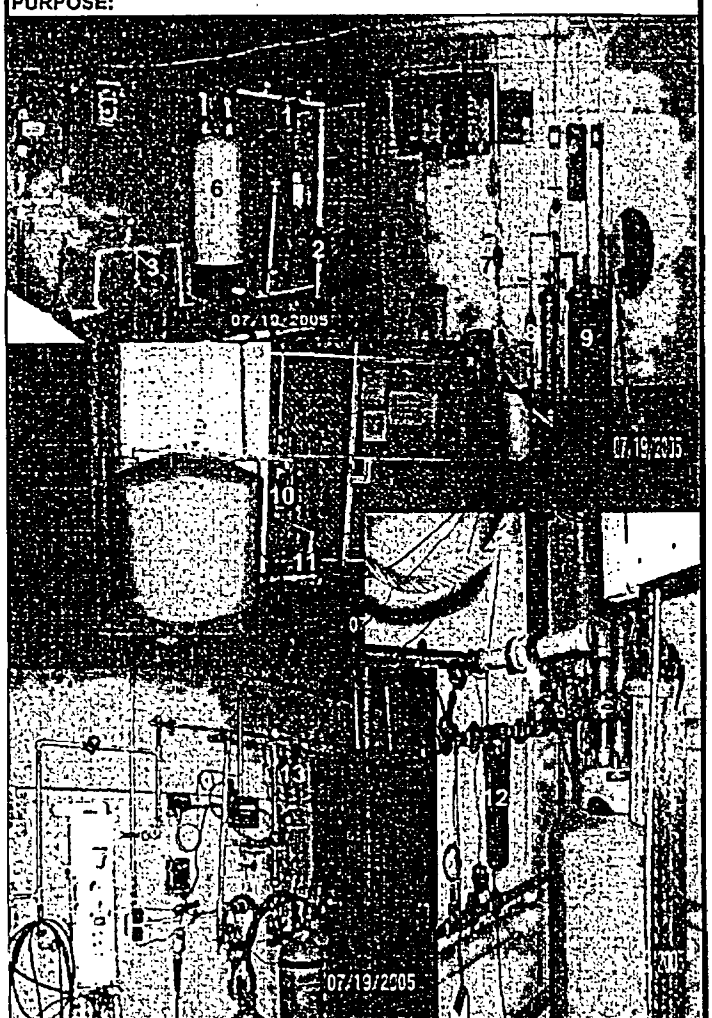
Reviewer: Kevin Taylor — 9/27/2005 Date

Survey Meters

Meter Model #	2220 / 44-10	Meter Model #		Meter Model #	
Meter / Probe Serial #	50067 / 230163	Meter / Probe Serial #		Meter / Probe Serial #	
Cal Due	6/29/2005	Cal Due		Cal Due	
Efficiency	NA	Efficiency		Efficiency	
Type α, β γ	γ	Type α, β γ		Type α, β γ	
Sample Time	1	Sample Time		Sample Time	
Background Time	1	Background Time		Background Time	
BKG	11008.7	BKG		BKG	
MDA	NA	MDA		MDA	
Guideline		Guideline		Guideline	
Action Level		Action Level		Action Level	

LOCATION: Water Filters

PURPOSE:



Location #	Direct cpm gross γ	Background	net γ
1	38605	37090	1515
2	59397	65694	-6297
3	29451	33291	-3840
5	43493	45042	-1549
6	52851	35224	17627
7	653884	654829	-945
8	641103	618948	22155
9	517781	539470	-21689
10	802413	810747	-8334
11	743032	766605	-23573
12	28368	36316	-7948
13	4777	5000	-223
14	4944	5000	-56
15	No data		

Survey Cover Sheet

Surveyor: Katie Streit Date: 7/21/2005

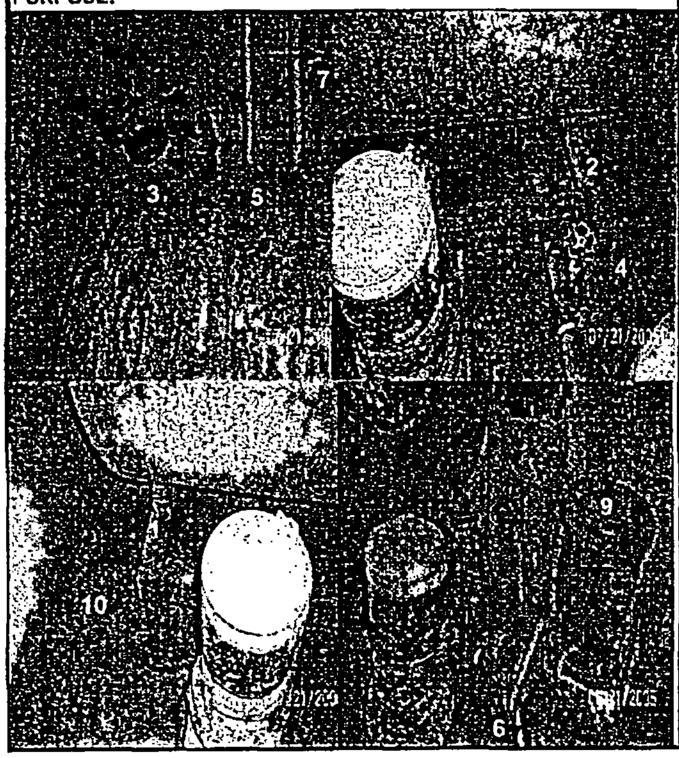
Reviewer: Kevin Taylor Date: 9/27/2005

Survey Meters

Meter Model #	2360		Meter Model #	2220	Meter Model #	Beckman 6500
Meter / Probe Serial #	202455		Meter / Probe Serial #	50067	Meter / Probe Serial #	
Cal Due	7/6/2006		Cal Due	6/29/2006	Cal Due	
Efficiency	14%	20%	Efficiency	NA	Efficiency	64%
Type α, $\beta\gamma$	$\beta\gamma$	α	Type α, $\beta\gamma$	γ	Type α, $\beta\gamma$	H-3
Sample Time	1	1	Sample Time	1	Sample Time	10
Background Time	10	10	Background Time	10	Background Time	10
BKG	307.7	1.1	BKG	12447	BKG	11.4
MDA	453.8	33.1	MDA	NA	MDA	10
Guideline			Guideline		Guideline	
Action Level			Action Level		Action Level	

LOCATION: 500 Gallon Tank

PURPOSE:



Location	Direct dpm/100cm2		cpm	Removable dpm	
#	$\beta\gamma$	α	γ	H-3	
1	-	-	7200	-	
2	-	-	11400	-	
3	-	-	12400	-	
4	-290.7	99.5		-0.45	
5	-	-	14000	-	
6	2435.7	49.5	13200	8.32	
7	-	-	13400	-	
8	-	-	11000	-2.76	
9	1900.0	134.5	10000	-	
10	1800.0	199.5	13200	22.68	

7 - Corner Wall

Tank Scan 200 - 250 Beta-gamma

Survey Cover Sheet

Surveyor: Katie Streit — Date: 7/26/2005

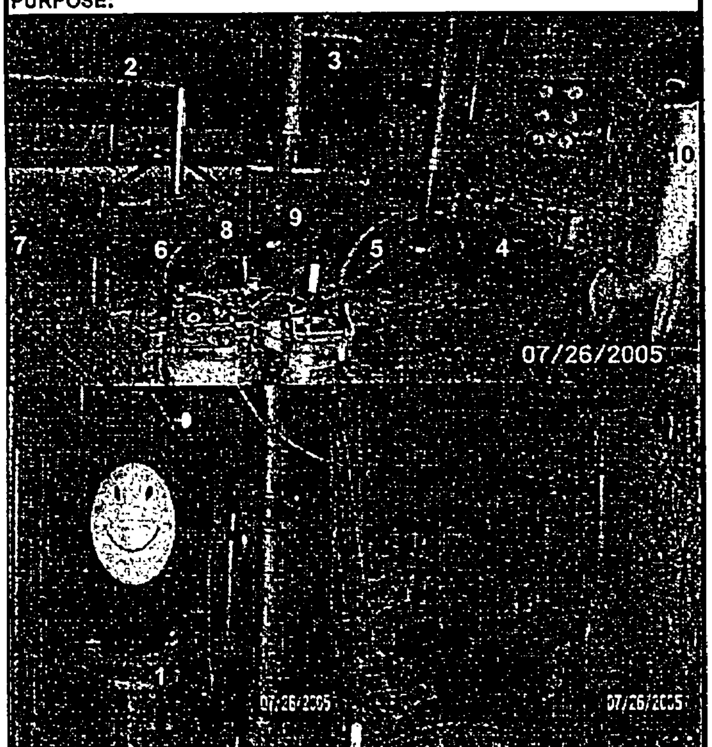
Reviewer: Kevin Taylor — Date: 9/27/2005

Survey Meters

Meter Model #	2221 / 44-10	Meter Model #		Meter Model #	
Meter / Probe Serial #	50357 / 230163	Meter / Probe Serial #		Meter / Probe Serial #	
Cal Due	6/26/2006	Cal Due		Cal Due	
Efficiency	NA	Efficiency		Efficiency	
Type $\alpha, \beta \gamma$	γ	Type $\alpha, \beta \gamma$		Type $\alpha, \beta \gamma$	
Sample Time	1	Sample Time		Sample Time	
Background Time	10	Background Time		Background Time	
BKG	12842.8	BKG		BKG	
MDA	NA	MDA		MDA	
Guideline		Guideline		Guideline	
Action Level		Action Level		Action Level	

LOCATION: Cooling Pumps

PURPOSE:



Location #	Direct cpm γ	Removable dpm H-3	Removable dpm Gross B
1	5181		
2	4265		
3	3581		
4	2907		
5	3979		
6	6499		
7	3658		
8	4160		
9	4911		
10	4256		
11	4102		

Survey Cover Sheet

Surveyor: Katie Streit — 7/20/2005 Date

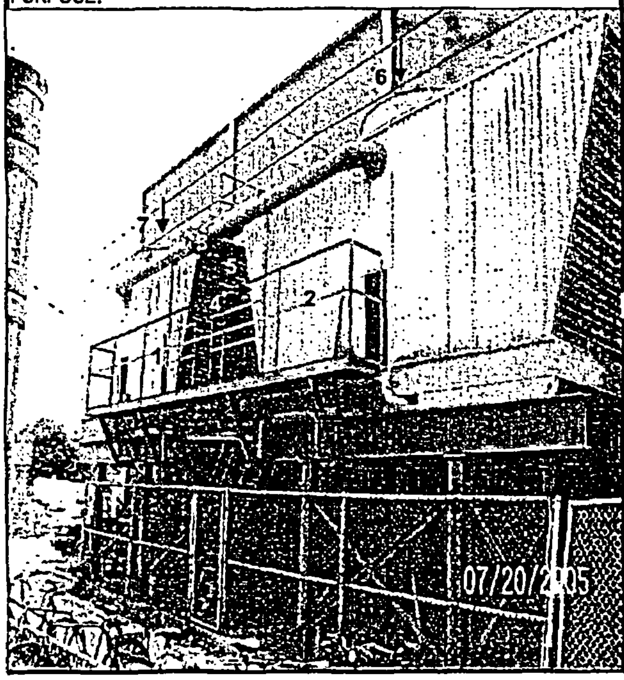
Reviewer: Kevin Taylor — 9/27/2005 Date

Survey Meters

Meter Model #	2360	Meter Model #	2360	Meter Model #	Beckman 6500		
Meter / Probe Serial #	202455	Meter / Probe Serial #	202455	Meter / Probe Serial #			
Cal Due	7/6/2006	Cal Due	7/6/2006	Cal Due			
Efficiency	14%	Efficiency	20%	Efficiency	64%	75%	
Type α, βγ	βγ	Type α, βγ	α	Type	H-3	Total β	
Sample Time	1	Sample Time	1	Sample Time	1	1	
Background Time	10	Background Time	10	Background Time	1	1	
BKG	295.9	BKG	1.1	BKG	6	59	
MDA	445.4	MDA	33.1	MDA	27	52	
Guideline		Guideline		Guideline			
Action Level		Action Level		Action Level			

LOCATION: Cooling Towers

PURPOSE:

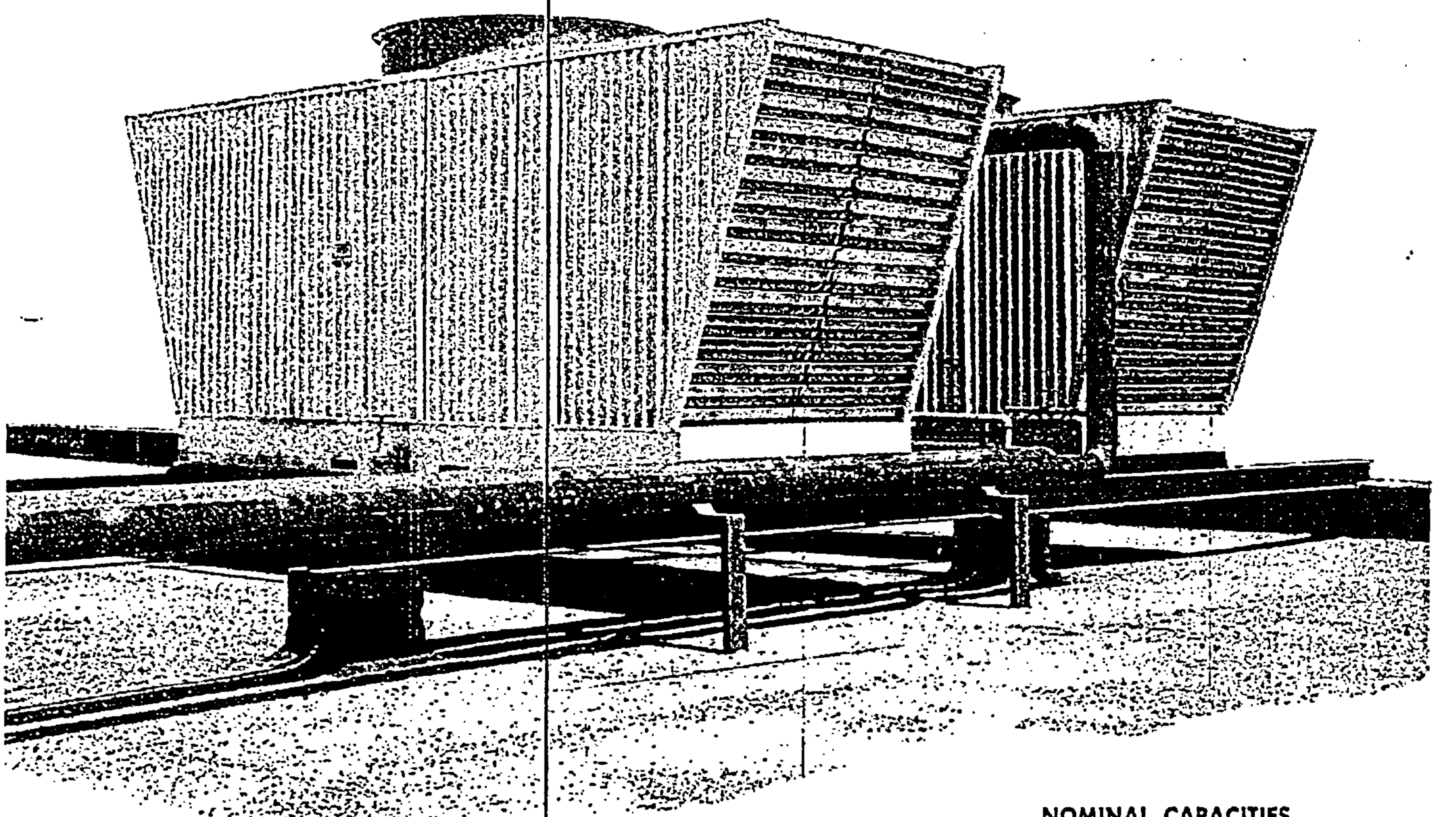


Location #	Direct dpm/100cm2 βγ	Direct dpm/100cm2 α	Removable dpm H-3	Removable dpm Total β
1	-314	34.5	43.48	22.7
2	-478	14.5	49.46	33.3
3	-614	49.5	123.16	93.3
4	422	-5.5	568.1	693.3
5	244	9.5	112.65	120.0
6	186	124.5	46.19	45.3
7	758	174.5	42.39	37.3
8	-749	-0.5	71.95	80.0
9	-642	19.5	42.39	18.7

Note:
The alpha measuremnts for points 6 and 7 are from a natural phenomina resulting from a static charge built up on a non-galvanized screw in contact with galvanized sheet metal. The static charge can appear as elevated alpha counts on the detector.

MARLEY

SERIES HCT™ STEEL Double-Flow®



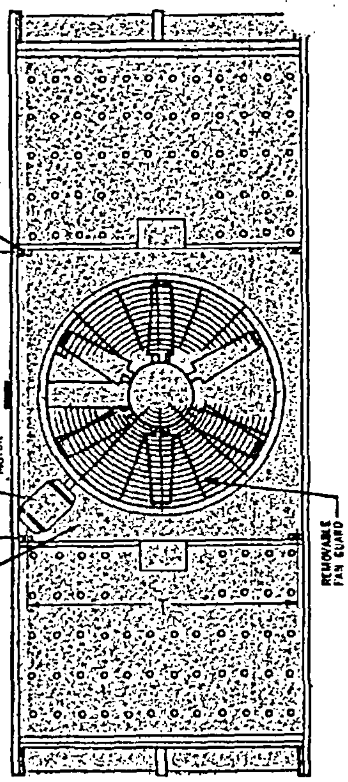
HOT DIP GALVANIZED AFTER FABRICATION

NOMINAL CAPACITIES
100 to 600 Tons

the Marley company/222 w. gregory/kansas city/mo.

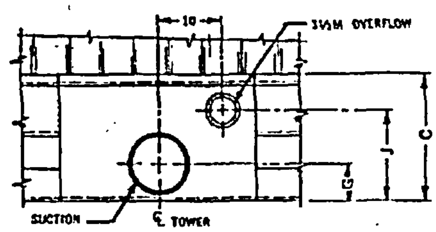
65B

SERIES HCT STEEL DOUBLE-FLOW®
ENGINEERING DATA

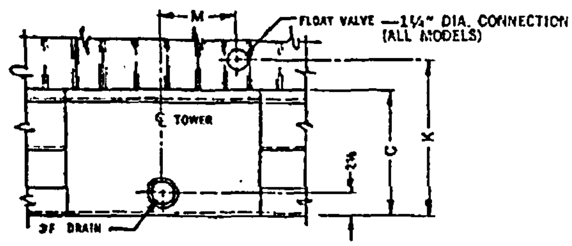


Top View Single-Cell Models

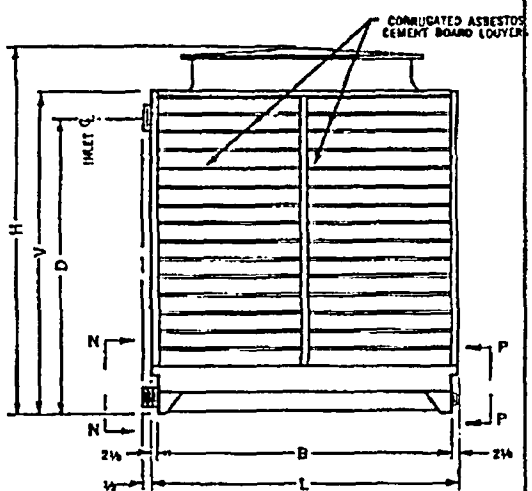
PIPING DETAILS



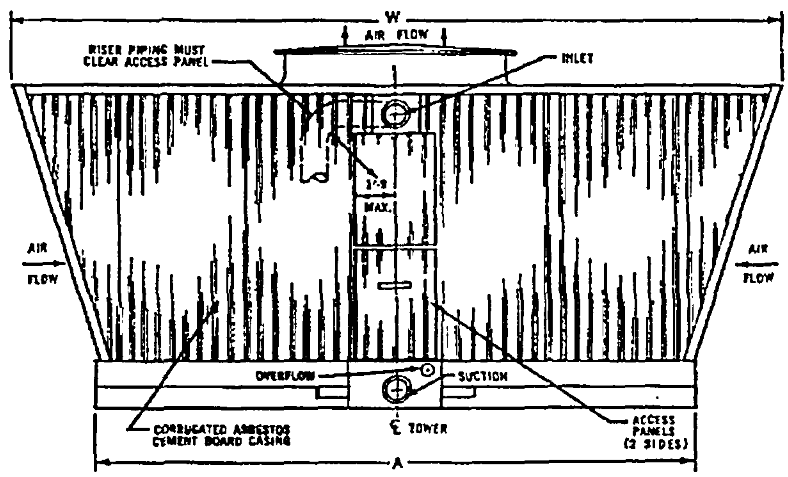
View N-N



View P-P
SUPPLY LINE TO FLOAT VALVE SHOULD BE ADEQUATE TO HANDLE MAKE-UP WATER AT 25 PSI MINIMUM.



End View Single-Cell Models

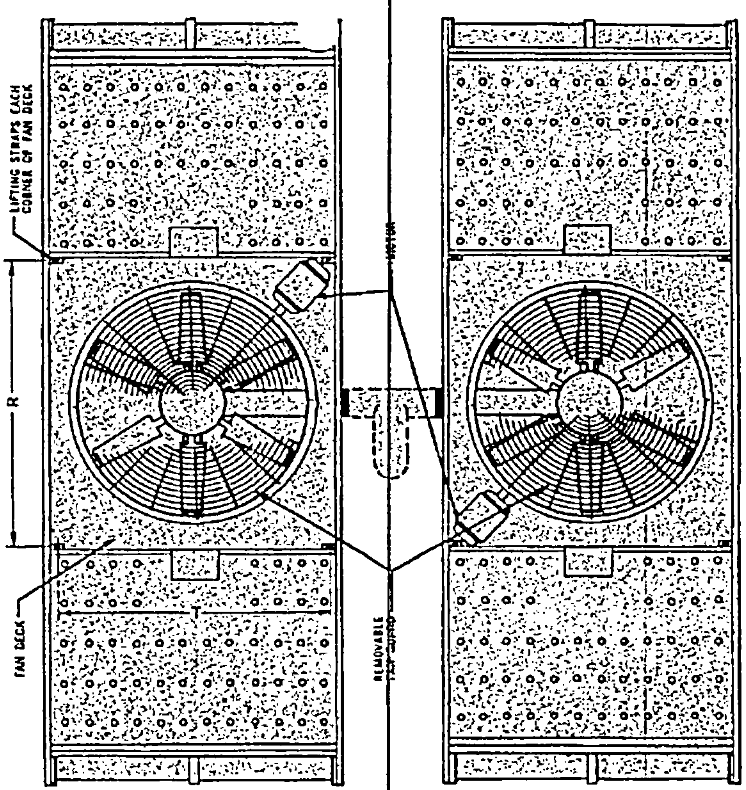


Side View (from Inlet and Suction Face)—Typical All Models

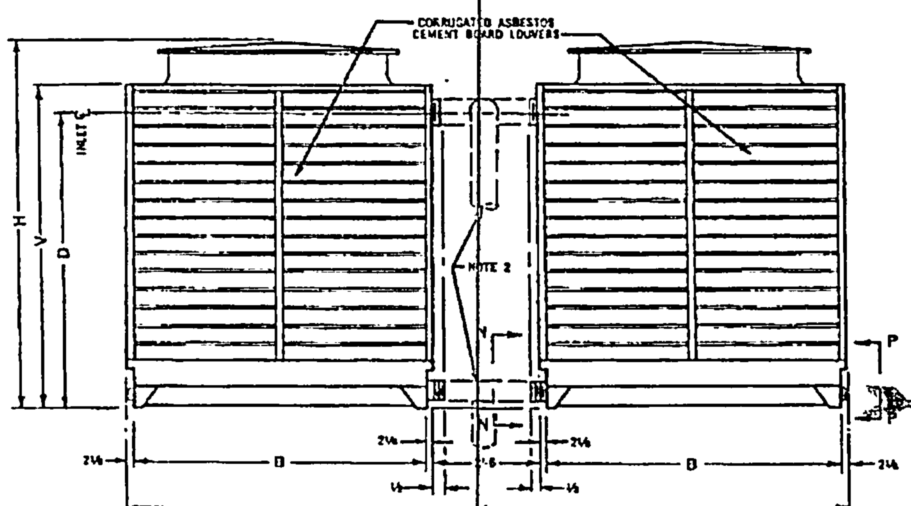
Dimensional Data—Models 8441 through 8457

MODEL NO.	TOWER OVERALL† L	TOWER OVERALL† W	TOWER OVERALL† H	A	B	C	D	V	FAN DIA.	PIPING DATA INLET	PIPING DATA SUCTION	PIPING DATA FLOAT VALVE	PIPING DATA G	PIPING DATA J	PIPING DATA K	PIPING DATA M	MOTOR H.P. Each Cell	OPER.‡ WT. (Lbs.)	SHIP WT. (Lbs.)	PUMP HEAD
8441	6'-4½	16'-8	8'-7	12'-11	6'-0¼	1'-2	6'-3¾	7'-1⅛	54	6m	6m	1¼	4⅜	0'-11⅝	1'-7	1'2¾	5	6500	4500	8
8443	6'-4½	18'-5	8'-7	14'-9	6'-0¼	1'-2	6'-3¾	7'-1⅛	60	6m	6m	1¼	4⅜	0'-11⅝	1'-7	1'2¾	7½	7300	5100	8
8445	6'-4½	19'-1	10'-1	14'-9	6'-0¼	1'-4	7'-9¼	8'-6⅝	60	6m	8m	1¼	5½	1'-1⅛	1'-9	1'2¾	7½	8800	6200	10
8447	8'-4½	18'-2	10'-2	13'-8¼	8'-0¼	1'-5	7'-8¼	8'-8⅝	72	8m	8m	1¼	5½	1'-2⅛	1'-10	1'7	7½	9900	6800	[illegible]
8449	8'-4½	21'-2	10'-2	16'-8¼	8'-0¼	1'-5	7'-8¼	8'-8⅝	72	8m	8m	1¼	5½	1'-2⅛	1'-10	1'9	7½	11900	[illegible]	10
8451	8'-4½	22'-2	11'-2	17'-0¼	8'-0¼	1'-6	8'-8¼	9'-8⅝	72	8m	8m	1¼	5½	1'-3¼	1'-11	1'2¾	7½	13900	[illegible]00	11
8453	8'-4½	23'-2	11'-7	18'-0¼	8'-0¼	1'-6	8'-7¼	9'-8⅝	84	10m	10m	1¼	6½	1'-3¼	1'-11	1'2¾	10	14300	9700	11
8455	8'-4½	23'-2	11'-7	18'-0¼	8'-0¼	1'-6	8'-7¼	9'-8⅝	84	10m	10m	1¼	6½	1'-3¼	1'-11	1'2¾	15	14370	9750	11
8457	8'-4½	23'-2	11'-7	18'-0¼	8'-0¼	1'-6	8'-7¼	9'-8⅝	84	10m	10m	1¼	6½	1'-3¼	1'-11	1'2¾	20	14400	9780	11

†Installed dimensions. ‡Based on circulating maximum GPM.



Top View Two-Cell Models



End View Two-Cell Models

General Notes

TOWERS

Note 1. All piping external to tower to be furnished by customer.

Note 2. Standard Marley piping terminates with threaded connection immediately outside endwall casing. On two-cell towers, a fabricated pipe tee is available at additional cost. Inlets on these tees are same pipe size as tower inlet and suction fittings. Tees are fabricated to permit welded connection of supply and suction lines by customer. Pipe tee is black steel, prime coated.

SUPPORTING STEEL

See Page 4

Note 3. Minimum dimension for piping may be increased if more space is desired between towers. This dimension must be held if Marley provides pipe tee (optional) to connect towers.

Note 4. Supporting steel to be designed, furnished and installed by purchaser.

Note 5. Anchor bolt holes are ⅝-inch diameter. Use ½ x 1½ inch hot dip galvanized machine bolts for tower anchorage.

Note 6. To place beams at a location other than that shown, dimension "E" may be less (but not more) than indicated. Towers may be supported at corners of basin if desired. When towers are supported at points other than those shown, holes for anchor bolts must be field-drilled in basin channel.

Note 7. Towers may be supported on a concrete slab. Load per lineal foot under basin channel is equal to one-half the operating weight divided by the basin channel length (Dimension "A").

Note 8. Obtain certified prints for construction.

HOISTING DATA

MODEL NO.	R	T
8441 & 2-8441	6'-11	5'-3½
8443 & 2-8443	7'-5	5'-3½
8445 & 2-8445	7'-5	5'-3½
8447 & 2-8447	8'-5	7'-3½
8449 & 2-8449	8'-5	7'-3½
8451 & 2-8451	8'-5	7'-3½
8453 & 2-8453	9'-5	7'-3½
8455 & 2-8455	9'-5	7'-3½
8457 & 2-8457	9'-5	7'-3½

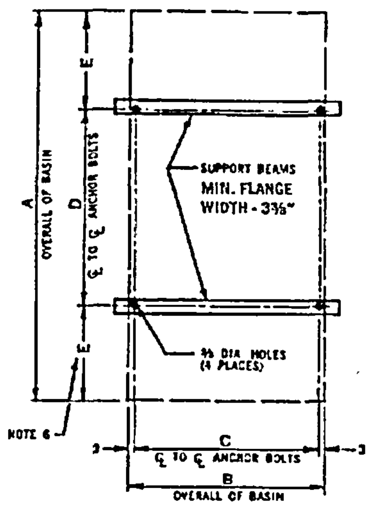
Dimensional Data—Models 2-8441 through 2-8457

MODEL NO.	TOWER OVERALL			A	B	C	D	V	FAN DIA.	PIPING DATA							MOTOR H.P. Each Cell	OPER. WT. (Lbs.)	SHIP. WT. (Lbs.)	PUMP HEAD
	L	W	H							INLET	SUCTION	*FLOAT VALVE	G	J	K	M				
2-8441	15'-3	16'-8	8'-7	12'-11	6'-0¼	1'-2	6'-3¾	7'-1⅛	54	6m	6m	1¼	4½	0'-11½	1'-2	1'-2¾	5	13000	9000	8'
2-8443	15'-3	18'-5	8'-7	14'-9	6'-0¼	1'-2	6'-3¾	7'-1⅛	60	6m	6m	1¼	4½	0'-11½	1'-7	1'-2¾	7½	14600	10200	8'
2-8445	15'-3	19'-1	10'-1	14'-9	6'-0¼	1'-4	7'-9¼	8'-6⅛	60	6m	8m	1¼	5½	1'-1¼	1'-9	1'-2¾	7½	17600	12400	10'
2-8447	19'-3	18'-2	10'-2	13'-8¼	8'-0¼	1'-5	7'-8¼	8'-8⅛	72	8m	8m	1¼	5½	1'-2¼	1'-10	1'7	7½	19800	13600	10'
2-8449	19'-3	21'-2	10'-2	16'-8¼	8'-0¼	1'-5	7'-8¼	8'-8⅛	72	8m	8m	1¼	5½	1'-2¼	1'-10	1'9	7½	23800	15000	10'
2-8451	19'-3	22'-2	11'-2	17'-0¼	8'-0¼	1'-6	8'-8¼	9'-8⅛	72	8m	8m	1¼	5½	1'-3¼	1'-11	1'2¾	7½	27800	18800	11'
2-8453	19'-3	23'-2	11'-7	18'-0¼	8'-0¼	1'-6	8'-7¼	9'-8⅛	84	10m	10m	2½	6½	1'-3¼	1'-11	1'2¾	10	28600	19400	11'
2-8455	19'-3	23'-2	11'-7	18'-0¼	8'-0¼	1'-6	8'-7¼	9'-8⅛	84	10m	10m	2½	6½	1'-3¼	1'-11	1'2¾	15	28740	19500	11'
2-8457	19'-3	21'-2	11'-7	18'-0¼	8'-0¼	1'-6	8'-7¼	9'-8⅛	84	10m	10m	2½	6½	1'-3¼	1'-11	1'2¾	20	28860	19560	11'

*Minimum dimension; may be increased to provide more space between towers. †Installed dimensions. ‡Based on circulating maximum GPM.

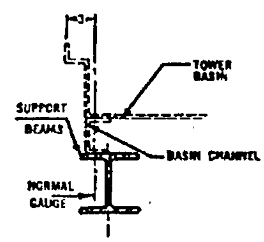
*One float valve supplied with two-cell models.

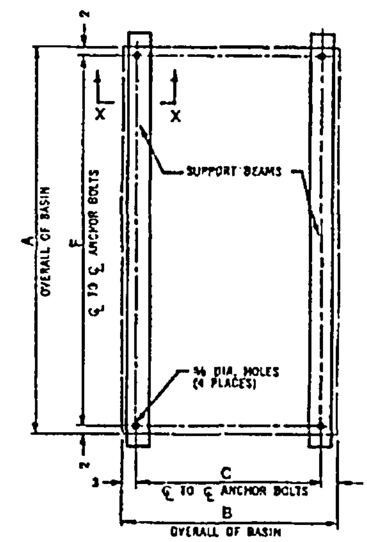
..JFPORTING STEEL ARRANGEMENTS



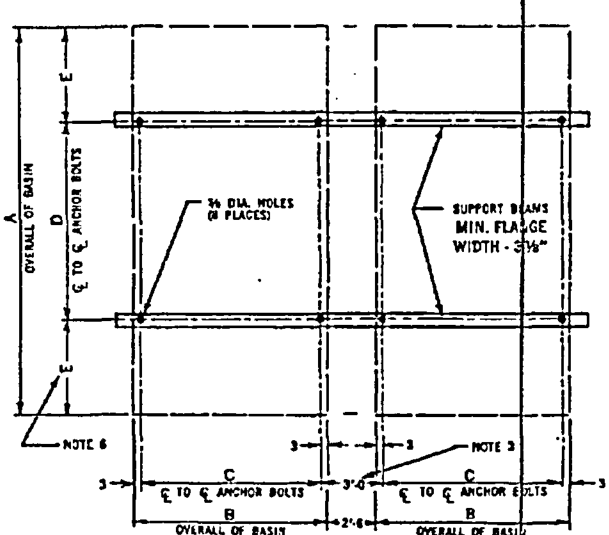
Plan 1—One-Cell

Section X-X

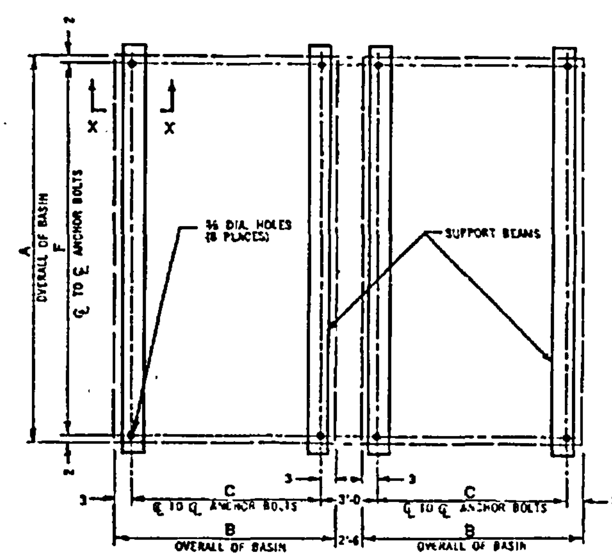




Plan 2—One-Cell



Plan 3—Two-Cell



Plan 4—Two-Cell

Dimensional Data—Supporting Steel

MODEL NO.	A	B	C	D	E	F	DEAD LOAD* (Lbs.)	MAX. LIVE* LOAD (Lbs.)
8441 & 2-8441	12'-11	6'-4½	5'-10½	6'-7	3'-2	12'-7	1625	1100
8443 & 2-8443	14'-9	6'-4½	5'-10½	7'-5	3'-8	14'-5	1825	1220
8445 & 2-8445	14'-9	6'-4½	5'-10½	7'-5	3'-8	14'-5	2200	1800
8447 & 2-8447	13'-8¼	8'-4½	7'-10½	6'-10½	3'-5	13'-4¼	2475	1300
8449 & 2-8449	16'-8¼	8'-4½	7'-10½	8'-4¼	4'-2	16'-4¼	2975	1540
8451 & 2-8451	17'-0¼	8'-4½	7'-10½	8'-6¼	4'-3	16'-8¼	3475	2050
8453 & 2-8453	18'-0¼	8'-4½	7'-10½	9'-2¼	4'-5	17'-8¼	3575	2180
8455 & 2-8455	18'-0¼	8'-4½	7'-10½	9'-2¼	4'-5	17'-8¼	3500	2180
8457 & 2-8457	18'-0¼	8'-4½	7'-10½	9'-2¼	4'-5	17'-8¼	3600	2180

*At any anchorage point.

Series HCT STEEL DOUBLE-FLOW Capacity Chart

All Ratings Shown Are Gallons Per Minute (GPM)

HOT WATER	90°	90°	90°	95°	95°	95°	95°	97°	96°	96°	97°	103°	103°	105°	106°
COLD WATER	80°	80°	80°	85°	85°	85°	85°	87°	86°	86°	87°	85°	85°	85°	86°
WET BULB	65°	70°	72°	72°	75°	76°	78°	78°	79°	80°	80°	75°	78°	78°	80°
MODEL 8441	441	340	290	450	380	355	300	370	308	280	317	255	206	193	182
MODEL 8443	548	420	360	560	475	442	375	465	385	350	395	320	260	243	230
MODEL 8445	655	510	435	665	570	530	450	555	462	418	472	390	320	293	275
MODEL 8447	765	590	510	730	665	620	525	650	537	488	550	450	365	342	322
MODEL 8449	876	675	580	885	760	708	600	745	615	558	628	520	415	390	370
MODEL 8451	975	760	655	1000	855	795	675	835	690	628	705	590	470	440	415
MODEL 8453	1090	840	730	1190	945	880	750	925	770	700	785	645	525	492	465
MODEL 8455	1195	925	800	1210	1035	957	825	1010	845	772	865	710	575	545	512
MODEL 8457	1290	1005	870	1315	1125	1055	900	1095	922	843	940	805	635	600	565

SPECIFICATIONS

75.7°

STRUCTURE: The Series HCT structure is designed to withstand wind loads of 30 psf. It consists of modular welded sub-assemblies cross-braced with tension rods. Unwelded structural joints are double-bolted.

CASING: Corrugated asbestos cement board casing is applied with vertical corrugations. Casing joints are lapped.

LOUVERS: Removable, slip-fit, corrugated asbestos cement board air inlet louvers are standard. Close louver spacing improves appearance.

DISTRIBUTION BASIN: A one-piece, welded steel basin assembly is installed over each filling section. Basins are fitted with polyethylene metering orifices located on close centers to give uniform water distribution over fill. Velocity breaker hood located at hot water basin inlet assures equal distribution to all metering orifices.

FAN DECK: One-piece fan deck is fabricated of heavy-gauge steel reinforced with stiffening bars. Flanges on all four sides of the fan deck and welding of distributor tee into fan deck give this assembly additional strength and rigidity.

FILLING: Pressure-treated wood splash bars are supported on close centers by non-corrosive grids of exclusive Marley design, which assures permanent fill alignment and configuration. Patented Marley diffusion decks located immediately below distribution basins provide initial water break-up and assure uniform water distribution over filling.

DRIFT ELIMINATORS: Two-pass neoprene coated cellular honeycomb drift eliminators control moisture carry over.

COLLECTION BASIN: Dual-level basin consists of self-cleaning sections under filling areas and deeper depressed center sump section. One-piece basin of heavy-gauge steel is designed to support the tower when resting on two grillage beams, four corner posts or a concerte slab. Basin is equipped with non-vortexing suction connection with removable screen, oversized drain-cleanout to make depressed section cleanout easy, overflow and a direct action Marley float valve.

MECHANICAL EQUIPMENT:

Fan: Marley multi-blade, cast aluminum type H-3 fans are standard equipment on all Marley Series HCT towers. Blade pitch is pre-set at the factory.

Gear Drive: All models are equipped with Marley Geareducer drive and Marley driveshaft equipped with non-lubricated flexible couplings.

Motors: Single-speed, protected type motor is mounted on welded angle and bar frame in one corner of fan deck.

Mechanical Equipment Support: Hot water distributor tee welded into the fan deck also serves as support for Geareducer and fan.

Fan Cylinder: One-piece, welded steel assembly is welded to fan deck. Full height, eased inlet cylinder assures maximum fan efficiency. Complete assembly is hot dip galvanized.

Fan Guard: Welded, one-piece guard, consisting of radial spokes and perimeter ring of 5/16" rod and concentric rings of 7-gauge wire is hot dip galvanized after fabrication.

ACCESS: Large access doors are provided on both sides of the tower for easy access to center air plenum. In this open area are located the suction screen, float valve, fan, Geareducer and depressed portion of the basin. Easy maintenance and cleanability are design features.

FINISH: All steel is hot dip galvanized after fabrication.

OPTIONAL EQUIPMENT

The following equipment is available for Series HCT towers at nominal cost:

Optional Casing and Louver Material—corrugated metal or plastic may be specified.

Steel Ladder—18-inches wide, constructed of ¼ x 2 bar rails with ⅞" diameter rod rungs welded to side rails. Ladder is field-bolted to the tower.

Steel Handrail—fabricated of steel angle—field-bolted to tower.

Special Motors—two-speed motors and motors with TEFC enclosures are available.

Hot Water Basin Covers—removable basin covers of heavy gauge steel may be provided.

Air Inlet Screens—#2 mesh with 16-gauge metal U-edging. Screens are attached to tower outside of louvers with self-tapping screws.

Outside Oil Fill and Drain Lines—a fill standpipe and dipstick assembly is located on fan deck outside of fan cylinder. A pipe terminating in a capped fitting extends downward to a position immediately above access panels for oil draining purposes.

TWO CELL TOWERS

Two single-cell towers may be joined to operate as one tower. Welded tee assemblies terminating in single connections are available for joining inlets and suctions. Standard weight steel pipe with matched flanged connections and gaskets are provided. Prime painted tee connections are fabricated for field welding to external piping.

TOWERS FOR INSTALLATION OVER BASIN BY CUSTOMER

One or two-cell towers may be field assembled on concrete or steel basins provided by the customer. Towers are shipped unassembled without the cold water basin. Special anchor clips are provided to connect tower to basin provided by customer. Basin design drawings are furnished by Marley.

FREIGHT CLASSIFICATION

Towers, water cooling, with or without blowers or fans; sheet steel, thicker than 20 gauge or steel and wood. Class 100.

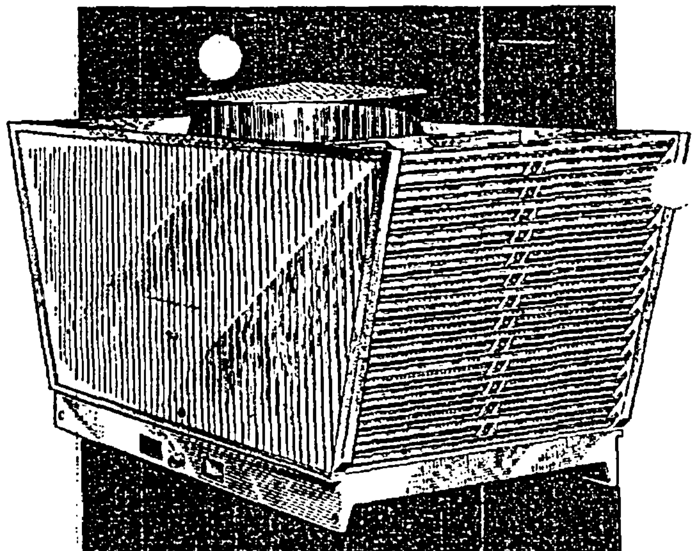
MARLEY

SERIES HCT STEEL Double-Flow®

FLEXIBLE — Series HCT Steel Double-Flows are available for quick delivery either completely assembled, partially assembled for easy field erection or "less basin" for installation over customer's basin.

DURABLE — Liberal utilization of non-corrosive, inert materials, hot dip galvanizing of all steel components and asbestos cement board casing and louvers give these units long durability and service.

CAPABLE — Series HCT Steel Double-Flows are High Capacity units. Marley's advanced crossflow design produces maximum water cooling per cubic foot of tower volume.



NON-COMBUSTIBLE FILL (Alternate)

HCT Steel Double-Flows are available with alclad protected surface aluminum splash bars. HCT Double-Flows with this type of fill are designated "non-combustible" by FM and FIA.

SERIES HCT DESIGN

Speeds Installation — Increases Performance — Lengthens Service Life

Structural Simplicity

Unique structural design facilitates speedy field assembly when required. Tower structure consists of 8 basic sub-assemblies which can be assembled in a few hours.

Permanent Fill Alignment

Marley's exclusive fill support grids position splash surfaces permanently. Water break-up, air-water contact and flow pattern through the towers do not vary from design level. Grid support system with treated wood splash bars has proved its durability in thousands of applications.

Heavy Duty Mechanical Equipment

Service proved Marley Geareducer drive is standard equipment on all Series HCT Steel Double-Flows. Heavy-duty, multi-blade cast aluminum fans, designed and made by Marley for rugged cooling tower duty, give longer service life. Protected type motors are factory mounted outside the fan cylinder. Full height, cased inlet fan cylinder is hot dip galvanized after fabrication for lasting durability.

Self-Cleaning Basin

One-piece, hot dip galvanized steel basin is self-cleaning. Sludge is washed continuously to the accessible depressed center section of the basin. All cleaning is performed in the unobstructed plenum area through the oversized cleanout fitting. Minimum water level is maintained in the basin under the fill. The depressed center section holds adequate volume of water for High Capacity tower operation and gives minimum operating weight. Suction and overflow fittings are located on one side of the basin—cleanout-drain and float valve on the opposite side.

Other Series HCT Features

ASBESTOS CEMENT BOARD CASING AND LOUVERS — require no maintenance and blend harmoniously with all surroundings.

NEOPRENE HONEYCOMB DRIFT ELIMINATORS — Neoprene coated, phenolic impregnated fiber honeycomb is impervious to all corrosive elements—designed for long life. Drift is kept to a minimum with low draft loss—more cooling performance per Hp.

INTERNAL PIPING—Distribution tee carries water from hot water inlet to distribution basins.

UNITIZED MECHANICAL EQUIPMENT SUPPORT—Geareducer and fan are mounted on distribution tee which is welded into fan deck forming a strong, rigid unitized assembly. The motor is supported on a unitized weldment in one corner of the fan deck.

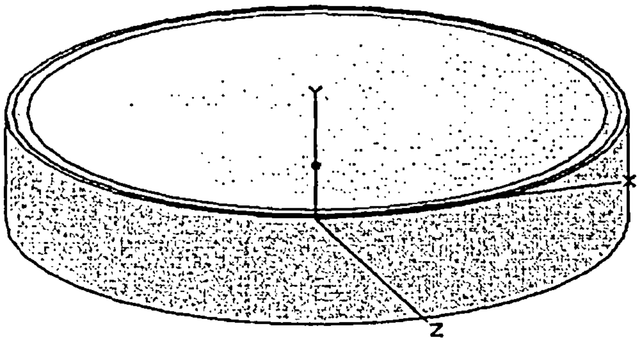
HOISTING AND HANDLING — Hoisting lugs are located at each corner of the fan deck to facilitate rigging and hoisting of factory assembled units and for easy field assembly when units are shipped unassembled.

MicroShield v6.02 (6.02-00111)
SCIENTECH_Inc.

Page : 1
DOS File : Lazy Susan Center.ms6
Run Date: September 28, 2005
Run Time: 3:22:50 PM
Duration : 00:00:01

File Ref: 23566
Date: 9/28/05
By: K. Taylor
Checked: ______

Case Title: Case 1
Description: Case 1
Geometry: 11 - Annular Cylinder - Internal Dose Point



Source Dimensions

Height	10.16 cm	4.0 in
Inner Cyl Radius	25.4 cm	10.0 in
Inner Cyl Thickness	1.27 cm	0.5 in
Source	1.0 cm	0.4 in

Dose Points

	X	Y	Z
# 1	0 cm	5.08 cm	0 cm
	0.0 in	2.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Cyl. Radius	25.4 in	Water	1
Shield 1	.5 in	Aluminum	2.6
Source	66.759 in³	Aluminum	2.6

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Co-60	4.0500e+000	1.4985e+011	3.7021e+003	1.3698e+008

Buildup
The material reference is : Source

Integration Parameters

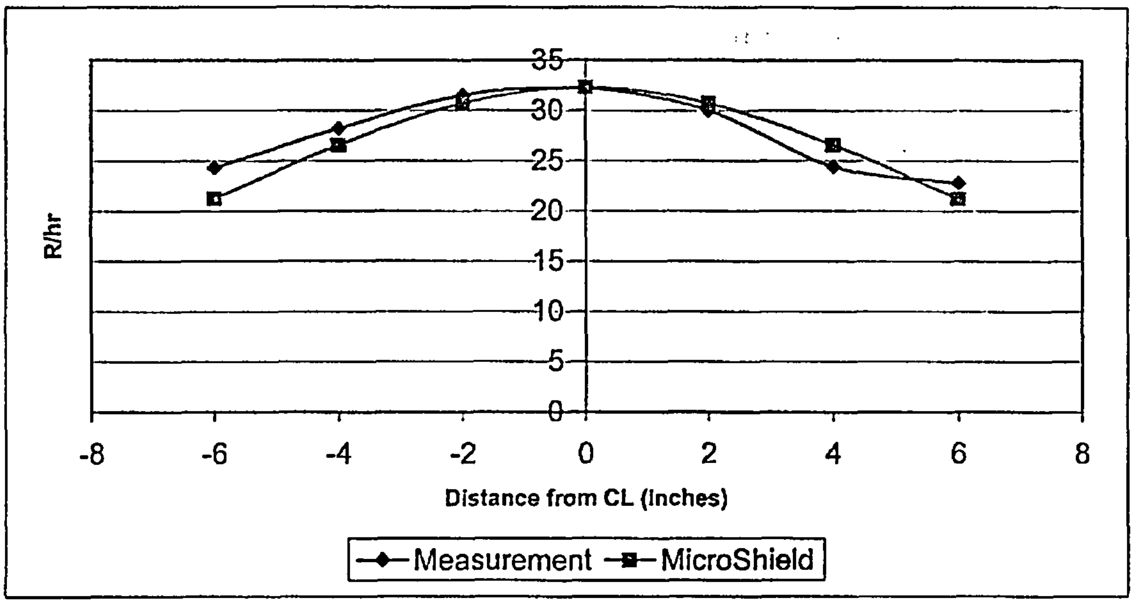
Radial	10
Circumferential	10
Y Direction (axial)	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	2.444e+07	1.591e+02	7.086e+02	3.071e-01	1.368e+00
1.1732	1.499e+11	2.837e+06	8.447e+06	5.071e+03	1.510e+04
1.3325	1.499e+11	3.632e+06	9.919e+06	6.301e+03	1.721e+04
TOTALS:	2.997e+11	6.470e+06	1.837e+07	1.137e+04	3.231e+04

Comparison of Underwater Lazy Susan Dose Measurements to Modeled MicroShield Dose Rates

Height (inches)	Underwater Measurement (R/hr)	MicroShield Dose Rate (R/hr)
-6	24.3	21.23
-4	28.2	26.56
-2	31.5	30.7
0	32.2	32.31
2	30	30.7
4	24.4	26.56
6	22.8	21.23



APPENDIX A-6

OTHER EQUIPMENT SURVEYS AND SAMPLES

Survey Cover Sheet

Surveyor: Katie Streit — 7/19/2005 Date

Reviewer: Kevin Taylor — 9/28/2005 Date

Survey Meters

Meter Model #	2350	Meter Model #	2360	Meter Model #	Beckman 6500	
Meter / Probe Serial #	202455	Meter / Probe Serial #	202455	Meter / Probe Serial #		
Cal Due	7/6/2005	Cal Due	7/6/2005	Cal Due		
Efficiency	14%	Efficiency	20%	Efficiency	53%	75%
Type α, βγ	βγ	Type α, βγ	α	Type α, βγ	H-3	Total β
Sample Time	1	Sample Time	1	Sample Time	1	1
Background Time	10	Background Time	10	Background Time	1	1
BKG	321	BKG	2.2	BKG	14	54
MDA	463.0160203	MDA	40.59019441	MDA	39	50
Guideline		Guideline		Guideline		
Action Level		Action Level		Action Level		

LOCATION: Thermo Shield, tract, and graphite cart

PURPOSE:

Location #	Direct dpm/100cm2 βγ	Direct dpm/100cm2 α	Removable dpm H-3	Removable dpm Gross β
1	664.3		69.1	1.3
2	135.7		20.5	4.0
3	42.9		22.5	32.0
4	378.6		78.6	21.3
5	8935.7		22.0	-6.7
6	11778.6		-2.7	-17.3
7	11042.9		12.2	-13.3
8	11007.1		26.0	8.0
9	5078.6		21.0	-8.0
10	1878.6		3.0	8.0
11	2078.6		28.6	8.0
12	307.1		92.7	66.7
13	7428.6		9.2	6.7
14	7728.6		35.0	4.0
15	457.1		17.4	0.0
16	-578.6		7.9	13.3
17	-		12.3	16.0
18	6321.4		-	-

Survey Cover Sheet

Surveyor: Katie Streit Date: 7/20/2005

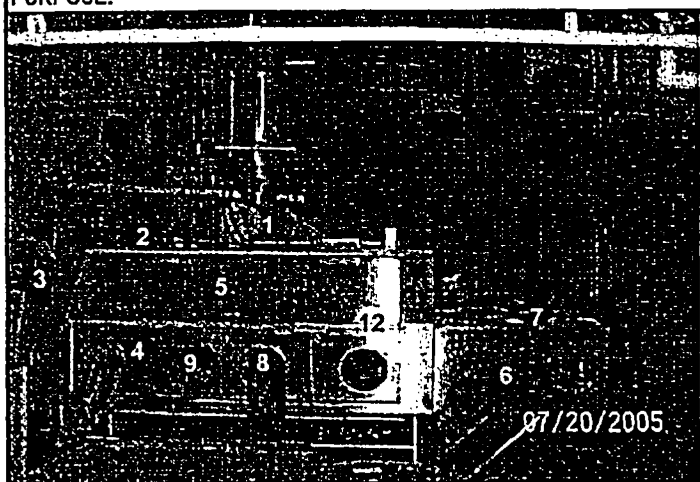
Reviewer: Kevin Taylor Date: 9/28/2005

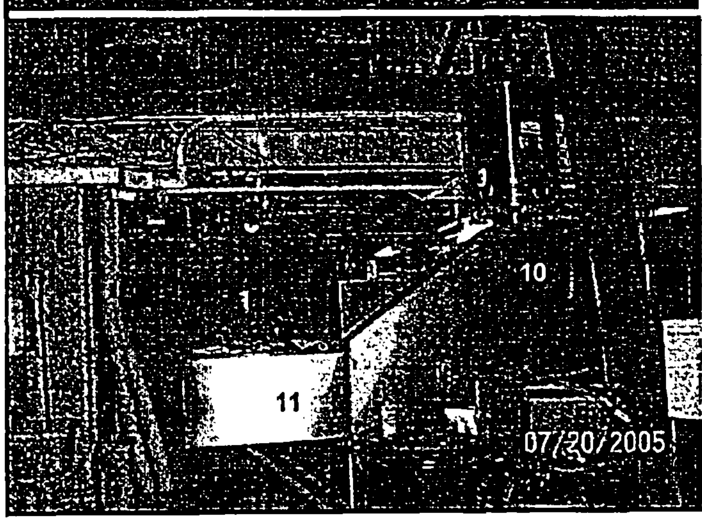
Survey Meters

Meter Model #	2360	2360	2929	2929
Meter / Probe Serial #	202455	202455	167852	167852
Cal Due	7/6/2006	7/6/2006	6/22/2006	6/22/2006
Efficiency	14%	20%	21%	38%
Type α, βγ	βγ	α	βγ	α
Sample Time	1	1	1	1
Background Time	10	10	10	10
BKG	295.9	1.1	70	0.5
MDA	445.4	33.1	151.8	14.3
Guideline				
Action Level				

LOCATION: Glove Box

PURPOSE:





Location #	Direct dpm/100cm2 βγ	Direct dpm/100cm2 α
1	4629.3	39.5
2	5407.9	104.5
3	9336.4	119.5
4	4950.7	19.5
5	5[illegible]93.6	129.5
6	25[illegible]93.6	319.5
7	8150.7	74.5
8	18465.0	39.5
9	12257.9	574.5
10	4229.3	74.5
11	11815.0	69.5
12	20793.6	74.5
inside		
1	-	-
2	-	-
3	-	-
4	-	-
5	-	-
6	-	-

Removable dpm/100 cm2 Inside	α	β
1	571.1	985.7
2	555.3	995.2
3	1786.8	3533.3
4	1544.7	2719.0
5	2260.5	3966.7
6	3171.1	4481.0

Direct beta-gamma measurments rejected due to high background.

Survey Cover Sheet

Surveyor: Katie Streit — Date: 7/18/2005

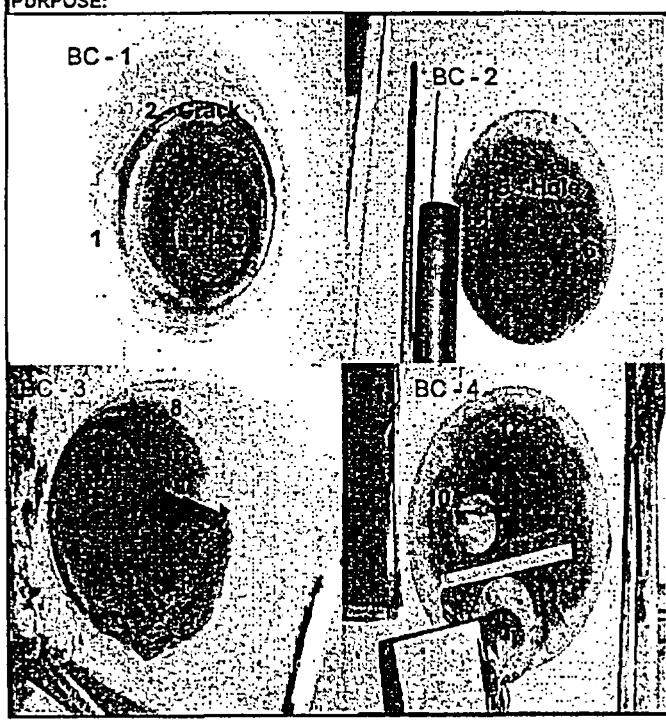
Reviewer: Kevin Taylor — Date: 9/28/2005

Survey Meters

Meter Model #	2360	Meter Model #	2360	Meter Model #	Beckman 6500	
Meter / Probe Serial #	202455	Meter / Probe Serial #	202455	Meter / Probe Serial #		
Cal Due	7/6/2006	Cal Due	7/6/2006	Cal Due		
Efficiency	14%	Efficiency	20%	Efficiency	54%	75%
Type α, βγ	βγ	Type α, βγ	α	Type α, βγ	H-3	Total β
Sample Time	1	Sample Time	1	Sample Time	1	1
Background Time	10	Background Time	10	Background Time	1	1
BKG	313.4	BKG	2.2	BKG	16	63
MDA	457.6	MDA	40.6	MDA	40	53
Guideline		Guideline		Guideline		
Action Level		Action Level		Action Level		

LOCATION: Beam Catchers

PURPOSE:



Location #	Direct dpm/100cm2 βγ	Direct dpm/100cm2 α	Removable dpm H-3	Removable dpm Gross beta
1	1275.7	5	-17.97	40
2	2268.6	5	21.88	21.33
3	9547.1	15	156.52	117.33
4	1118.6	15	57.58	57.33
5	875.7	10	425.18	289.33
6	775.7	40	229.99	109.33
7	618.6	20	53.77	18.67
8	190.0	0	30.78	16.00
9	147.1	20	100.76	85.33
10	13118.6	10	31.56	-9.33
11	11161.4	35	15.34	-20.00
12	9404.3	20	8.74	-17.33

10 - 12: High direct bata-gamma because of the presence of radioactive materials in the beam catcher.

BC-1 Across from southeastern beam port
BC-2 Across from western beam port
BC-3 Across from northwestern beam port
BC-4 Across from northen beam port

Survey Cover Sheet

Surveyor: Chris Higgins — 7/14/2005 Date

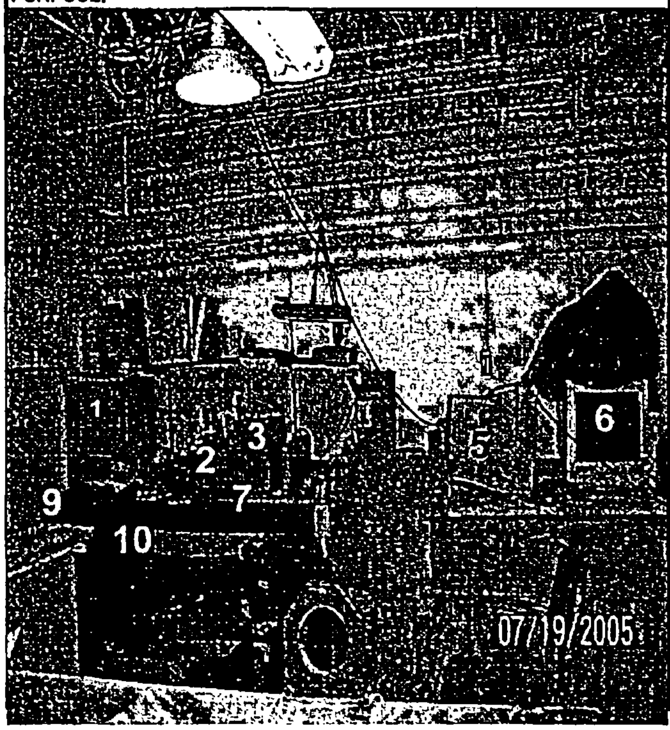
Reviewer: Kevin Taylor — 7/28/2005 Date

Survey Meters

Meter Model #	2360	Meter Model #	2360	Meter Model #	Beckman 6500	
Meter / Probe Serial #	202455/200135	Meter / Probe Serial #	202455/200135	Meter / Probe Serial #		
Cal Due	7/6/2006	Cal Due	7/6/2006	Cal Due		
Efficiency	13%	Efficiency	20%	Efficiency	54%	75%
Type α, βγ	βγ	Type α, βγ	α	Type α, βγ	H-3	Total β
Sample Time	1	Sample Time	1	Sample Time	1	1
Background Time	10	Background Time	10	Background Time	1	1
BKG	265.9	BKG	1.5	BKG	11	64
MDA	455.9	MDA	36.1	MDA	34	54
Guideline		Guideline		Guideline		
Action Level		Action Level		Action Level		

LOCATION: Bench NorthWest Side of Reactor Floor

PURPOSE:



Location #	Direct dpm/100cm2 H-3	Direct dpm/100cm2 α	Removable dpm H-3	Removable dpm Gross β
1	-	-	40.74	-9.33
2	-	-	38.74	42.67
3	-	-	23.22	21.33
4	362.3	7.5	11.15	-6.67
5	-	-	11.15	0.00
6	-	-	17.57	-1.33
7	3077.7	17.5	26.19	12.00
8	54.6	12.5	5.62	18.67
9	-	-	40.06	25.33
10	-	-	45.67	2.67

Survey Cover Sheet

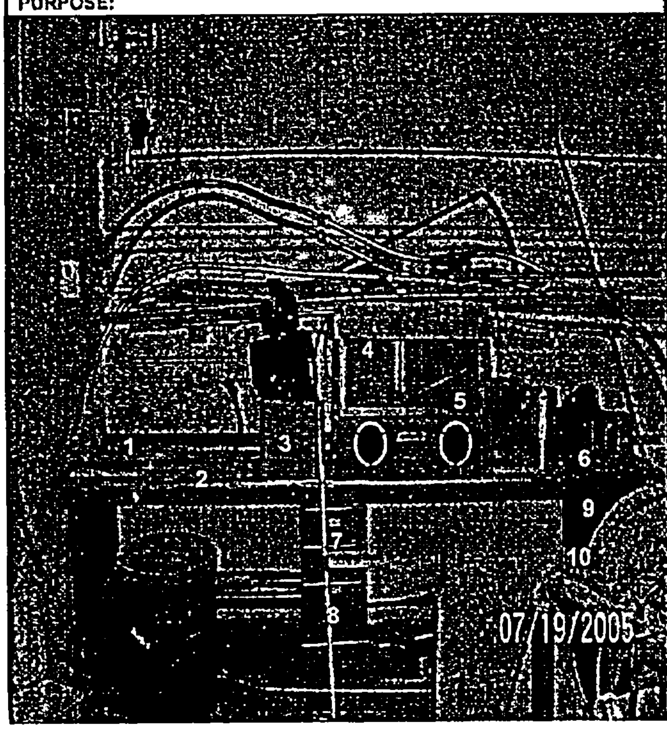
Surveyor:	Chris Higgins	7/14/2005	Date
Reviewer:	Kevin Taylor	7/28/2005	Date

Survey Meters

	Meter 1	Meter 2	Meter 3	
Meter Model #	2360	2360	Beckman 6500	
Meter / Probe Serial #	202455/200135	202455/200135		
Cal Due	7/6/2006	7/6/2006		
Efficiency	13%	20%	64%	75%
Type α, βγ	βγ	α	H-3	Total β
Sample Time	1	1	1	1
Background Time	10	10	1	1
BKG	265.9	1.5	11	64
MDA	455.9	36.1	34	54
Guideline				
Action Level				

LOCATION: Bench South East Rector Floor

PURPOSE:



Location #	Direct dpm/100 cm2 βγ	Direct dpm/100 cm2 α	Removable dpm H-3	Removable dpm Gross β
1	12777.7	7.5	0.62	18.67
2	37500.8	7.5	19.73	10.67
3	57777.7	32.5	61.21	8.00
4	5923.85	7.5	19.73	618.67
5	4077.69	32.5	-32.61	49.33
6	4431.54	17.5	34.33	53.33
7	-	-	13.94	201.33
8	-	-	12.48	28.00
9	-	-	51.35	32.00
10	-	-	12.78	13.33

1 - 6 High background area due to contaminated materials stored on the work bench.

Survey Cover Sheet

Surveyor: Chris Higgins — 7/18/2005 (Date)

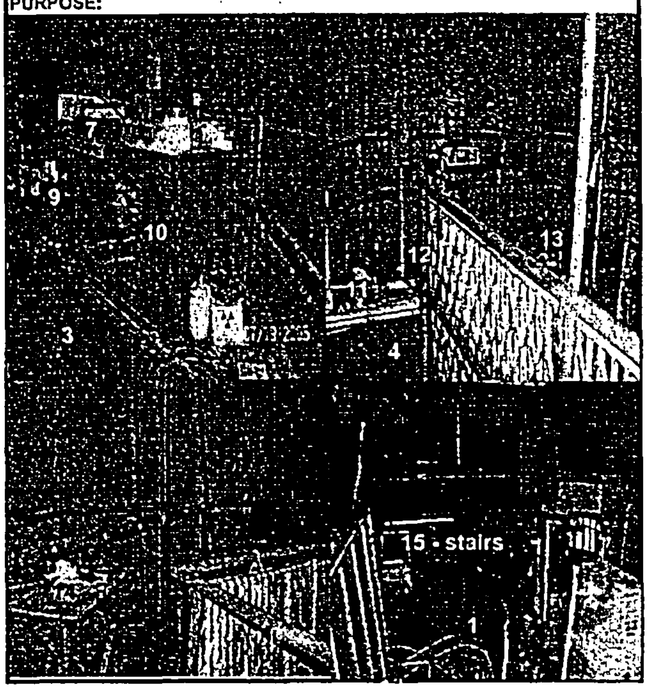
Reviewer: Kevin Taylor — 9/28/2005 (Date)

Survey Meters

	Meter 1	Meter 2	Meter 3	
Meter Model #	2360	2360	Beckman 6500	
Meter / Probe Serial #	202455/200135	202455/200135		
Cal Due	7/6/2006	7/6/2006		
Efficiency	13%	20%	54%	75%
Type (α, $\beta\gamma$)	$\beta\gamma$	α	H-3	Total β
Sample Time	1	1	1	1
Background Time	10	10	1	1
BKG	265.9	1.5	11	64
MDA	455.9	36.1	34	54
Guideline				
Action Level				

LOCATION: Reactor Top

PURPOSE:



Location #	Direct dpm/100cm2 $\beta\gamma$	Direct dpm/100cm2 α	Removable dpm H-3	Removable dpm Gross beta
1	1093.1	14	35.69	17.33
2	1946.9	19	25.15	24.00
3	16577.7	4	28.11	-1.33
4	3370.0	4	57.45	16.00
5	1846.9	9	8.87	30.67
6	-	-	10.77	8.00
7	-	-	22.84	24.00
8	-	-	20.7	97.33
9	-	-	58.57	1606.67
10	-	-	-6.3	58.67
11	-	-	20.25	37.33
12	-	-	27.73	-4.00
13	-	-	26.81	33.33
14	-	-	13.7	20.00
15	-	-	33.84	8.00
16			-36.94	14949.33

16 - under the steel plate on the bridge

APPENDIX A-7

SUMMARY OF TRITIUM ANALYSIS IN SOIL SAMPLES

Tritium in Subsurface Soil Analysis Results
University of Illinois Nuclear Research Laboratory
Sampling Date: July 25, 2005

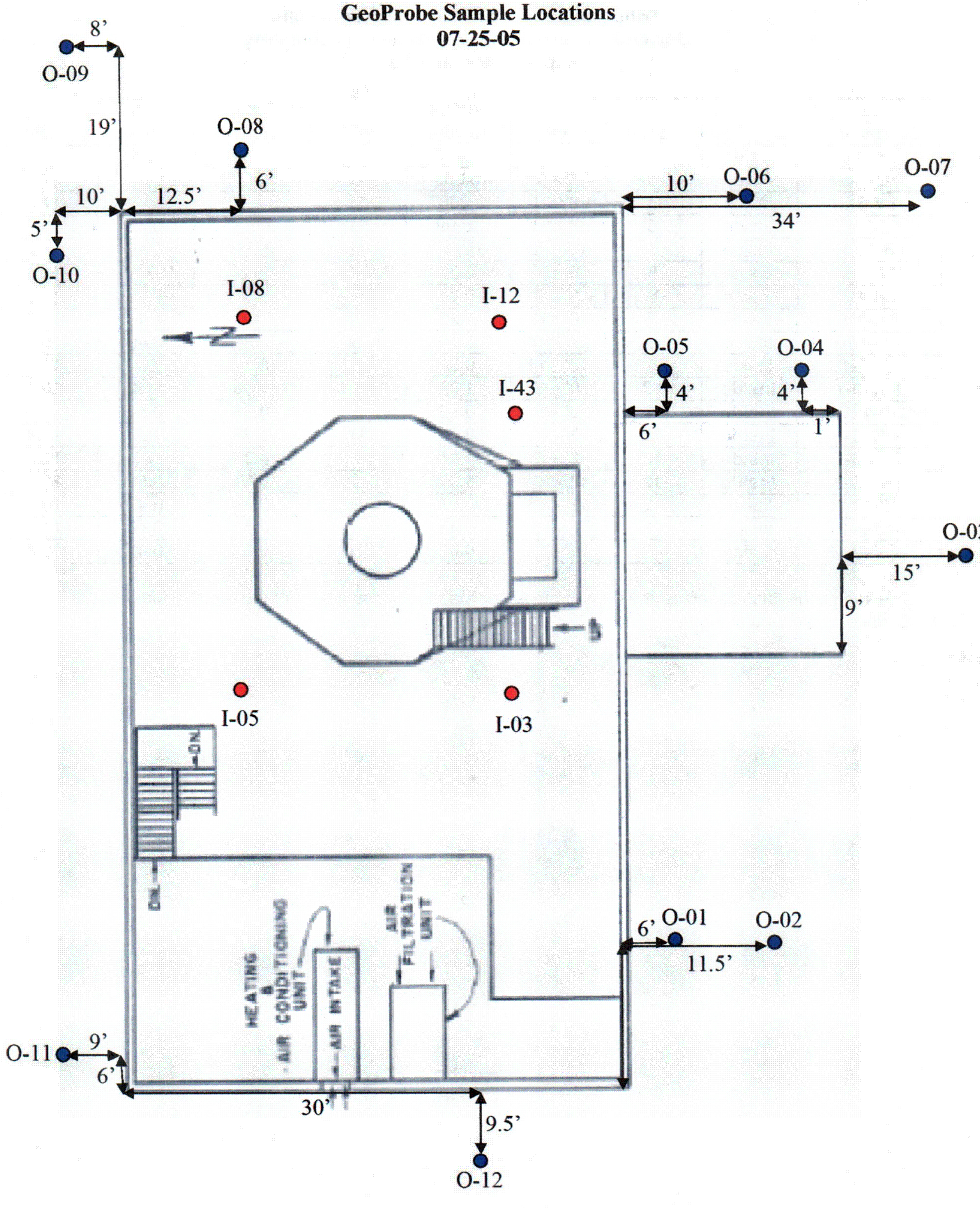
Sample ID #	Location	Loc. #	Depth[a]	Result (pCi/g)	Qualifier	+/- 2 sigma (pCi/g)	MDC (pCi/g)	% Moisture
O-1-4-5	Outside	1	4' - 5'	-0.14	U	0.69	0.5	10.4
O-1-4-5 (Dup)	Outside	1	4' - 5'	-0.2	U	3.5	0.5	-
O-1-8-9	Outside	1	8' - 9'	0.00043	U	0.00052	0.48	12.3
O-1-12-13	Outside	1	12' - 13'	0.081	U	0.071	0.47	10.9
O-2-4-5	Outside	2	4' - 5'	-0.13	U	0.28	0.45	3.0
O-2-8-9	Outside	2	8' - 9'	0.01	U	0.013	0.47	3.7
O-2-12-13	Outside	2	12' - 13'	-0.2	U	8.2	0.5	10.0
O-3-4-5	Outside	3	4' - 5'	0.2	U	0.13	0.51	21.1
O-3-8-9	Outside	3	8' - 9'	0.13	U	0.12	0.51	16.1
O-3-12-13	Outside	3	12' - 13'	-0.3	U	3.1	0.5	11.6
O-4-4-5	Outside	4	4' - 5'	0.07	U	0.66	0.56	16.7
O-4-8-9	Outside	4	8' - 9'	-0.12	U	0.3	0.47	8.0
O-4-12-13	Outside	4	12' - 13'	0.078	U	0.065	0.47	10.4
O-5-8-9	Outside	5	8' - 9'	-0.16	U	0.3	0.5	16.0
O-5-12-13	Outside	5	12' - 13'	-0.05	U	0.41	0.49	13.5
O-6-4-5	Outside	6	4' - 5'	-0.22	U	0.32	0.57	24.7
O-6-8-9	Outside	6	8' - 9'	-0.33	U	0.28	0.55	14.8
O-6-12-13	Outside	6	12' - 13'	-0.27	U	0.31	0.51	12.9
O-7-4-5	Outside	7	4' - 5'	-0.27	U	0.27	0.51	5.6
O-7-8-9	Outside	7	8' - 9'	-0.04	U	0.29	0.49	3.7
O-7-12-13	Outside	7	12' - 13'	-0.01	U	0.13	0.54	13.8
O-8-4-5	Outside	8	4' - 5'	0.13	U	0.31	0.51	21.0
O-8-8-9	Outside	8	8' - 9'	0.04	U	0.24	0.5	19.8
O-8-12-13	Outside	8	12' - 13'	0.12	U	0.28	0.48	11.3
O-9-4-5	Outside	9	4' - 5'	-0.2	U	0.38	0.53	20.8
O-9-4-5 (Dup)	Outside	9	4' - 5'	-0.27	U	0.32	0.54	-
O-9-8-9	Outside	9	8' - 9'	-0.13	U	0.28	0.55	16.9
O-9-12-13	Outside	9	12' - 13'	0.06	U	0.27	0.54	19.4
O-10-4-5	Outside	10	4' - 5'	0.04	U	0.35	0.52	21.2
O-10-8-9	Outside	10	8' - 9'	0.31	U	0.31	0.51	19.4
O-10-12-13	Outside	10	12' - 13'	0.02	U	0.19	0.5	20.0
O-11-4-5	Outside	11	4' - 5'	-0.21	U	0.31	0.57	21.9
O-11-8-9	Outside	11	8' - 9'	-0.34	U	0.38	0.48	12.6
O-11-12-13	Outside	11	12' - 13'	-0.12	U	0.29	0.52	12.2
O-12-4-5	Outside	12	4' - 5'	0.02	U	0.19	0.51	10.6
O-12-8-9	Outside	12	8' - 9'	-0.14	U	0.31	0.53	13.5
O-12-12-13	Outside	12	12' - 13'	-0.26	U	0.27	0.52	10.8
B-01[b]	Outside	B-01	0' - 1'	-0.18	U	0.32	0.55	18.9
B-02[b]	Outside	B-02	0' - 1'	-0.35	U	0.32	0.56	22.6
B-03[b]	Outside	B-03	0' - 1'	-0.24	U	0.26	0.54	14.5
B-04[b]	Outside	B-04	0' - 1'	-0.04	U	0.31	0.58	26.6
I-3-0-1	**Inside**	**3**	**0' - 1'**	**0.61**	**J**	0.23	0.47	15.3
I-3-3-4	**Inside**	**3**	**3' - 4'**	**0.56**	**J**	0.23	0.49	13.6
I-3-6-7	Inside	3	6' - 7'	0.34	U	0.19	0.5	16.4
I-3-12-15	Inside	3	12' - 15'	-0.3	U	4.4	0.5	10.1

Tritium in Subsurface Soil Analysis Results
University of Illinois Nuclear Research Laboratory
Sampling Date: July 25, 2005

Sample ID #	Location	Loc. #	Depth[a]	Result (pCi/g)	Qualifier	+/- 2 sigma (pCi/g)	MDC (pCi/g)	% Moisture
I-5-0-1	Inside	5	0' - 1'	0.079	U	0.088	0.5	10.3
I-5-3-4	Inside	5	3' - 4'	0.003	U	0.0033	0.48	12.0
I-5-6-7	Inside	5	6' - 7'	0.0087	U	0.0088	0.49	17.5
I-5-9-10	Inside	5	9' - 10'	-0.09	U	0.14	0.46	10.0
I-5-12-13	Inside	5	12' - 13'	0.117	U	0.084	0.46	10.7
I-8-0-1	Inside	8	0' - 1'	0.32	U	0.3	0.5	18.6
I-8-4-5	Inside	8	3' - 4'	0.24	U	0.31	0.49	15.1
I-8-6-7	Inside	8	6' - 7'	0.22	U	0.3	0.48	11.8
I-8-9-10	Inside	8	9' - 10'	0.19	U	0.29	0.49	15.8
I-8-12-13	Inside	8	12' - 13'	0.38	U	0.32	0.5	15.2
I-12-0-1	**Inside**	**12**	**0' - 1'**	**1.05**		**0.38**	**0.52**	**20.1**
I-12-0-1 (Dup)	**Inside**	**12**	**0' - 1'**	**0.79**	**J**	**0.35**	**0.51**	**-**
I-12-3-4	**Inside**	**12**	**3' - 4'**	**0.25**	**U**	**0.29**	**0.5**	**19.8**
I-12-6-7	**Inside**	**12**	**6' - 7'**	**1.07**		**0.37**	**0.5**	**17.2**
I-12-8-9	**Inside**	**12**	**9' - 10'**	**0.99**	**J**	**0.37**	**0.52**	**21.7**
I-12-12-13	Inside	12	12' - 13'	0.07	U	0.24	0.47	10.2
I-43-9-10	Inside	43	9' - 10'	0.41	U	0.31	0.48	11.3
I-43-12-13	Inside	43	12' - 13'	0.07	U	0.33	0.49	10.6

a: The sample depth is based on the depth from either the reactor room floor of the ground surface.
b: Surface backgrlund samples.

GeoProbe Sample Locations
07-25-05



C01

APPRNDIX B

DATA SUMMARY TABLES AND GAMMA SPECTROSCOPY REPORTS (On CD)

TABLE B-1
SUMMARY OF ON-SITE CONCRETE AND SOIL LSC ANALYSIS

Sample Number	Media	Gross Cont Rate (cpm)[a]		LSC	Net Count Rates (cpm)[b]		Net Activity (pCi/g)[b]	
		H-3	Beta	Det #	H-3	Beta	H-3	Beta[c]
NMNT- 01	Concrete	212, 215	9210, 5950	1, 2	Background sample			
NMNT- 02	Concrete	186, 184	12068, 8199	1, 2	Background sample			
NMNT- 03	Concrete	346	78362	1	180	69032	2.28E+04	9.94E+05
NMNT- 04	Soil	224	9321	1	<MDA	<MDA	<MDA	<MDA
NMNT- 05	Concrete	321	97678	1	155	88348	1.97E+04	1.24E+06
NMNT- 06	Concrete	373	61927	2	191	53638	1.26E+04	4.08E+05
NMNT- 07	Concrete	340	89237	1	174	79907	2.21E+04	1.13E+06
NMNT- 08	Concrete	415	73132	2	233	64843	1.54E+04	4.82E+05
NMNT- 09	Concrete	309	61639	1	143	52309	1.81E+04	7.82E+05
NMNT- 10	Soil	280	9714	1	<MDA	<MDA	<MDA	<MDA
NMNT- 11	Concrete	429	199069	1	263	189739	3.34E+04	2.53E+06
NMNT- 12	Concrete	395	69145	2	213	60856	1.41E+04	4.56E+05
NMNT- 13	Concrete	330	84300	1	164	74970	2.08E+04	1.07E+06
NMNT- 14	Concrete	507	93237	2	325	84948	2.15E+04	6.15E+05
NMNT- 15	Soil	295	28269	2	<MDA	19980	<MDA	1.86E+05
NMNT- 16	Concrete	412	55440	2	230	47151	1.52E+04	3.66E+05
NMNT- 17	Concrete	376	123908	1	210	114578	2.66E+04	1.57E+06
NMNT- 18	Concrete	374	66769	2	192	58480	1.27E+04	4.40E+05
NMNT- 19	Concrete	371	116040	1	205	106710	2.60E+04	1.47E+06
NMNT- 20	Concrete	1122	82347	2	940	74058	6.21E+04	5.43E+05
NMNT- 22	Concrete	137	11710	1	<MDA	<MDA	<MDA	<MDA
NMNT- 23	Concrete	174	9679	1	<MDA	<MDA	<MDA	<MDA
NMNT- 24	Concrete	221	21702	2	<MDA	13413	<MDA	1.43E+05
NMNT- 25	Concrete	148	7703	1	<MDA	<MDA	<MDA	<MDA
NMNT- 26	Concrete	172	11119	2	<MDA	<MDA	<MDA	<MDA
NMNT- 27	Concrete	154	31225	1	<MDA	21895	<MDA	3.96E+05
NMNT- 28	Concrete	184	20163	2	<MDA	<MDA	<MDA	<MDA
NMNT- 29	Concrete	251	108160	1	<MDA	98830	<MDA	1.37E+06
NMNT- 30	Concrete	326	69836	2	<MDA	61547	<MDA	4.61E+05

Notes: a Provided by New Millennium Nuclear Technologies (NMNT).
b Calculated by Scientech, LLC.
c Estimated based on LSC efficiency for tritium.

TABLE B-1 (Continued)
SUMMARY OF ON-SITE CONCRETE AND SOIL LSC ANALYSIS

Sample Number	Media	Gross Cont Rate (cpm)[a]		LSC	Net Count Rates (cpm)[b]		Net Activity (pCi/g)[b]	
		H-3	Beta	Det #	H-3	Beta	H-3	Beta[c]
NMNT- 31	Concrete	263	120617	1	<MDA	111287	<MDA	1.53E+06
NMNT- 32	Concrete	296	52355	2	<MDA	44066	<MDA	3.45E+05
NMNT- 33	Concrete	179	52846	1	<MDA	43516	<MDA	6.71E+05
NMNT- 34	Concrete	235	32918	2	<MDA	24629	<MDA	2.17E+05
NMNT- 35	Concrete	243	27027	1	<MDA	17697	<MDA	3.43E+05
NMNT- 36	Concrete	242	27699	2	<MDA	19410	<MDA	1.83E+05
NMNT- 37	Concrete	280	93413	1	<MDA	84083	<MDA	1.19E+06
NMNT- 38	Concrete	308	40087	2	<MDA	31798	<MDA	2.64E+05
NMNT- 39	Concrete	329	75426	1	163	66096	2.07E+04	9.57E+05
NMNT- 40	Concrete	306	39294	2	<MDA	31005	<MDA	2.59E+05
NMNT- 41	Soil	317	16044	2	<MDA	<MDA	<MDA	<MDA
NMNT- 42	Concrete	317	31688	2	<MDA	23399	<MDA	2.09E+05
NMNT- 43	Soil	256	19622	2	<MDA	<MDA	<MDA	<MDA
NMNT- 44	Concrete	278	25481	2	<MDA	17192	<MDA	1.68E+05
NMNT- 45	Soil	222	8715	2	<MDA	<MDA	<MDA	<MDA
NMNT- 46	Soil	279	29250	2	<MDA	20961	<MDA	1.93E+05
NMNT- 47	Concrete	288	43431	2	<MDA	35142	<MDA	2.86E+05
NMNT- 48	Soil	236	34490	2	<MDA	26201	<MDA	2.28E+05
NMNT- 49	Soil	191	24486	2	<MDA	16197	<MDA	1.62E+05
NMNT- 50	Concrete	305	118769	1	139	109439	1.76E+04	1.51E+06
NMNT- 51	Concrete	1029	278369	2	847	270080	5.59E+04	1.84E+06
NMNT- 52	Concrete	457	232032	1	291	222702	3.69E+04	2.94E+06
NMNT- 53	Concrete	455	82031	2	273	73742	1.80E+04	5.41E+05
NMNT- 54	Concrete	160	8510	1	<MDA	<MDA	<MDA	<MDA
NMNT- 55	Concrete	171	11245	2	<MDA	<MDA	<MDA	<MDA
NMNT- 56	Concrete	193	11358	1	<MDA	<MDA	<MDA	<MDA
NMNT- 57	Concrete	213	5320	2	<MDA	<MDA	<MDA	<MDA
NMNT- 58	Concrete	147	24669	1	<MDA	15339	<MDA	3.13E+05
NMNT- 59	Concrete	159	15915	1	<MDA	<MDA	<MDA	<MDA
NMNT- 60	Concrete	197	32769	2	<MDA	24480	<MDA	2.16E+05

Notes: a Provided by New Millennium Nuclear Technologies (NMNT).
b Calculated by Scientech, LLC
c Estimated based on LSC efficiency for tritium.

TABLE B-1 (Continued)
SUMMARY OF ON-SITE CONCRETE AND SOIL LSC ANALYSIS

Sample Number	Media	Gross Cont Rate (cpm)[a]		LSC Det #	Net Count Rates (cpm)[b]		Net Activity (pCi/g)[b]	
		H-3	Beta		H-3	Beta	H-3	Beta[c]
NMNT- 61	Concrete	130	8084	1	<MDA	<MDA	<MDA	<MDA
NMNT- 62	Concrete	137	7807	2	<MDA	<MDA	<MDA	<MDA
NMNT- 63	Concrete	158	9881	1	<MDA	<MDA	<MDA	<MDA
NMNT- 64	Concrete	182	7831	1	<MDA	<MDA	<MDA	<MDA
NMNT- 65	Concrete	206	13448	2	<MDA	<MDA	<MDA	<MDA
NMNT- 66	Concrete	199	20104	1	<MDA	10774	<MDA	2.55E+05
NMNT- 67	Concrete	219	9842	2	<MDA	<MDA	<MDA	<MDA
NMNT- 68	Concrete	318	102154	1	152	92824	1.93E+04	1.30E+06
NMNT- 69	Concrete	329	42384	2	<MDA	34095	<MDA	2.80E+05
NMNT- 70	Concrete	200	37809	1	<MDA	28479	<MDA	4.80E+05
NMNT- 71	Concrete	181	14868	1	<MDA	<MDA	<MDA	<MDA
NMNT- 72	Concrete	187	18271	2	<MDA	<MDA	<MDA	<MDA
NMNT- 73	Concrete	171	23455	1	<MDA	14125	<MDA	2.98E+05
NMNT- 74	Concrete	312	29955	2	<MDA	21666	<MDA	1.98E+05
NMNT- 75	Graphite	208	30743	1	<MDA	21413	<MDA	3.90E+05
NMNT- 76	Graphite	189	4651	2	<MDA	<MDA	<MDA	<MDA
NMNT- 77	Graphite	292	5197	1	<MDA	<MDA	<MDA	<MDA
NMNT- 78	Graphite	567	4218	2	385	<MDA	2.54E+04	<MDA

Notes: a Provided by New Millennium Nuclear Technologies (NMNT).
b Calculated by Scientech, LLC.
c Estimated based on LSC efficiency for tritium.

TABLE B-2
SUMMARY OF ON-SITE CONCRETE AND SOIL
GAMMA SPECTROSCOPY ANALYSIS

Sample No.	Media	Location	Isotope	Weighted Mean Concentration (uCi/g)	Weighted Mean +/- Error (1 σ)	MDA	
NMNT-01	Concrete	Bkgd Bioshield	K-40	3.59E-06	1.07E-05	3.86E-05	<MDA
			Co-60	3.52E-06	8.24E-07	3.25E-06	-
NMNT-02	Concrete	Bkgnd Floor	K-40	6.81E-08	8.24E-06	3.04E-05	<MDA
			Co-60	7.11E-07	5.18E-07	2.29E-06	<MDA
NMNT-03	Concrete	Floor Loc. 3	K-40	5.12E-06	1.08E-05	3.80E-05	<MDA
			Co-60	6.18E-07	6.72E-07	3.29E-06	<MDA
			Eu-152	3.58E-06	1.14E-06	4.21E-06	<MDA
NMNT-04	Soil	Floor Loc. 3	K-40	5.11E-06	6.03E-06	2.09E-05	<MDA
NMNT-05	Concrete	Floor Loc. 4	K-40	8.19E-06	1.01E-05	3.49E-05	<MDA
			Co-60	1.51E-06	9.28E-07	3.03E-06	<MDA
			Cs-137	9.37E-07	4.69E-07	1.45E-06	<MDA
NMNT-06	Concrete	Floor Loc. 5	K-40	-3.36E-06	1.05E-05	3.97E-06	<MDA
			Co-60	9.67E-07	6.77E-07	3.16E-06	<MDA
			Eu-154	4.91E-06	1.50E-06	4.25E-06	-
NMNT-07	Concrete	Floor Loc. 6	K-40	-5.28E-06	1.16E-05	4.45E-05	<MDA
			Co-60	7.16E-07	8.21E-07	3.98E-06	<MDA
NMNT-08	Concrete	Floor Loc. 7	K-40	-1.46E-06	1.01E-05	3.79E-05	<MDA
NMNT-09	Concrete	Floor Loc. 8	K-40	-9.19E-06	1.13E-05	4.46E-05	<MDA
NMNT-10	Soil	Floor Loc. 8	K-40	1.27E-06	9.48E-06	3.46E-05	<MDA
			Co-60	1.29E-06	6.20E-07	2.81E-06	<MDA
NMNT-11	Concrete	Floor Loc. 9	K-40	1.12E-07	9.31E-06	3.43E-05	<MDA
			Co-60	1.39E-06	6.42E-07	2.81E-06	<MDA
NMNT-12	Concrete	Floor Loc. 10	K-40	1.86E-06	1.19E-05	4.37E-05	<MDA
NMNT-13	Concrete	Floor Loc. 11	K-40	5.77E-06	1.25E-05	4.46E-05	<MDA
			Co-60	1.26E-06	8.23E-07	3.70E-06	<MDA
NMNT-14	Concrete	Floor Loc. 12	K-40	-2.00E-06	1.25E-05	4.67E-05	<MDA
NMNT-15	Soil	Floor Loc. 12	K-40	4.92E-06	5.02E-06	1.72E-05	<MDA
NMNT-16	Concrete	Floor Loc. 13	K-40	1.12E-05	1.32E-05	4.57E-05	<MDA
			Co-60	6.97E-07	7.76E-07	3.67E-06	<MDA
NMNT-17	Concrete	Floor Loc. 14	K-40	5.93E-06	1.36E-05	4.85E-05	<MDA
			Co-60	2.68E-06	1.29E-06	4.04E-06	<MDA
			Cs-137	1.40E-06	6.12E-07	1.79E-06	<MDA
NMNT-18	Concrete	Floor Loc. 15	K-40	2.80E-07	1.16E-05	4.30E-05	<MDA
NMNT-19	Concrete	Floor Loc. 16	K-40	2.68E-06	1.59E-05	5.83E-05	<MDA
			Co-60	1.77E-06	1.03E-06	4.72E-06	<MDA
			Ra-226+D	3.31E-06	9.09E-07	3.10E-06	-

TABLE B-2 (Continued)
SUMMARY OF ON-SITE CONCRETE AND SOIL GAMMA SPECTROSCOPY ANALYSIS

Sample No.	Media	Location	Isotope	Weighted Mean Concentration (uCi/g)	Weighted Mean +/- Error (1 σ)	MDA	
NMNT-20	Concrete	Bioshield Loc. 17	K-40	4.89E-04	4.96E-05	1.39E-04	-
			Co-60	8.39E-03	1.83E-04	6.07E-05	-
			Eu-152	9.01E-03	1.42E-04	8.79E-05	-
			Eu-154	5.78E-04	2.28E-05	8.01E-05	-
NMNT-21	Metal (Al)	Al from tank wall	K-40	3.87E-06	2.85E-04	1.04E-03	<MDA
			Co-60	8.99E-03	2.35E-04	1.69E-04	-
			Eu-152	4.73E-03	1.60E-04	2.74E-04	-
NMNT-21a	Resin	Filter resin	K-40	1.30E-07	9.28E-07	3.35E-06	<MDA
			Co-60	4.80E-05	7.94E-07	5.25E-07	-
NMNT-22	Concrete	Bioshield Loc. 18-1	K-40	-9.41E-08	9.34E-06	3.46E-05	<MDA
			Co-60	3.07E-05	1.72E-06	3.42E-06	-
			Eu-152	3.31E-05	1.89E-06	5.13E-06	-
NMNT-23	Concrete	Bioshield Loc. 18-2	K-40	1.31E-05	9.53E-06	3.17E-05	<MDA
NMNT-24	Concrete	Bioshield Loc. 18-3	Co-60	9.78E-07	6.10E-07	2.82E-06	<MDA
			Ra-226+D	1.29E-06	4.47E-06	1.82E-06	<MDA
NMNT-25	Concrete	Bioshield Loc. 19-1	K-40	1.38E-07	3.67E-06	1.35E-05	<MDA
			Co-60	3.65E-07	2.61E-07	1.22E-06	<MDA
NMNT-26	Concrete	Bioshield Loc. 19-2	K-40	5.75E-06	4.31E-06	1.44E-05	<MDA
			Co-60	1.51E-05	7.59E-07	1.39E-06	-
			Eu-152	1.79E-05	9.01E-07	2.22E-06	-
NMNT-27	Concrete	Bioshield Loc. 20-1	K-40	0.00E+00	5.88E-06	2.19E-05	<MDA
			Co-60	1.96E-06	4.65E-07	1.77E-06	-
			Cs-137	2.97E-07	2.46E-07	8.43E-07	<MDA
NMNT-28	Concrete	Bioshield Loc. 20-2	Co-60	8.08E-06	8.94E-07	2.67E-06	-
			Eu-152	7.13E-06	8.77E-07	2.93E-06	-
NMNT-29	Concrete	Bioshield Loc. 20-3	K-40	-1.47E-06	1.09E-05	4.08E-05	<MDA
			Co-60	2.31E-05	1.62E-06	3.85E-06	-
			Eu-152	3.13E-05	1.95E-06	5.75E-06	-
NMNT-30	Concrete	Bioshield Loc. 20-4	K-40	6.32E-06	1.10E-05	3.88E-05	<MDA
			Co-60	6.67E-05	2.83E-06	4.37E-06	-
			Eu-152	5.91E-05	2.85E-06	7.20E-06	-
NMNT-31	Concrete	Bioshield Loc. 20-5	Co-60	8.01E-05	2.92E-06	3.10E-06	-
			Eu-152	7.90E-05	3.22E-06	6.69E-06	-
NMNT-32	Concrete	Bioshield Loc. 21-1	K-40	-4.60E-07	5.89E-06	2.17E-05	<MDA
			Co-60	1.26E-05	8.51E-07	2.02E-06	-
			Eu-152	9.39E-06	8.08E-07	2.55E-06	-

TABLE B-2 (Continued)
SUMMARY OF ON-SITE CONCRETE AND SOIL GAMMA SPECTROSCOPY ANALYSIS

Sample No.	Media	Location	Isotope	Weighted Mean Concentration (uCi/g)	Weighted Mean +/- Error (1 σ)	MDA	
NMNT-33	Concrete	Bioshield Loc. 21-2	K-40	4.92E-06	5.85E-06	2.03E-05	<MDA
			Co-60	1.22E-05	8.17E-07	1.78E-06	-
			Cs-137	6.51E-07	3.43E-07	1.09E-06	<MDA
			Eu-152	1.02E-05	8.19E-07	2.50E-06	-
NMNT-34	Concrete	Bioshield Loc. 21-3	K-40	1.38E-05	1.01E-05	3.37E-05	<MDA
			Co-60	5.89E-05	2.46E-06	3.57E-06	-
			Eu-152	5.26E-05	2.55E-06	6.53E-06	-
NMNT-35	Concrete	Bioshield Loc. 21-4	Co-60	2.91E-04	7.80E-06	6.36E-06	-
			Eu-152	2.62E-04	7.62E-06	1.23E-05	-
NMNT-36	Concrete	Bioshield Loc. 21-5	Co-60	5.33E-04	1.27E-05	6.30E-06	-
			Eu-152	5.34E-04	1.21E-05	1.56E-05	-
			Eu-154	3.26E-05	4.28E-06	1.35E-05	-
NMNT-37	Concrete	BST Floor Loc. 22	K-40	6.51E-06	1.23E-05	4.34E-05	<MDA
			Co-60	2.54E-04	7.20E-06	5.01E-06	-
			Cs-137	3.01E-06	1.35E-06	4.32E-06	<MDA
			Eu-152	8.15E-05	3.99E-06	1.09E-05	-
			Eu-154	1.06E-05	2.45E-06	7.19E-06	-
NMNT-38	Concrete	BST Floor Loc. 23	K-40	-5.93E-06	1.27E-05	4.85E-05	<MDA
			Co-60	8.33E-05	3.50E-06	5.54E-06	-
			Cs-137	2.13E-06	8.93E-07	2.72E-06	<MDA
			Eu-152	8.91E-06	1.49E-06	6.13E-06	-
NMNT-39	Concrete	BST Floor Loc. 24	K-40	-1.29E-07	1.34E-05	4.96E-05	<MDA
			Co-60	2.86E-06	9.62E-07	4.12E-06	<MDA
NMNT-40	Concrete	Tunnel Loc. 25	K-40	-3.96E-06	1.28E-05	4.85E-05	<MDA
			Co-60	2.36E-05	1.82E-06	4.73E-06	-
			Eu-152	1.87E-05	1.67E-06	5.56E-06	-
NMNT-41	Soil	Tunnel Loc. 25	Co-60	4.85E-06			
NMNT-42	Concrete	Tunnel Loc. 26	K-40	-1.87E-06	1.19E-05	4.48E-05	<MDA
			Co-60	1.64E-06	8.20E-07	3.42E-06	<MDA
NMNT-43	Soil	Tunnel Loc. 27	Co-60	1.39E-05			
NMNT-44	Concrete	Tunnel Loc. 28	K-40	2.43E-05	1.15E-05	3.61E-05	<MDA
NMNT-45	Soil	Tunnel Loc. 29	K-40	1.54E-05	1.87E-05	6.51E-05	<MDA
			Co-60	3.26E-06	1.21E-06	5.10E-06	<MDA
NMNT-46	Soil	Tunnel Loc. 30	K-40	-6.77E-06	1.16E-05	4.47E-05	<MDA
NMNT-47	Concrete	Tunnel Loc. 31	K-40	7.44E-06	8.97E-06	3.11E-05	<MDA
NMNT-48	Soil	Tunnel Loc. 31	Co-60	4.00E-06			<MDA
NMNT-49	Soil	Tunnel Loc. 32	K-40	7.25E-06	1.67E-05	5.98E-05	<MDA
			Co-60	4.56E-06	1.24E-06	5.20E-06	<MDA

TABLE B-2 (Continued)
SUMMARY OF ON-SITE CONCRETE AND SOIL
GAMMA SPECTROSCOPY ANALYSIS

Sample No.	Media	Location	Isotope	Weighted Mean Concentration (uCi/g)	Weighted Mean +/- Error (1 σ)	MDA	
NMNT-50	Concrete	Tunnel Ceiling 33-1	K-40	7.53E-09	8.21E-06	3.04E-05	<MDA
			Co-60	2.73E-07	5.24E-07	2.46E-06	<MDA
NMNT-51	Concrete	Tunnel Ceiling 33-2	Co-60	1.56E-06	6.09E-07	2.60E-06	<MDA
NMNT-52	Concrete	Tunnel Ceiling 33-3	K-40	-4.52E-06	7.15E-06	2.77E-05	<MDA
NMNT-53	Concrete	Tunnel Ceiling 33-4	K-40	1.39E-05	9.04E-06	2.98E-05	<MDA
NMNT-54	Concrete	Bioshield Loc. 34-1	K-40	-3.79E-06	5.77E-06	2.24E-05	<MDA
			Co-60	1.90E-06	4.91E-07	1.95E-06	<MDA
NMNT-55	Concrete	Bioshield Loc. 34-2	K-40	8.90E-06	6.93E-06	2.32E-05	<MDA
NMNT-56	Concrete	Bioshield Loc. 34-3	Co-60	1.27E-06	6.38E-07	2.76E-06	<MDA
NMNT-57	Concrete	Bioshield Loc. 34-4	K-40	-1.66E-07	8.12E-06	3.00E-05	<MDA
NMNT-58	Concrete	Bioshield Loc. 34-5	K-40	3.35E-06	7.49E-06	2.68E-05	<MDA
			Co-60	8.88E-07	5.04E-07	2.23E-06	<MDA
NMNT-59	Concrete	Bioshield Loc. 34-6	K-40	5.50E-06	9.13E-06	3.24E-05	<MDA
			Co-60	6.40E-06	8.77E-07	2.99E-06	-
			Eu-152	4.44E-06	5.54E-07	3.72E-06	-
NMNT-60	Concrete	Bioshield Loc. 34-7	K-40	-1.60E-05	1.17E-05	4.51E-05	<MDA
			Co-60	3.88E-04	9.93E-06	6.63E-06	-
			Eu-152	4.76E-04	1.15E-05	1.55E-05	-
			Eu-154	2.69E-05	3.81E-06	1.02E-05	-
NMNT-61	Concrete	Bioshield Loc. 35-1	K-40	5.68E-06	6.20E-06	2.14E-05	<MDA
			Co-60	1.59E-06	4.36E-07	1.74E-06	<MDA
NMNT-62	Concrete	Bioshield Loc. 35-2	K-40	-1.61E-06	1.13E-05	4.19E-05	<MDA
			Co-60	5.55E-05	2.55E-06	4.72E-06	-
			Eu-152	6.02E-05	2.86E-06	7.28E-06	-
NMNT-63	Concrete	Bioshield Loc. 35-3	K-40	1.82E-06	5.49E-06	1.98E-05	<MDA
			Co-60	1.02E-06	3.92E-07	1.74E-06	<MDA
NMNT-64	Concrete	Bioshield Loc. 35-4	K-40	-3.43E-07	9.62E-06	3.56E-05	<MDA
			Co-60	2.13E-06	6.89E-07	2.80E-06	<MDA
NMNT-65	Concrete	Bioshield Loc. 35-5	K-40	-5.30E-06	1.21E-05	4.57E-05	<MDA
			Co-60	1.75E-05	1.50E-06	3.94E-06	-
			Eu-152	1.09E-05	1.40E-06	4.87E-06	-
NMNT-66	Concrete	Bioshield Loc. 35-6	Co-60	2.32E-04	6.29E-06	4.51E-06	-
			Eu-152	2.06E-04	5.76E-06	9.70E-06	-
			Eu-154	1.29E-05	2.49E-06	7.35E-06	-
NMNT-67	Concrete	Bioshield Loc. 35-7	Co-60	3.55E-03	7.83E-05	2.89E-05	-
			Eu-152	3.89E-03	6.54E-05	5.03E-05	-
			Eu-154	2.29E-04	1.39E-05	4.18E-05	-

TABLE B-2 (Continued)
SUMMARY OF ON-SITE CONCRETE AND SOIL
GAMMA SPECTROSCOPY ANALYSIS

Sample No.	Media	Location	Isotope	Weighted Mean Concentration (uCi/g)	Weighted Mean +/- Error (1 σ)	MDA	
NMNT-68	Concrete	Floor Loc. 36 (Bay)	K-40	-2.94E-06	9.06E-06	3.41E-05	<MDA
			Co-60	8.26E-05	3.00E-06	3.31E-06	-
			Eu-152	8.82E-05	3.44E-06	6.94E-06	-
NMNT-69	Concrete	Floor Loc. 37 (Bay)	K-40	-4.93E-06	9.91E-06	3.82E-05	<MDA
			Co-60	2.83E-06	8.07E-07	3.38E-06	<MDA
NMNT-70	Concrete	Therm. Col. Loc. 38	K-40	8.21E-06	9.28E-06	3.20E-05	<MDA
			Co-60	2.03E-04	5.67E-06	4.59E-06	-
			Eu-152	2.41E-04	6.17E-06	9.44E-06	-
			Eu-154	2.37E-05	2.90E-06	6.78E-06	-
NMNT-71	Concrete	Bioshield Loc. 39	K-40	0.00E+00	1.29E-05	4.75E-05	<MDA
			Co-60	2.65E-04	7.63E-06	6.38E-06	-
			Eu-152	1.81E-04	6.43E-06	1.13E-05	-
			Eu-154	1.48E-05	3.45E-06	1.00E-05	-
NMNT-72	Concrete	Bioshield Loc. 40	K-40	-5.94E-06	7.80E-06	3.10E-05	<MDA
			Co-60	2.68E-05	1.49E-06	2.98E-06	-
			Eu-152	1.59E-05	1.24E-06	4.44E-06	-
NMNT-73	Concrete	Bioshield Loc. 41	K-40	2.89E-06	5.09E-06	1.80E-05	<MDA
			Co-60	4.74E-05	1.69E-06	1.89E-06	-
			Eu-152	3.87E-05	1.66E-06	4.11E-06	-
NMNT-74	Concrete	Bioshield Loc. 42	K-40	2.44E-07	8.10E-06	2.98E-05	<MDA
			Co-60	6.02E-06	7.67E-07	2.48E-06	-
			Eu-152	5.15E-06	7.65E-07	2.97E-06	-
NMNT-75	Graphite	Graphite 1	K-40	-1.09E-05	1.21E-05	4.74E-05	<MDA
			Co-60	1.19E-04	4.31E-06	5.64E-06	-
			Eu-152	1.09E-04	4.55E-05	7.78E-06	-
NMNT-76	Graphite	Graphite 2	K-40	1.27E-05	9.98E-06	3.35E-05	<MDA
			Co-60	4.91E-06	8.10E-07	2.78E-06	-
			Eu-152	1.46E-05	1.16E-06	3.76E-06	-
NMNT-77	Graphite	Graphite 3	Co-60	3.56E-05	2.91E-06	7.95E-06	-
			Eu-152	5.43E-03	8.58E-05	4.97E-05	-
			Eu-154	3.35E-04	1.04E-05	3.59E-05	-
NMNT-78	Graphite	Graphite 4	Co-60	1.05E-04	6.56E-06	1.77E-05	-
			Eu-152	1.72E-02	2.61E-04	1.11E-04	-
			Eu-154	1.20E-03	2.86E-05	8.16E-05	-
NMNT-79	Metal (Al)	Grid Plate Center	Co-60	1.10E-01	2.38E-03	6.28E-04	-
			Eu-152	6.35E-03	2.06E-04	3.87E-04	-
NMNT-80	Metal (Al)	Grid Plate Edge	Co-60	1.21E-01	2.61E-03	7.03E-04	-
			Cs-137	7.50E-05	6.45E-05	2.13E-04	<MDA

TABLE B-2 (Continued)
SUMMARY OF ON-SITE CONCRETE AND SOIL
GAMMA SPECTROSCOPY ANALYSIS

Sample No.	Media	Location	Isotope	Weighted Mean Concentration (uCi/g)	Weighted Mean +/- Error (1 σ)	MDA	
NMNT-81	Metal (SS)	Grid Plate Bolt	Co-60	6.00E+01	1.30E+00	3.08E-01	-
NMNT-82	Metal (Al)	Pipe @ 16"	K-40	2.67E-05	1.87E-04	6.81E-04	<MDA
			Co-60	2.64E-03	8.29E-05	7.30E-05	-
NMNT-83	Metal (Al)	Pipe @ 24"	Co-60	6.53E-05	5.81E-06	1.62E-05	-
			Eu-152	9.62E-05	6.95E-06	2.23E-05	-
NMNT-84	Metal (Al)	Pipe @ 32"	K-40	4.06E-06	8.97E-05	3.29E-04	<MDA
			Co-60	1.20E-04	1.06E-05	3.06E-05	-
			Eu-152	6.51E-05	8.99E-06	3.44E-05	-
NMNT-85	Metal (Al)	Pipe @ 8"	Co-60	9.91E-05	6.79E-06	3.10E-05	-
			Eu-152	6.85E-03	1.32E-04	9.59E-05	-
			Eu-154	4.44E-04	2.87E-05	8.53E-05	-
NMNT-86	Soil	Floor Loc. 5	K-40	4.69E-06	4.90E-06	1.69E-05	<MDA
			Co-60	4.56E-06	4.92E-07	1.53E-06	-
NMNT-87	Soil	Trench loc. 43-1	K-40	1.84E-05	2.25E-05	7.82E-05	<MDA
			Co-60	9.59E-06	1.77E-07	6.62E-06	-
NMNT-88	Soil	Trench loc. 43-2	K-40	2.82E-06	3.34E-06	1.16E-05	<MDA
			Co-60	1.14E-06	2.51E-07	9.78E-07	-
Boral	Metal (Boral)	Boral Curtain	K-40	-9.33E-05	1.45E-04	5.35E-04	<MDA
			Co-60	1.48E-02	3.34E-04	1.52E-04	-
			Eu-152	5.14E-04	4.35E-05	1.91E-04	-
Bkgnd Soil	Soil	Background soil	K-40	9.87E-06	3.52E-06	1.71E-05	<MDA
			Co-60	1.61E-06	3.68E-07	1.46E-06	-
Unknown 1	Soil	Off-site sand	K-40	2.49E-06	1.05E-06	2.35E-06	-
			Co-60	3.42E-07	7.27E-08	2.82E-07	-
Unknown 2	Soil	Subsurface soil	K-40	4.95E-06	1.95E-06	5.93E-06	<MDA
			Co-60	8.37E-07	1.43E-07	4.81E-07	-

ENVIRONMENTAL REPORT

NUCLEAR RESEARCH LABORATORY
UNIVERSITY OF ILLINOIS AT CHAMPAIGN-URBANA

Prepared by:
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143 West Street
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December 2005

Project Application	*Prepared By*	*Date*
23566	Kevin Taylor, PE, CHP	12/1/05

APPROVALS:

Title	*Signature*	*Date*
Technical Reviewer	Richard D. Moss, CSP	12/1/05
Operations Manager	Lee G. Penney	12/1/05

Revision Log

Revision Number	Affected Pages	CRA Number	Approval

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1.0 EXECUTIVE SUMMARY

Envirocare of Utah, LLC has conducted a review of the decommissioning options for the University of Illinois (University) Nuclear Research Laboratory (NRL) located on the university Urbana-Champaign campus in Urbana, Illinois. The NRL facility is located in a typical urban or suburban setting. It is a small site located on only about 5,000 square feet of University property surrounded by other structures and paved parking areas. The information provided in the following sections may not always apply directly to the exact location of the NRL facility but may apply in general to the Cities of Champaign, Urbana or Champaign County.

The review of the decommissioning options included an assessment of the impact of each decommissioning option on the environs surrounding the facility and the University. The assessment provides information on the site environs including a description of the current population distribution; a summary of current and future land use; descriptions of site meteorology, geology, seismology, climatology, surface and groundwater hydrology; descriptions of the natural and water resources at the site and surrounding areas; a description of the ecology of the site and surrounding areas; and a summary of the endangered species in the surrounding areas.

This Environmental Report (ER) was prepared in accordance with the guidance provided in Chapter 6.0 of the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Material and Safety and Safeguards' (NMSS) NUREG-1748, "Environmental Review Guidance for Licensing Actions Associated with NMSS Programs" (NRC 2003). This ER is designed to be used by the NRC in conducting its environmental assessment in accordance with the National Environmental Policy Act (NEPA) of 1969. NEPA requires Federal agencies, as part of their decision-making process, to consider the environmental impacts of actions under their jurisdiction. The NRC's NEPA requirements are provided in 10 CFR 51.

2.0 PROPOSED ACTION

The University discontinued operations of its Advanced TRIGA nuclear research reactor at the NRL in 1998. Prior to shutdown of the reactor and the removal of the reactor fuel elements in 2004, the University decided to decommission and dismantle the NRL and terminate NRC Facility Operating License No. R-115. The University plans on beginning on-site decontamination and decommissioning (D&D) activities during the fourth quarter of 2007.

The NRL facility was used by University faculty and students for research, experiments, and classes beginning in 1960. It also provided services to other institutions and organizations for research and testing. As with other facilities of this nature, the NRL has become contaminated with various levels of radioactive materials. Because the University has decided that the reactor is no longer needed, D&D of the facility would eliminate the potential for future inadvertent environmental releases.

The action proposed is complete demolition of the facility and license termination. This action will require dismantlement and removal of clean and contaminated facility components, dismantlement of the facility structure, and disposal of the demolition waste according to their radiological condition. Some decontamination may be required to reduce radioactive waste volumes and to reduce the potential for the spread of contamination. As such, the final status survey (FSS) for license termination is expected to include only the site location (soil) and none of the facility structure or components.

The D&D project will be conducted in accordance with the following regulations and NRC guidance (NUREG) documents:

- 10 CFR 20, "Standards for the Protection Against Radiation"
- 10 CFR 20, Subpart E, "Radiological Criteria for License Termination"
- 10 CFR 51, "Environmental Protection Regulations for Domestic Licensing and Regulated Regulatory Functions"
- NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual"
- NUREG-1537-PT-1, "Guidelines for Preparing and Reviewing Applications for Licensing of Non-Power Reactors"
- NUREG-1757, "Consolidated NMSS Decommissioning Guidance"

2.1 ALTERNATIVES

The decommissioning alternatives and the proposed action are described in the following sections. The alternatives include the no-action alternative and the facility re-use alternative.

2.1.1 No-Action Alternative

The no-action alternative would involve leaving the NRL facility in its current condition. Instituting the no-action alternative would require the following:

- Maintaining current radiological controls
- Maintaining site security
- Maintaining NRC License R-115
- Maintaining a Reactor Administrator
- Maintaining the NRL facility structure and systems (power, heating, air conditioning, ventilation, etc.)

The no-action alternative, however, is not allowed under NRC regulations, specifically, NRC regulation 10 CFR 30.36 (the Timeliness Rule). This requires licensees to decommission their facilities when licensed activities cease and to request termination of their licenses. The purpose of the Timeliness Rule is to reduce the potential risk to the public and environment that may result from delayed decommissioning of inactive facilities and sites. Although, the Timeliness Rule includes options for license termination or transfer to another entity, the licensee is still expected to initiate and complete decommissioning in a timely manner.

2.1.2 Proposed Action

The proposed action, facility decommissioning and NRC license termination, will result in the greatest benefit to the University. The proposed action, which will be directed by the University's College of Engineering, will result in the removal of all radioactively contaminated materials from the facility area and unrestricted use of the area. The decommissioning approach will be detailed in the NRC-approved Decommissioning Plan.

With the facility structure completely removed, the final status survey for license termination and unrestricted release of the area will involve the survey and sampling of an open excavation. The final status survey will be conducted in accordance with the "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" (NRC 2000).

Following license termination, the facility footprint will be backfilled and will likely be converted to a paved parking area. However, because the area will be released for unrestricted use, the area will be available for future construction. No long-term monitoring needs are expected for the unrestricted release.

2.1.3 Alternative Action (Decontamination)

The only alternative for D&D would be removal of the radioactive components and subsequent release of the building for unrestricted use. This alternative, while similar to the proposed alternative in addressing radioactive and hazardous materials, would leave the existing structure in place. The final status would involve surveys of the remaining structure and systems and sampling of the soil around and under the structure.

Due to the age of the building and minimal services into the building, it was determined that reuse of the building would not be in the best interests of the University. Furthermore, the radiological release of a structure is more complicated and requires a greater level of survey and sampling work than releasing an open excavation.

3.0 SITE LOCATION AND DESCRIPTION

The NRL facility contains the University's TRIGA research nuclear reactor. The University campus is located in the adjoining cities of Urbana and Champaign and is centered on the dividing line of these cities. A topographical map of the University area is provided as Figure 3-1 with the NRL facility located in the center of the map.

The NRL building is located in the northeastern portion of campus between Springfield Street to the north, Green Street to the south, Goodwin Street to the west, and Gregory Street to the east in the City of Urbana. The coordinates of the building are approximately 40° 6.73'N, 88° 13.40'W at an elevation of about 715 feet above mean sea level. The Figure 3-2 provides campus map with concentric circles with radii of approximately 600, 1,200, and 1,800 feet.

The location of the NRL building is provided in Figure 3-3. Concentric circles with radii of approximately 200, 400, and 600 feet are drawn to show the proximity of the adjacent buildings. In general, the NRL building is surrounded by research buildings associated with the University's College of Engineering. The most prominent buildings within 600 feet of the NRL building are the Materials Research Laboratory (MRL), Loomis Laboratory, and the Engineering Science Building. Beyond the immediate surroundings, structures include University dormitories to the south and southwest and private dwellings (homes and apartments) to the northeast across Springfield Avenue.

The streets surrounding the facility are shown in Figure 3-3. The major traffic flow from the NRL facility is east and west on Springfield Avenue and north and south on Lincoln Avenue. Interstate 74 lies about two miles north of the NRL facility and Interstate 57 lies about four miles to the west.

The City of Urbana is located about 110 miles southwest of Lake Michigan and about 35 miles from the Illinois-Indiana border. Urbana is approximately in the center of Champaign County, which covers an area of about 990 square miles in the east-central section of Illinois. The cities of Champaign and Urbana lie just to the southeast of the intersection of Interstates 74 and 57.

The NRL building is a steel frame concrete block building that is approximately 80 feet east-west by 45 feet north-south. The building is supported by 30 metal-shells, cast in place concrete piles with minimum lengths of 40.5 feet. A photograph of the building is provided in Figure 3-3. A 6.5-foot deep by 1-foot wide concrete footing, which is laid on concrete pile caps, supports the walls.

The Mechanical Equipment Room containing the heat exchanger and the primary and secondary cooling system pumps is located off the south side of the building and is accessed through a door on the reactor room floor. The exterior dimensions of the room are approximately 18 feet by 22 feet. The exterior height is less than 6 feet above grade. Located below the heat exchanger room is a vault containing two N-16 delay tanks.

The interior of the building contains three levels: the mezzanine or main level, the storage level (located above the mezzanine), and the lower level (reactor room). The mezzanine floor is 10 feet above the reactor room floor and the storage floor is 21 feet above the reactor room floor. The clearance from the reactor room floor to the roof supports is 35 feet. The reactor room is about 44 feet wide by 57 feet long. The mezzanine floor, storage floor, and roof are placed on standard bar joists which are tied to horizontal I-beams and the main support columns.

The reactor room floor is a six inch concrete slab laid on undisturbed earth. The mezzanine and storage floors are 2.5-inch concrete slabs poured on corrugated steel plate. The roof is composed of gravel on a gypsum roof deck which is covered with four-ply asphalt paper. A special concrete base is used to support the reactor and the thermal column door railway. Ten piles similar to those used to carry the building are used to support this special base.

TopoZone - The Web's Topographic Map　Page 1 of 1

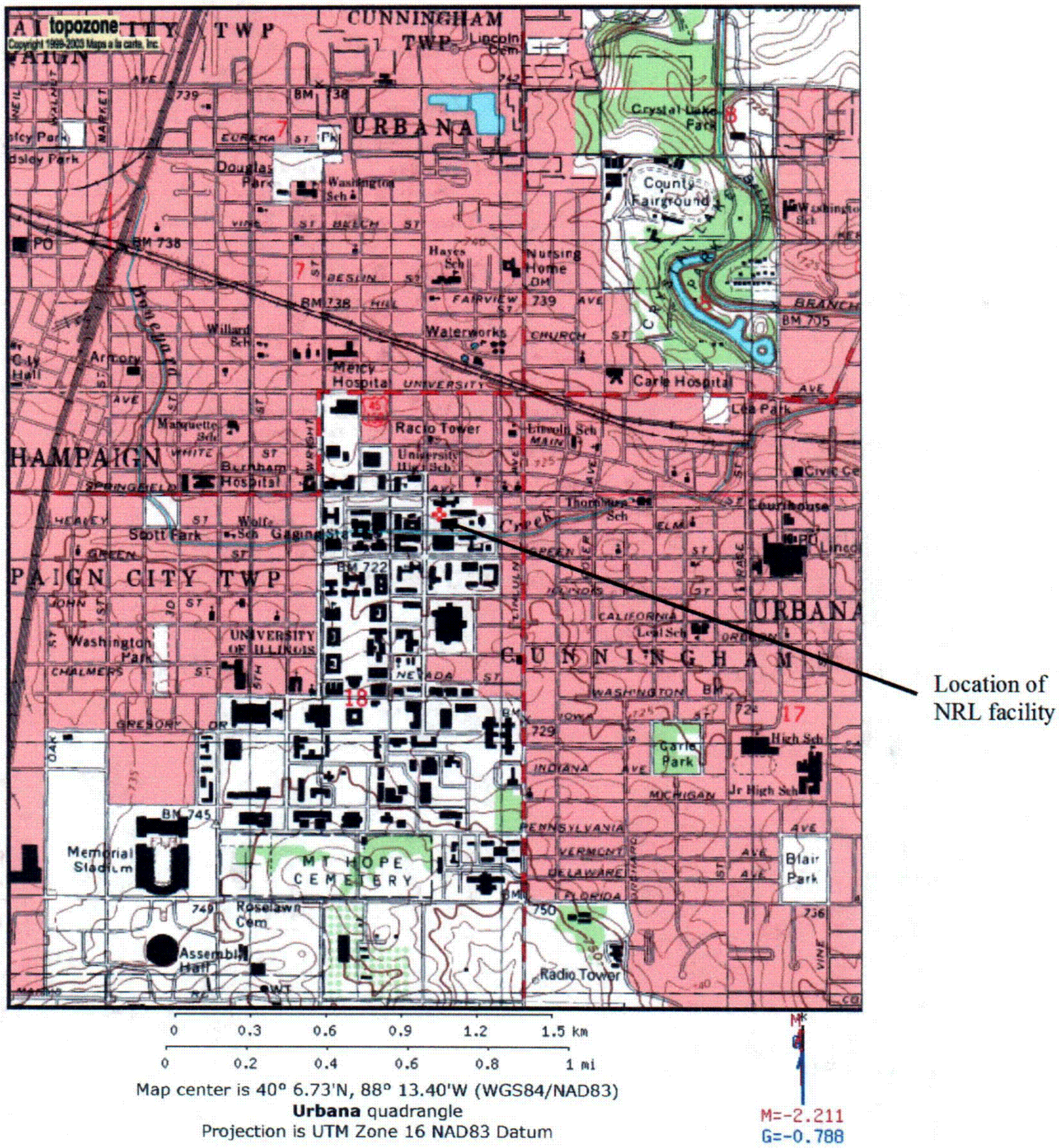

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FIGURE 3-1
UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN AREA TOPO MAP

C02

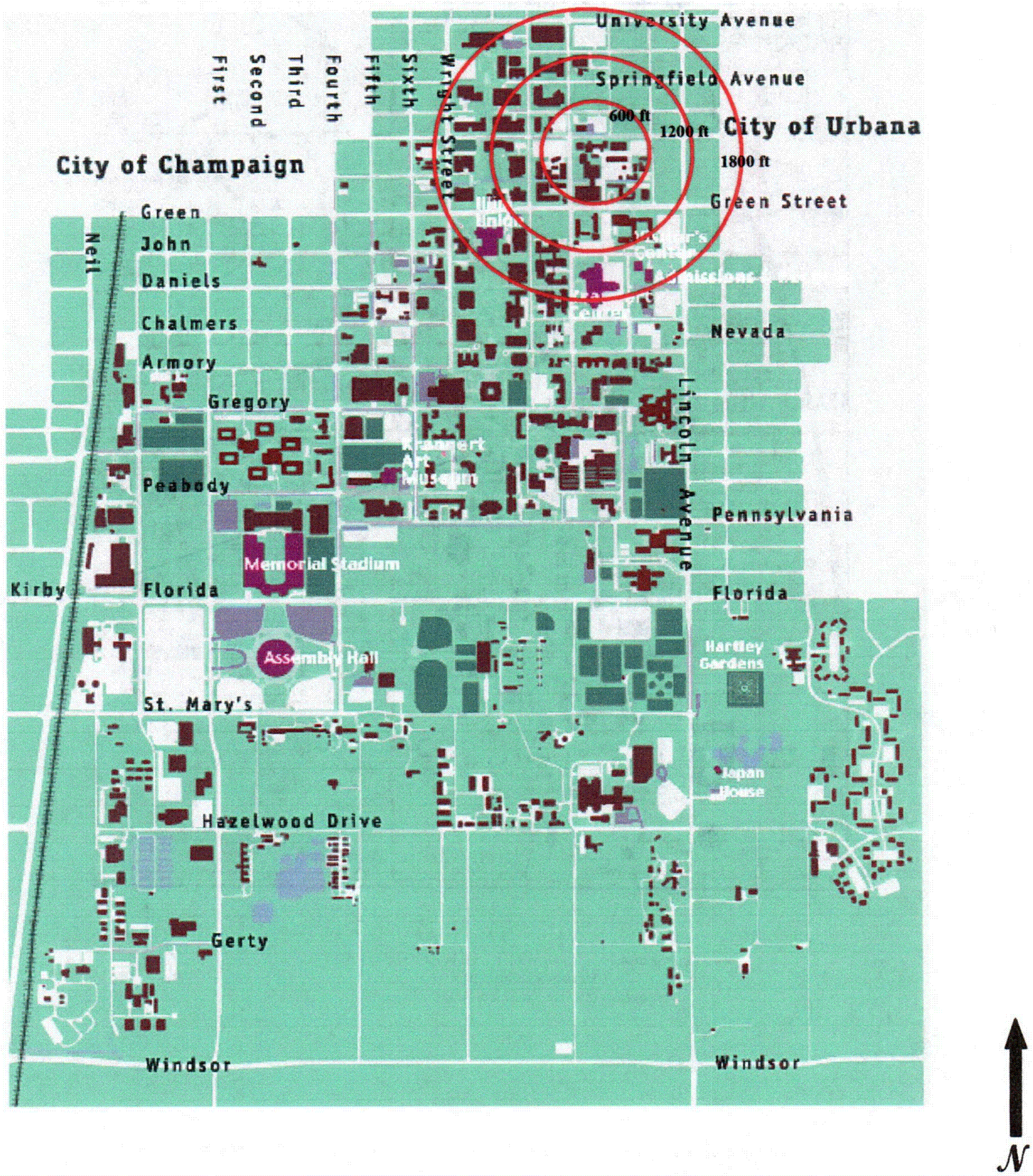

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FIGURE 3-2
UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN CAMPUS MAP

C03

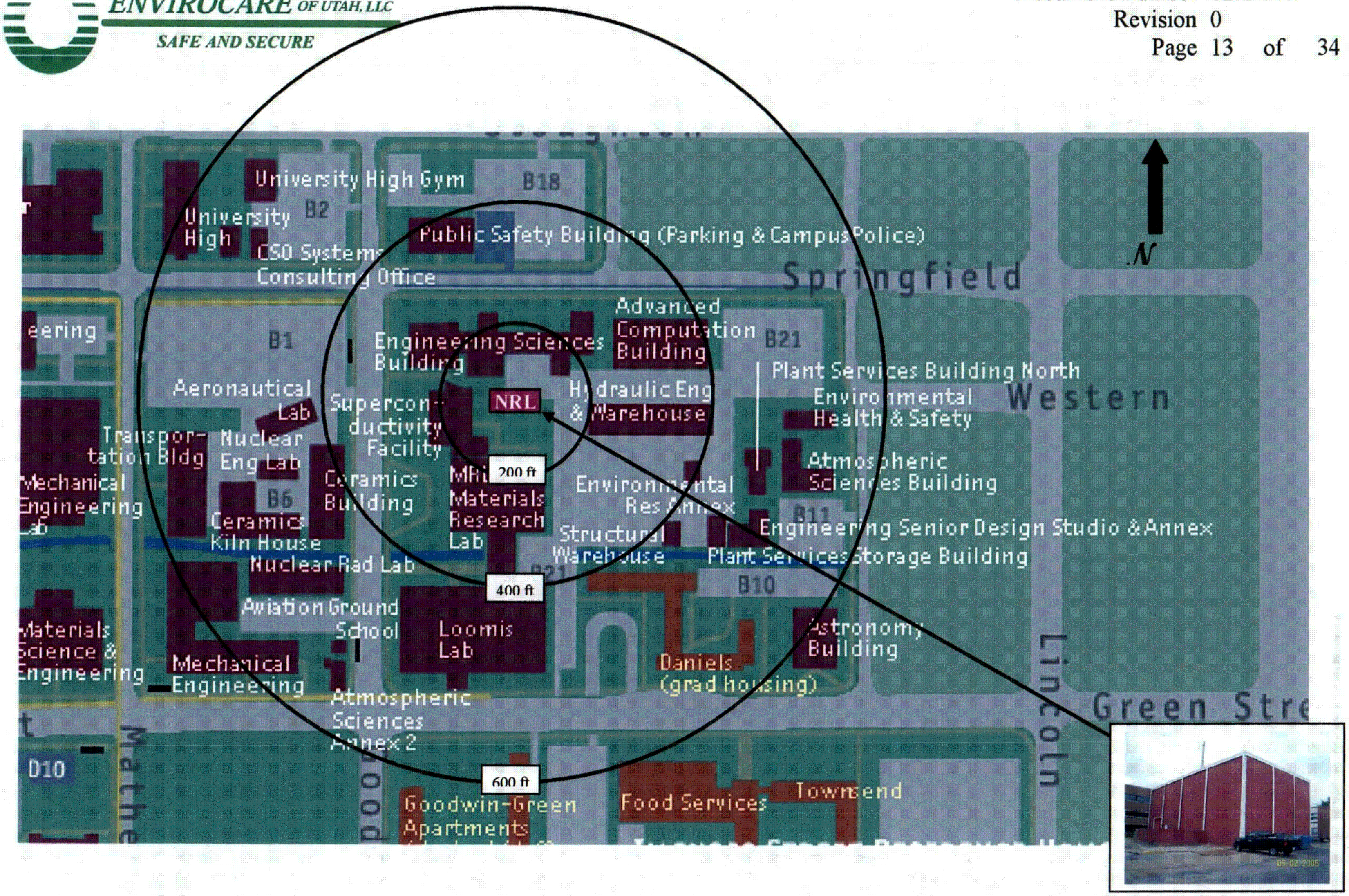

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FIGURE 3-3
UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN
AREA SURROUNDING NRL FACILITY

C04

4.0 DESCRIPTION OF THE AFFECTED ENVIRONMENT

The environment immediately surrounding the NRL facility is primarily man-made. The NRL facility contains radioactive materials and residual radioactive contamination resulting from past reactor operations. The facility also contains hazardous materials including elemental lead, asbestos, and possibly small amounts of mercury (Scientech 2005). While the facility dates back to the late 1950s, it has not been designated as a place of historical or cultural significance.

The following sections describe the environs surrounding the facility including the University Campus, the cities of Champaign and Urbana, and Champaign County. The information is provided to describe the baseline conditions so that the environmental impacts of the proposed D&D actions can be evaluated.

4.1 POPULATION DISTRIBUTION

The University is located in the Champaign-Urbana metropolitan area. The City of Champaign, which extends mostly west of the University, has a population of approximately 72,000 (US Census Bureau 2005a). The City of Urbana, which extends to the east of the University, has a population of about 39,000 (US Census Bureau 2005b). The student population of the University varies from semester to semester but total enrollment is more than 40,000 (University of Illinois 2005). Champaign County has a population of approximately 185,000 (US Census Bureau 2005c). The nearest residents are with 600 feet northeast of the NRL facility.

The populations of Champaign and Urbana have grown slightly from April 2000 to July 2003 with the Champaign population growing by 5% and the Urbana population growing by 2.8% (US Census Bureau 2003). The white population of the metropolitan area is about 70% with the African American and Asian populations making up around 15% and 10% of the total population respectively. The remaining minority population is primarily Hispanic.

4.2 CURRENT AND FUTURE LAND USE

The NRL facility currently contains the University's TRIGA research reactor which was shut down in 1998. The reactor fuel was removed in 2004; however, there is still a significant volume of radioactive material stored inside the building. Most of these materials are secured within a Radioactive Materials Storage Cage. The remaining radioactive materials are either contaminated or activated items located throughout the reactor room. The major contaminated or activated components of the facility include the reactor assembly, the reactor tank, the reactor bioshield, the primary cooling system, and the nitrogen-16 decay tanks (Scientech 2005).

The facility operations are currently governed by U.S. Nuclear Regulatory Commission (NRC) Facility Operating License No. R-115. The University is currently planning the complete demolition and decommissioning of the facility and eventual license termination. Therefore, there are no future plans for the use of the NRL facility.

Following the removal of all radioactive materials and complete demolition of the NRL facility, the foot-print of the building will become available for future development by the University. Once the area is backfilled with clean soil, potential uses include paving the area for additional parking or expansion of the neighboring buildings for office or laboratory space.

4.3 METEOROLOGY AND CLIMATOLOGY

The Champaign-Urbana area has a temperate continental climate; it is a region of climatic transition between northern and southern Illinois. The majority of the annual precipitation occurs during the summer growing season, thereby making the area exceptionally well suited for agriculture. The growing season is six months long with the average day of the last spring frost is April 15 and the average day of the first fall frost is October 17 (Illinois State Water Survey 2004).

4.3.1 Precipitation, Temperature, and Humidity

The average precipitation in the Champaign-Urbana area is 41.0 inches with an average snowfall of 26.2 inches (Illinois State Water Survey 2004). Historically, May and July are the months with the greatest amounts of precipitation and January and February have the least.

The average annual high and low temperatures for the area are approximately 61.3°F and 42.0°F (Illinois State Water Survey 2004). Average monthly temperatures range from 32.0°F in January to 85.2°F in July. On average, 7.6 days per year have a low temperature of 0°F or colder while 20.8 days per year have a high temperature 90°F or greater (Illinois State Climatologists Office 2002). Recorded extreme temperatures range from -25°F on four occasions (1988, 1905, 1994, and 1999) to 109°F (1954) (Illinois State Water Survey 2004).

The normal relative humidity in the area varies from an average minimum of about 59% in July to an average maximum of about 82% in December and January (University of Illinois 1971).

4.3.2 Wind

Information from the Springfield, Illinois Airport (approximately 18 miles southwest of the Site) indicates that the prevailing wind direction is from the south at an annual average speed of 9.61 knots (8.35 miles per hour) (Illinois State Climatologists Office 2004). The wind rose in Figure 4-1 is an indication of the typical wind direction and velocity. This data agrees well with wind rose data presented in the NRL facility Safety Analysis Report (University of Illinois 1971).

4.3.3 Storms, Tornadoes, Hail, Sleet, and Lightning

The Champaign-Urbana area is often unstable atmospherically during late spring and early summer. Approximately 41 thunderstorms per year occur in the area; total thunderstorm time averages four days per month in June and July, the months in which thunderstorms are most frequent (University of Illinois 1971).

There were 36 tornados reported in Champaign County during the period from 1950 through 1995 (Tornado Project Online 1999). A tornado is most likely to occur during the months of April and June. Though there is very little hail in the areas, sleet storms occur on average of six times per year (University of Illinois 1971).

Data on the intensity and frequency of lightning in the Champaign-Urbana area were not reviewed. However, the frequency of thunderstorms clearly indicates that this is an area of great lightning activity. There were 360 deaths and injuries due to lightning from 1959 through 1994 in Illinois (NOAA 1997). This ranks the state eleventh in lightning-related deaths and injuries for that period.

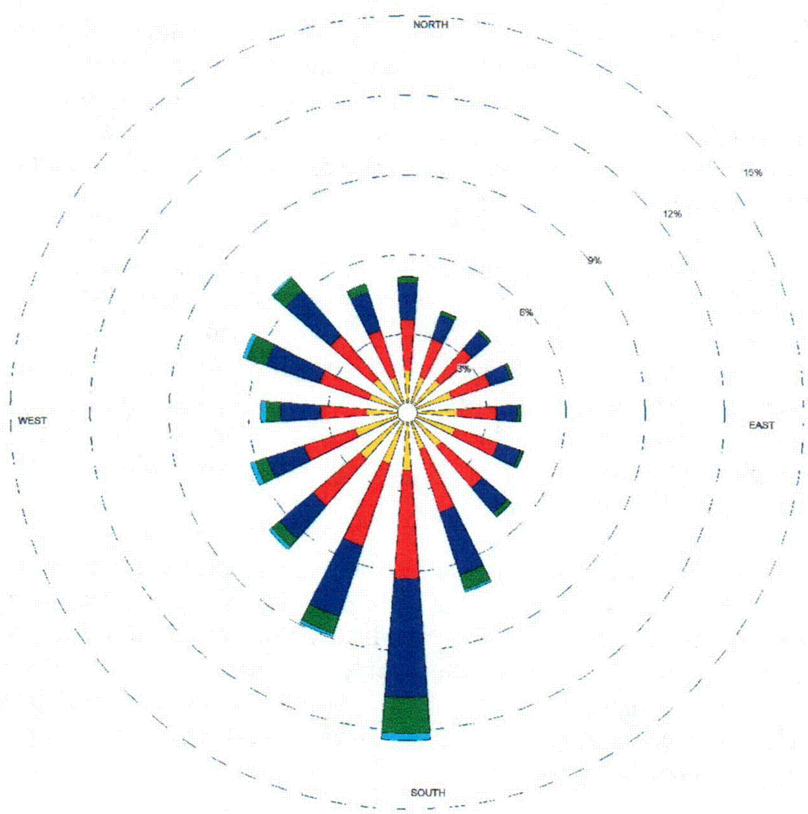
4.3.4 Air Quality

The 2003 Illinois Annual Air Quality Report reports that the particulate matter ($PM_{2.5}$) annual standard of 15.0 ug/m^3 was met at the Champaign monitoring station in 2000, 2001, 2002, and 2003. The average annual particulate matter concentrations for these years were 14.8, 12.6, 12.2, and 13.1 respectively (IEPA 2004). In 2005, the Air Quality Index (AQI) was rated as "Good" 88% of the time (AQI = 0 to 50) and "Moderate" 11% of the time (AQI = 51 to 100) while there was only one day with an AQI of 112 in the "Unhealthy for Sensitive Groups" range (AQI = 101 to 150) (EPA 2005a). The mean annual AQI was 32. The primary AQI pollutant was ozone on 186 days and $PM_{2.5}$ on 26 days.

Champaign County is not designated as a U.S. Environmental Protection Agency (EPA) nonattainment area for any primary air pollutants. The EPA Class 1 nonattainment areas in Illinois are the counties south and west of Chicago and east of St. Louis, Missouri (EPA 2005b).

WIND ROSE PLOT

Station #93822 - SPRINGFIELD/CAPITAL ARPT, IL



Wind Speed (Knots)			
> 21	MODELER	DATE **10/7/2004**	COMPANY NAME **Illinois State Climatologist Office**
17 - 21	DISPLAY **Wind Speed**	UNIT **Knots**	COMMENTS **1961-1990 Annual Average**
11 - 16	AVG. WIND SPEED **9.61 Knots**	CALM WINDS **2.52%**	
7 - 10			
4 - 6	ORIENTATION **Direction (blowing from)**	PLOT YEAR-DATE-TIME **Jan 1 - Dec 31 Midnight - 11 PM**	PROJECT/PLOT NO.
1 - 3			

WRPLOT View 3.5 by Lakes Environmental Software - www.lakes-environmental.com

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FIGURE 4-1
WIND ROSE
SPRINGFIELD/CAPITAL AIRPORT, ILLINOIS

C05

4.4 GEOLOGY

The following paragraphs were reproduced from the site Safety Analysis Report (SAR) (University of Illinois 1971) without the accompanying figures.

The bedrock formations in east-central Illinois are layers of sandstone, limestone, dolomite, shale, and coal arranged one above the other like pages of a book. The rock layers, now firm and compact, were originally deposited as unconsolidated sediments in costal marshes or in shallow seas, repeatedly invaded the continent. These sedimentary layers hardened into rock and were later warped and tilted so that today they are no longer horizontal.

After the Pennsylvanian sediments were deposited, the seas retreated and the upper part of the bedrock was deeply eroded. One of the major valleys of the old drainage system extends across the central part of the state from the Indiana border to the Illinois River Valley. The valley takes its name from the village of Mahomet located over the deepest part of the channel in Champaign County. The valley of the Mahomet River and its tributaries, and much of the ancient Mississippi Valley, were completely buried by the deposits of the great glaciers which covered much of Illinois during the Pleistocene epoch.

Glacial ice sheets which advanced outward from centers of snow accumulation in Canada transported a great volume of rock debris. As the ice sheets melted, the rock debris was deposited as an irregular blanket which covered the eroded, layered bedrock. Till, outwash, and the sediments of modern streams now cover the bedrock surface of east-central Illinois, resulting in a relatively level plain. This plain is broken by broad ridges (moraines) which were deposited along the ice front in a roughly concentric pattern.

Three major glaciations, Kansas, Illinois, and Wisconsin, have affected the Champaign-Urbana area. Each glaciation was followed by an interglacial period during which the climate warmed and the ice front melted back.

4.5 SEISMOLOGY

The following six paragraphs were reproduced from the site SAR (University of Illinois 1971) without the accompanying figures. The last two paragraphs provide information on earthquake activity reported by the U.S. Geological Survey (USGS) not included in the SAR discussion.

There is ample evidence that the Champaign-Urbana area has not been severely affected by any earth tremors for the past 350 years. There have been two tremors having epicenters on or near the fault line about 140 miles north of Champaign-Urbana which has affected the area only slightly.

The first tremor, which occurred in 1909, had its center near the Illinois-Wisconsin border and affected the Champaign-Urbana area with an intensity five shock wave on the Rossi-Forell scale. A wave of that intensity is felt by almost everyone, and it may rattle or crack window glass or crack plaster walls. In some cases, pendulum clocks may stop. It is not severe enough to crack foundations, however.

The second tremor, which occurred in 1912, had its center between the towns of Morris and Aurora, Illinois. The tremor was less severe than the 1909 tremor, affecting the Champaign-Urbana area with an intensity three shock wave. This intensity tremor is barely perceptible to anyone on the ground and feels like vibrations from a passing truck; however, it may be felt more easily by people on the upper floors of tall buildings. Seismologists presently believe the fault line on which these two tremors were centered to be quite old and quite stable. There is little reason to suspect that the fault should cause any tremors more severe than those occurring in 1909 and 1912. In all probability, if another tremor should occur, it would be less severe than those described.

Tremors which occurred in 1811-1812 and were centered near New Madrid, Missouri (approximately 300 miles southwest of Champaign-Urbana) and which covered two million square miles, undoubtedly affected the Champaign-Urbana area; however, no data are available to show just how severely the area was affected. It is doubtful that Champaign-Urbana was affected more severely by these tremors than by either of the two tremors previously described.

Another tremor occurring in 1895 in Charleston, Missouri (about 200 miles southwest of Champaign-Urbana) covered approximately one million square miles and must have also undoubtedly affected the Champaign-Urbana area. Records do not show that any damage was noted.

There are many fault lines in the southwestern portion of Illinois. Earth tremors are frequently centered in this area, though none has been severe enough to affect Champaign-Urbana.

The USGS reports one significant earthquake not discussed in the previous paragraphs (USGS 2003a). An earthquake of intensity VII centered near Dale, Illinois occurred on November 9, 1968. A magnitude 5.3 shock was felt over 580,000 square miles in 23 states. This was the strongest felt earthquake in southern Illinois since the 1895 Missouri event. Property damage in the area consisted mainly of fallen bricks from chimneys, broken windows, toppled television aerials, and cracked or fallen plaster. Most buildings that sustained damage to chimneys were 30 to 50 years old. A large two-story brick house near Dale, Illinois, sustained several thousand dollars damage. About 10 kilometers west of Dale, near Tuckers Corners, a concrete and brick cistern collapsed. A large amount of masonry damage occurred at the City Building at Henderson, Kentucky, 80 kilometers east-southeast of the epicenter. Moderate damage to chimneys and walls occurred in several towns in south-central Illinois, southwest Indiana, and northwest Kentucky. People in multistory buildings in Boston, Massachusetts and southern Ontario, Canada, felt the earthquake.

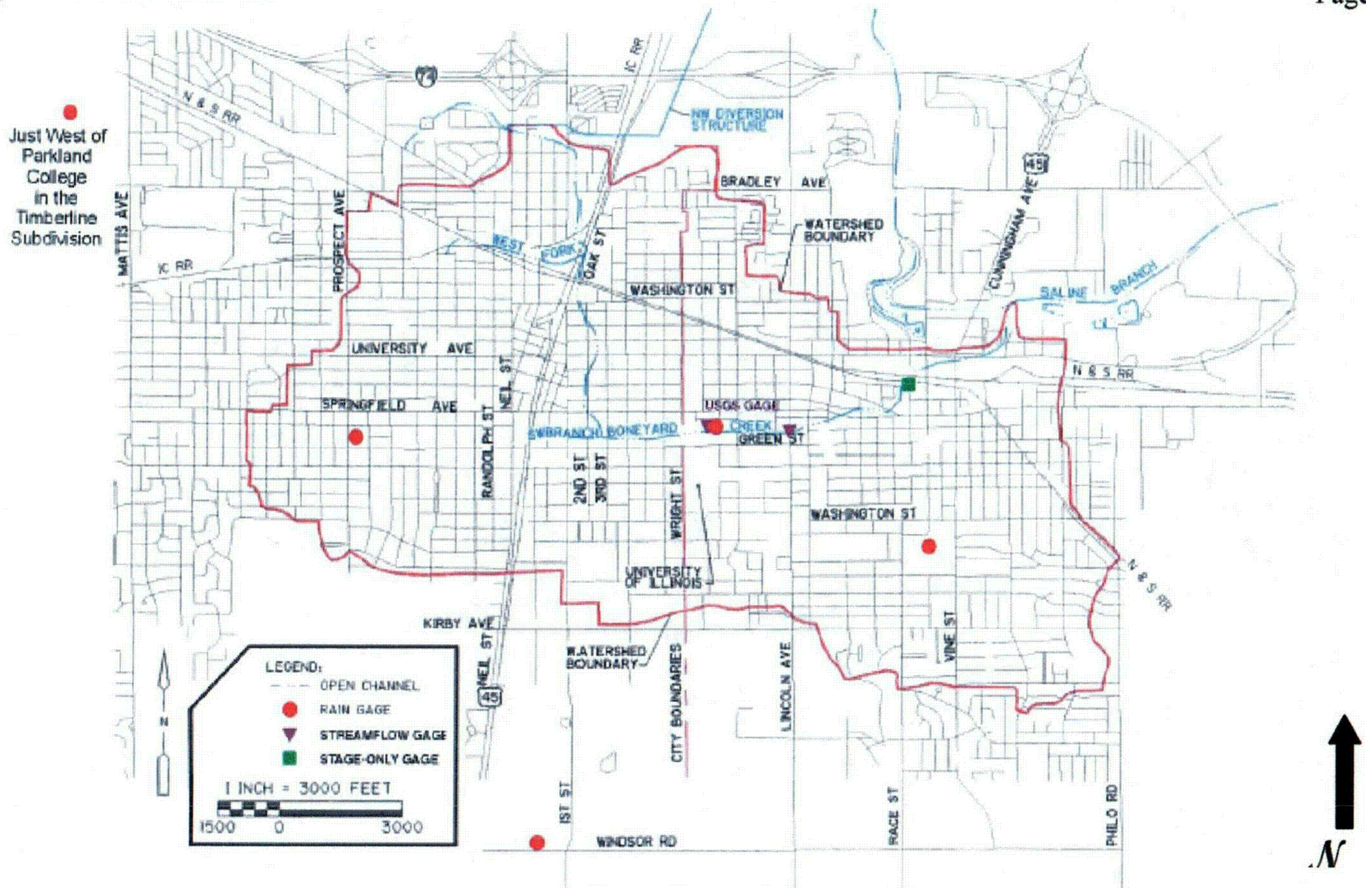
The USGS reports that three other earth quakes have occurred in Illinois since 1971 (USGS 2003a). These quakes ranged from 4.0 to 4.3 in magnitude and were located in Lee County in northern Illinois and Wabash County and Saline County in southeastern Illinois. The USGS places the Champaign-Urbana area is in a low seismic hazard region (USGS 2003b).

4.6 SURFACE WATER HYDROLOGY

The terrain in the vicinity of Champaign-Urbana varies from generally flat to very gently rolling. The soil is primarily silty loam, though in some areas it is silty clay. The general flow of runoff is to the south to Boneyard Creek. Boneyard Creek is located about 300 feet south of the NRL. The immediate location surrounding the NRL building, mostly paved areas, slopes to the south and surface runoff is collected by several grated storm sewer openings. The storm sewers of the University and the City of Urbana flow into Boneyard Creek. The surface runoff runs directly into the storm sewer system via grated openings to the southeast of the building.

Technically, Boneyard Creek is a tributary of the Saline Branch of the Salt Fork River. It was the main drainage artery serving the Boneyard Basin, a basin of nearly 5,000 acres. It is one of six watersheds within the City of Champaign. The 6.43-square mile Boneyard Creek watershed, shown in Figure 4-2, includes the oldest parts of the city, a business area adjacent to the University, and part of the University campus.

The Boneyard has experienced overbank and drainage system backwater flooding for decades. An extensive storm sewer system drains much of the Boneyard Creek watershed. In the late 1950s and early 1960s, the Boneyard Creek Channel was lined with sheet piling walls and concrete floor in Urbana near Rose Street to the upstream side of Lincoln Avenue. In addition, the Northwest Diversion Structure was completed in 1960, resulting in the upper 0.88 square miles of the watershed being diverted directly to the Saline Drainage Ditch. (USGS 2004)



Boneyard Creek Watershed (figure adapted from Berns, Clancy, and Associates Construction Permit Application Report, April 30, 1998)

FIGURE 4-2
BONEYARD CREEK WATERSHED
(USGS 2005)

C06

In the last 10 years, a significant effort has been underway to improve the flow of the Boneyard. These improvements included open channel and detention improvements through the University campus and other areas. The Healy Street Detention and Doner Drive basins were constructed to reduce flow to the creek. This Haley Street Basin, located upstream from the University campus, can reduce the discharge flows from approximately 1,000 cubic feet per second (cfs) to 60 cfs to the Boneyard during intense storms (Daily & Associates 2004). Because of the improvements completed by the City of Champaign and the University, the 100-year flood plain has been reduced, particularly upstream of Lincoln Avenue. The proposed construction of two additional drainage basins in Champaign will provide 100-year storm storage of 62-acre-feet along Second Street between Springfield and University Avenues (City of Champaign 2005).

4.7 GROUNDWATER HYDROLOGY

Ground water provides approximately one-third of Illinois' population with drinking water. The sources of this water can be broken down into three major units: 1) sand and gravel, 2) shallow bedrock, and 3) deep bedrock. The principal aquifers of Illinois are shown in Figure 2 of the Background chapter of this volume. The majority of the groundwater resources are centered in the northern two-thirds of Illinois. Sand- and-gravel aquifers are found along many of the major rivers and streams across the state and also within "buried bed-rock valley" systems. The buried bedrock valleys were created by the complex glacial and interglacial episodes of surface erosion in Illinois. There are also many instances of thin sand-and-gravel deposits within the unconsolidated materials above bedrock. These thin deposits are used throughout Illinois to supply small towns with their water requirements. The shallow bedrock units are more commonly used in the northern third of Illinois, whereas the deep bedrock units are most widely used in the northeastern quarter of Illinois (in and around the Chicago area). The use of these waters is highly variable throughout the state with respect to quantity and use classification. (IDNR 1994)

In Champaign County the fill of the Mohomet Valley contains sand and gravel deposits suitable for development of high-capacity wells. The Mahomet Bedrock Valley aquifer extends from the Illinois/Indiana border to the Illinois River. Near Champaign, the aquifer is buried beneath some fine grained glacial material that provides protection. The aquifer in Champaign County is less vulnerable to contamination than in the other two counties. The water for the towns of Champaign and Urbana is obtained from wells that penetrate the Mahomet Bedrock Valley aquifer. The bulk of the local water supply is provided by 4 to 6 wells that are approximately 300 feet deep. Other local supply wells that use a shallower aquifer are 120 to 150 feet deep. (IDNR 1994)

4.8 NATURAL RESOURCES

According to the Illinois State Geological Survey and the USGS there is one nonfuel-mineral production facility, a construction sand and gravel facility, located in southern Champaign County approximately 15 miles southeast of Champaign (USGS 2003c).

As described in Section 8.0, groundwater resources provide the drinking water to the populations of Champaign and Urbana. Therefore, these resources are to be protected from contamination.

There are three wildlife preserves in Champaign County. These are described in Section 4.9.

The agricultural resources of Champaign County are significant. Table 4-1 summarizes the 2002 agricultural resources for the county as provided by the U.S Department of Agriculture (USDA).

TABLE 4-1
AGRICULTURAL RESOURCES OF CHAMPAIGN COUNTY (USDA 2002)

Item	Quantity
Number of Farms	1,285
Land in farms	577,066 acres
Estimated market value of land and buildings	$1,339,210
Estimated market value of machinery/equipment	$132,373/farm
Average size of farm	449 acres
Number of farms > 1,000 acres	169
Harvested cropland	543,827 acres
Irrigated land	5,049 acres
Total income from farm-related sources	$5,500,000
Selected livestock and poultry:	
Cattle (beef and milk inventory)	5,062
Hogs (inventory)	5,704
Chickens (Layers inventory)	3,772
Chickens (Broilers sold)	412
Selected crops harvested:	
Corn for grain	277,527 acres
Corn for silage or greenchop	165 acres
Wheat (winter for grain)	2,405 acres
Oats for grain	30,143 acres

4.9 ECOLOGY AND ENDANGERED SPECIES

The flora and fauna in the immediate vicinity of the NRL are typical of urban and suburban settings. The site does not support any wildlife populations because of its small size and the highly-developed area lacks suitable habitats. The trees and shrubbery around the building in the area support a small population of animals such as rodents, insects, and birds. There is some aquatic life in Boneyard Creek.

The following paragraph and Section 4.9.1 through 4.9.4, which describe the ecology of Champaign County, are reproduced from the web page "Wildlife Preserve in Champaign County" (ESLARP 2005).

Currently, there are three significant forest preserves in Champaign County. Wildlife preserves are habitats set aside for the preservation and protection of native animal and plant species, with special attention paid to threatened and endangered species. These sites provide recreational opportunities for hiking, fishing, hunting, camping, nature watching, and canoeing. Lake of the Woods County Park, located near Mahomet, Illinois, runs along the Sangamon River. It occupies 900 acres of rolling woodlands. Middle Fork River Preserve occupies 1,530 acres of forest and grassy meadows. It provides areas for recreation, as well as the protection of wildlife. The Waterfowl Management area, is of significant importance at this site. The last preserve is the Salt Fork Forest Preserve, which is approximately 800 acres of recreation and wildlife habitat. The demand on recreational facilities has skyrocketed in recent years. In order to meet the demand and provide a high quality experience for users and still preserve quality habitats for wildlife, planning for another preserve will be necessary.

4.9.1 Amphibians and Reptiles

Only twelve types of amphibians reside in Champaign County area. The twelve types consist of four salamanders, six frogs, and two toads. The spotted, marbled and red-backed salamanders, have not been seen since the 1960s. These animals lived in the woodland areas, until they were timbered. The five most commonly seen amphibians are the dwarf American toad, small-mouthed salamander, Blanchard's cricket frog, upland chorus frog, northern leopard frog, and bullfrog.

There are twenty-two species of reptiles living in Champaign County. The twenty-two species are made up of, thirteen snakes, one lizard, and eight turtles. The broad-banded water snake, eastern plains garter snake, common snapping turtle, and eastern spiny soft shell turtle, are the most common reptiles found in Champaign County. The reason why these reptiles have survived, is the fact that they have adapted to their changing environment. They seem to thrive in vacant lots or anywhere where there are piles of trash. However, several are killed by man each year.

Amphibian and reptile populations are under a great amount of pressure, due to the fact that they are being reduced at a substantial rate. This is happening because of pollution and the destruction of their habitats. This factor has threatened four reptiles, which are the prairie kingsnake, eastern milk snake, graham's water snake, and Blanding's turtle.

4.9.2 Birds

There are 270 or more species of birds in the Champaign County area. Only around 45 of these birds stay year round. There are 115 species that nest in this area. However, in the last 165 years, it has been noticed that several of these birds are becoming extinct. The greater prairie-chicken, although not extinct has not been seen for several years.

Furthermore, other species are being reduced in numbers, due to the destruction of their natural habitats. Hawks and owls have been main targets, because it is thought that they kill farmer's chickens. This reoccurring event has drastically reduced the red-tailed hawk and the barred owl.

Several years past, it was common for people to kill and sell songbirds, either for food or money. Fortunately, these killing attacks have stopped, because they are now protected by law. Some types of birds are more in danger from the destruction of their natural habitats, then being hunted by man. The destruction of prairie and marshes has put suffering on rails, ducks, shorebirds, sparrows, wrens, owls, and hawks. Also, the destruction of woodland areas has depleted numbers of the black-crowned night heron, cooper's hawk, barred owl, hairy woodpecker, red-headed woodpecker, scarlet tanager, and Carolina wren. They use the woodlands for nesting purposes. However, the principal source of the depleting bird inhabitants, is the growing human population.

4.9.3 Fish

Over 90 species of fish live in Champaign County. However, some fish that were here before the land was settled have become extinct, due to the draining of marshes. However, several species have been introduced into this area, which has increased the fish population immensely. The important groups of fish are the minnows, suckers, sunfishes, bass, darters, catfishes and buffalo fishes. The other groups only contain a few fish.

The changing of stream characteristics has reduced the fish population. The cultivation of farmland causes erosion, which increases the muddiness in streams. This factor has caused fish populations to decrease. Pollution is another factor. Other fish have decreased in number, due to the destruction or changing of their original environments. However, several species of fish still survive. The most common fish found in Champaign County, are the following: northern hogsucker, green sunfish, bluegill, white crappie, and flathead catfish.

4.9.4 Mammals

There are 40 species of wild mammals that reside in the Champaign County area. They range from the tiny mole to the large coyote. There is a chance that there once lived buffalo or bison, bear, mountain lion, and otter in this area. The house mouse was introduced by man several years ago. The white-tailed deer and the beaver have been missing for many years. They have chosen to live in more suitable areas of Illinois.

The mammals that are common to the Champaign County area are the opossum, deer mouse, house mouse, prairie vole, muskrat, Franklin's ground squirrel, woodchuck, gray squirrel, fox squirrel, skunk, raccoon, red fox, and cottontail [rabbit]. Although the species of mammals in this area are not in immediate danger, the badger, least weasel, gray fox, coyote, flying squirrel, and chipmunk, are becoming rare. The destruction of their natural habitats, along with illegal killing, is leading to the extinction of these species.

4.9.5 Endangered and Threatened Species

Table 4-2 shows the number of each endangered or threatened species in Illinois. The notes provide the names of the federally endangered species.

TABLE 4-2
NUMBER OF ENDANGERED OR THREATENED FLORA AND FAUNA IN ILLINOIS (IDNR 2004a, 2004b)

Classification	Illinois Endangered Species	Federally Endangered Species	Illinois Threatened Species	Federally Threatened Species
Fishes	18	1 [a]	13	0
Amphibians & Reptiles	11	0	13	0
Birds	24	2 [b]	8	0
Mammals	5	2 [c]	4	1
Invertebrates	35	8 [d]	13	0
Flora	236	1 [e]	64	6

Notes:
a Pallid Sturgeon
b Piping Plover and Least Tern
c Gray bat and Indiana Bat
d Iowa Pleistocene Snail, Orange-foot Pimpleback, Clebshell, Fanshell, Fat Pocketbook, Higgins Eye, Hein's Emerald Dragonfly, and Karner Blue Butterfly
e Leafy Prairie Clover (Delea folioas)

4.10 TRANSPORTATION AND NOISE

The NRL facility is located in the center of a large university campus and urban area. As such there are many transportation routes in the immediate area. As described in Section 3.0, there are two major interstate highways, I-74 and I-57 within four-miles of the facility. There are also several main rail lines that pass near the site including CSX Transportation and Illinois Central lines.

Within the NRL facility area, vehicular traffic, construction, and heating, ventilation, and air conditioning equipment generate the ambient noise environment.

5.0 POTENTIAL ENVIRONMENTAL CONSEQUENCES

The potential environmental consequences of the proposed NRL facility D&D are primarily related to human health effects, waste disposal, noise, and air quality. These four impacts are discussed in the following sections. A list of Federal regulations that are applicable to the facility D&D is also provided.

There are expected consequences on ecological or natural resources as a result of the NRL facility D&D activities.

5.1 HUMAN HEALTH EFFECTS

Generally, the activities involved in the D&D of a facility such as the NRL could involve the following hazards for site workers and, to a much lesser degree, the general public:

- External radiation exposures
- Dermal contact with radioactive and hazardous materials
- Inhalation of hazardous or radioactive materials
- Exposures to nuisance dust

Potential hazards limited to the on-site D&D work forces include:

- Possible confined spaces
- Entry into trenches and excavations
- Heavy equipment movement dangers
- Electrical shock
- Falling debris and rubble
- Working in hot and cold environments

Hazards will be controlled by operating according to license requirements, applicable state and federal regulations, and approved safety procedures. Specifically, the following will be used to minimize the risk to the work force and/or the public from identified hazards:

- Radiation Work Permits (RWP) and approved work procedures
- Air filtering and monitoring procedures
- Site-Specific Health and Safety Plan (SSHASP)
- 10 CFR 20 radiation protection requirements to protect workers and the general public
- 29 CFR 1910.120 requirements for occupational safety
- 29 FR 1910.146 confined space entry procedures

Radiation exposures to site workers will be monitored such that the occupational exposure limits are not exceeded. The potential exposures to the public as a result of D&D activities and radioactive waste shipments are estimated to be negligible according to the dose estimates given for the "reference research reactor" in NUREG-

0586, "Final Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Facilities" (NRC 1988). The dose given in the GEIS for D&D and truck transportation is given as "negligible (less than 0.1 person-rem)."

The anticipated exposures to the public after D&D of the facility are also expected to be negligible. The site will be released for unrestricted use based on the criteria that the dose to a member of the target population will not exceed 25 mrem per year. National average background radiation dose is approximately 300 mrem per year.

The transportation of the waste materials from the site possibly provides the greatest human health risk associated with the project simply due to the increase in truck traffic in an area with a high traffic density and high pedestrian population. The primary impacts are accident risk and increased emission and noise from the vehicles, not the radiation exposure from the material being transported.

Traffic circulation and parking around the facility may be slightly impacted at times due to the location of the facility adjacent to a parking lot with limited access.

5.2 WASTE DISPOSAL

Small amounts of non-radioactive hazardous (asbestos containing material, lead, etc.) may be staged in an acceptable satellite accumulation area on-site for up to 90 days. Prior to the end of the accumulation period, a license contractor will transfer the waste to an authorized off-site disposal or recycling facility.

Low-level radioactive waste (LLRW) will be packaged in accordance with the designated waste processor or disposal facility site waste acceptance criteria. The waste will be classified according to the waste classifications in 10 CFR 61.55 as either Class A, B or C waste. Class A waste or unclassified LLRW will be disposed of at the Envirocare of Utah (Envirocare) disposal site in Clive, Utah. It is estimated that there will be between 10,000 and 15,000 cubic feet of LLRW shipped to Envirocare in "strong tight" containers that conform to all applicable U.S. Department of Transportation regulations.

There is not expected to be any Class B or Class C waste generated during the D&D activities (Scientech 2005). Class B and C waste, if encountered, will be disposed of at the Chem-Nuclear disposal site in Barnwell, SC.

There may be a small volume of mixed waste, primarily in the form of activated lead, that will be disposed of at Envirocare.

5.3 NOISE

As with any demolition project, a significant amount of noise will be generated. Noise will be generated indoors by equipment such as jackhammers, scabblers, and concrete saws. Backhoes and other heavy equipment will be used for outdoor work and potentially some indoor work.

On-site workers will be provided with hearing protection as required by the SSHASP.

Because most D&D activities will take place during normal business hours, noise may impact occupants of academic buildings around the facility.

5.4 AIR QUALITY

Several D&D activities could minimally impact air quality as a result of mobile and stationary source emissions. Small increases in the amounts of mobile source emissions, such as carbon monoxide and nitrogen oxides, could be released from contractor equipment, trucks, and vehicles and waste transportation vehicles. However, because of the small number of sources, the total impact of the overall air quality would be low.

Releases resulting from the demolition or decontamination of contaminated materials will occur within the NRL facility. The air from the facility will be filtered using high-efficiency particulate air (HEPA) filters prior to leaving the facility through the facility stack. Temporary containment structures with additional air handling and filtration units may also be used inside the facility.

Standard asbestos abatement procedures will be implemented by a licensed contractor for asbestos removal.

Releases from stationary sources will also include dust from final demolition activities involving uncontaminated construction materials. However, due to the size of the facility, there would be no impacts on the regional air quality.

5.5 REGULATORY ISSUES

Table 5-1 provides the applicability of various Federal regulations for the proposed action.

TABLE 5-1
APPLICABILITY OF ENVIRONMENTAL STATUES AND REGULATIONS

Statue/Regulation	Evaluation	Applicability
National Environmental Policy Act (NEPA)	The evaluation is contained in this document	Yes
Endangered Species Act	No endangered or threatened species will be impacted	No
Floodplain/Wetland Regulations	The proposed action is not with a flood plain of wetland	No
Fish and Wildlife Coordination Act	The proposed action would not impact resources	No
Farmland Protection Act	The proposed action would not impact resources	No
National Historic Preservation Act	The proposed action would not impact resources	No
American Indian Religious Freedom Act	No Native American populations will be impacted	No
Wild and Scenic Rivers Act	The proposed action would not impact resources	No
Resource and Conservation, Recovery Act (RCRA)	The proposed action may include the generation of RCRA wastes	Yes
Comprehensive Environmental Response, Compensation and Liability Act (CERCLA)	Releases of hazardous materials would be performed under CERCLA	Yes
Federal Insecticide, Fungicide, and Rodenticide Act (FIFRA)	The proposed action would not involve the use of these substances	No
Toxic Substance Control Act (TSCA)	Asbestos will be packaged and disposed of in accordance with TSCA	Yes
Clean Air Act (CAA)	Airborne asbestos will be contained in accordance with the CAA	Yes
Clean Water Act and Drinking Water Act	The proposed action would not impact resources	No
Noise Control Act		
Hazardous Materials Transportation Act	Hazardous and radioactive materials will be shipped from the site	Yes
National Emissions Standards for Hazardous Air Pollutants (NESHAPS)	The EPA has stated that NESHAPS are applicable to NRC-licenses facilities	Yes
Atomic Energy Act	Compliance with environmental and worker protection standards	Yes

5.6 AREAS NOT AFFECTED

The proposed action would not affect the following areas:

- Population and Land Use
- Natural Resources
- Surface and Groundwater Hydrology
- Cultural Resources

5.7 CUMMULATIVE EFFECTS

Due to the size of the NRL facility, no significant cumulative effects are expected from the proposed action as summarized below:

- **Human Health** – Worker exposures and safety would be controlled by complying with all applicable safety regulations. Additionally, all radiation dose will be kept as low as reasonably achievable. There are expected to be no measurable off-site health effects.
- **Waste Generation** – Hazardous, LLRW, and clean waste will be managed and disposed of according to all local, state, and federal regulations.
- **Cultural Resources** – Not impacted.
- **Population and Land Use** – Not impacted.
- **Noise** – There would be some increase in the off-site background noise during some D&D activities.
- **Traffic** – There would be a temporary increase in local traffic from the contract workers and waste shipments. This results in a slight increase in risk to the general public in the high-traffic and high-pedestrian urban environment.
- **Geology, Seismology, Hydrology** – Not impacted.
- **Regional Air Quality** – Not significantly impacted.
- **Ecology and Natural Resources** – Not impacted.

5.8 ALTERNATIVES TO THE PROPOSED ACTION

Alternative 1 – No Action: The no-action alternative is not allowed under current NRC regulations. However, if allowed by some regulatory variance, significant site surveillance would be required.

Alternative 2 – Decontamination: The activities involving the management of hazardous and radioactive materials required for this alternative would not vary significantly from the proposed alternative. However, the final status survey and site release methods would be more difficult. This alternative would also result in a greater risk of undirected radioactive contamination being left behind.

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DECOMMISSIONING PLAN

NUCLEAR RESEARCH LABORATORY
UNIVERSITY OF ILLINOIS AT CHAMPAIGN-URBANA
U.S. NUCLEAR REGULATORY COMMISSION FACILITY
OPERATING LICENSE NO. R-115

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REVISION LOG

Revision Number	Affected Pages	CRA Number	Approval

ACRONYMS AND ABBREVIATIONS

ACM	asbestos-containing material
ALARA	as low as reasonably achievable
ALI	Annual Limit on Intake
C&D	construction and demolition
CAM	continuous air monitor
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CHP	Certified Health Physicist
Ci	curies
DAC	derived air concentration
DCGL	derived concentration guideline level
D&D	decontamination and decommissioning
DDE	deep-dose equivalent
DOC	decommissioning operations contractor
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DP	decommissioning plan
dpm/100cm^2	disintegrations per minute per 100 square centimeters
DQO	data quality objective
ER	Environmental Report
FSS	final status survey
FSSP	Final Status Survey Plan
FSSR	Final Status Survey Report
HAZWOPER	Hazardous Waste Operations and Emergency Responses
HEPA	high-efficiency particulate air
HSA	Historical Site Assessment
HVAC	heating, ventilation, and air conditioning
ISO	International Organization for Standardization
kg	kilogram
kW	kilowatts
LLRW	low-level radioactive waste
LOPRA	Low Power Reactor Assembly
LSC	liquid scintillation counter
m^3	cubic meters
MARSSIM	*Multi-Agency Radiation Survey and Site Investigation Manual*
ml	milliliters
mrem/yr	millirem per year
MWhrs	metawatt-hours
NaI	sodium iodide
NEPA	National Environmental Policy Act
NRC	U.S. Nuclear Regulatory Commission
NRL	Nuclear Research Laboratory

ACRONYMS AND ABBREVIATIONS (CONTINUED)

NUREG	NRC technical report designation (Nuclear Regulatory Commission)
ODC	other direct costs
OSHA	Occupational Safety and Health Act
OSL	optically-stimulated luminescent (dosimeter)
pCi/g	picocurie per gram
PPE	personal protective equipment
QA	quality assurance
QAPP	quality assurance project plan
RA-BE	radium-beryelium
rem	roentgen-equivalent man
RSO	Radiation Safety Officer
SHASP	Site Health and Safety Plan
SHSO	Site Health and Safety Officer
TLD	thermoluminescent dosimeter
TRIGA	Teaching Research Isotope General Atomic
University	University of Illinois at Urbana-Champaign

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1.0 SUMMARY OF DECOMMISSIONING PLAN

1.1 INTRODUCTION

The University of Illinois' (University) Nuclear Research Laboratory (NRL) contains the University's Advanced Teaching Research Isotope General Atomic (TRIGA) Mark II nuclear research reactor, manufactured by the General Atomic Division of General Dynamics Corporation. The NRL is located on about 5,000 square feet on the campus of the University of Illinois at Urbana-Champaign in the City of Urbana, Illinois. The University campus is located in the adjoining cities of Urbana and Champaign and is centered on the dividing line of these cities. The University is about 110 miles south-west of Lake Michigan and about 35 miles from the Illinois-Indiana border. Figures A-1 and A-2 showing the physical location of the NRL facility are provided in Appendix A. Figures A-3 and A-4 are photographs of the facility.

This Decommissioning Plan (DP) was prepared in accordance with Chapter 17 of the Nuclear Regulatory Commission (NUREG)-1537, Part 1, "Guidance for Preparing and Reviewing Applications for Licensing of Non-Power Reactors" (NRC 1996). This DP provides guidance on the general process and methods that will be used to decontaminate and/or remove radioactive materials, equipment, components, systems, and soil from the NRL facility in a safe manner. The DP also describes the general deconstruction process, that will result in the complete removal of the NRL structure from the site location allowing an unrestricted release by the NRC. The final status survey process that will be implemented to demonstrate compliance with the derived concentration guideline levels (DCGL) and to support the unrestricted release and license termination is also described.

1.2 BACKGROUND

Construction began on the NRL in the summer of 1959 to house the training and research nuclear reactor. The construction of the building and installation of the reactor was overseen by the University. By 1960, the walls of the NRL were complete and the foundation for the reactor and bioshield and the thermal column trench were complete. By the spring of 1960, the reactor and reactor tank were installed along with the beam ports and the forms for bioshield concrete. The NRL was completed in the summer of 1960 and the reactor first went critical on August 16, 1960. The reactor was operated under NRC Facility Operating License No. R-115.

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

In 1967, the University decided to upgrade the reactor to utilize the most recent design characteristics of the TRIGA fuel and to install a new forced circulation cooling system. The original core was also replaced with a new core that was also light-water-cooled, graphite reflected, and contained uranium-zirconium hydride fuel-moderator elements with stainless steel cladding. The fuel elements in the new core, however, were positioned in a hexagonal lattice. The new reactor license permitted steady state operation to 1.5 MW and pulsing to 6000 MW.

The Bulk Shielding Tank, located on the south side of the reactor, allowed for neutron beam experiments to be conducted underwater for additional shielding. The Bulk Shielding Tank was also the home of the Low Power Reactor Assembly (LOPRA) which was a subcritical assembly that used TRIGA fuel. The LOPRA operated under its own NRC license (No. R-117) beginning in 1971. The R-117 license governed the use of the LOPRA until 1995 when the fuel and subcritical assembly were transferred to the NRL's current R-115 license and the R-117 license was terminated.

On August 6, 1998, nearly 30 years since its initial start-up and after 11,566.7 megawatt-hours (MWhrs) of operation, the NRL TRIGA reactor was shut down permanently. In 1999, the reactor was officially placed in a SAFSTOR condition while waiting for arrangements to be made to remove and ship the reactor fuel. The Bulk Shielding Tank was used for wet storage of the fuel following shutdown. On August 18, 2004, the reactor fuel was removed and shipped to the U.S. Department of Energy's (DOE) Idaho National Laboratory.

The current status of the NRL facility is described in the Historical Site Assessment (HSA) (Scientech 2005a) and the Site Characterization Report (Scientech 2005b). The facility is being managed according to current license conditions and technical specifications.

The NRL building is a steel frame concrete block building that is approximately 80 feet east-west by 45 feet north-south. The building is supported by 30 metal-shell, cast-in-place concrete piles with minimum lengths of 40.5 feet. A 6.5-foot deep by 1-foot wide concrete footing, which is laid on concrete pile caps, supports the walls. Figures showing the physical layout of the NRL facility and many of the reactor systems and components are provided in Appendix A. The figures in Appendix A are reproduced from the facility Safety Analysis Report (University of Illinois 1967)

The interior of the building contains three levels: the mezzanine level, (Figure A-5) the storage level (located above the mezzanine) (Figure A-6), and the lower level (reactor room) (Figure A-7). The mezzanine level is 10 feet above the reactor room floor and the storage level is 21 feet above the reactor room floor. The mezzanine level contains office space, the former control room, and two restrooms. The storage level is located above the mezzanine level and contains one office and storage space. The mezzanine floor, storage floor, and roof are placed on standard bar joists which are tied to horizontal I-beams and the main support columns. The mezzanine and storage floors are 2.5-inch concrete slabs poured on corrugated steel plate. Just south of the mezzanine is a small loading bay which contains four dry fuel storage tubes in the floor (Figure A-5).

The reactor level contains the TRIGA reactor, a radioactive materials storage cage, the Mechanical Equipment Room, access to the primary coolant water pipe tunnel and workshop/experimental areas. The reactor level is about 44 feet wide by about 80 feet long. The floor is a six inch concrete slab laid on undisturbed earth. A special two-foot-thick concrete base is used to support the reactor and the thermal column door railway. Ten piles similar to those used to carry the building are used to support this special base (Figure A-8).

The Mechanical Equipment Room containing the heat exchanger and the primary and secondary cooling system pumps is located off the south side of the building and is accessed through a door on the reactor room level. The exterior dimensions of the room are approximately 18 feet by 22 feet. The exterior height is less than 6 feet above grade. Located below the Mechanical Equipment Room is a vault containing two nitrogen-16 delay tanks with capacities of 1,500 gallons and 3,000 gallons. Access to the vault is through the Mechanical Equipment Room floor (Figures A-8 and A-9)

There is a tunnel that contains the primary coolant pipes that run from the bottom of the reactor tank to the vault containing the nitrogen-16 tanks (Figures A-8, A-11, A-13, A-14). The tunnel has a poured concrete floor about 6 inches thick and filled concrete block walls. The tunnel is 3 feet wide and has a clearance of about 5 feet. The length of the tunnel from the vault to below the reactor is about 20 feet. Access to the tunnel is through the reactor room floor near the southeastern corner of the bioshield (Figures A-8 and A-10).

The clearance from the reactor room floor to the roof supports is 35 feet except under the mezzanine level. The area under the mezzanine contains the radioactive materials storage cage, a small shower room, workshop areas, and a small mechanical room. The NRL facility roof is composed of gravel on a gypsum roof deck which is covered with four-ply asphalt paper. The roof contains the ventilation system blower and stack.

1.2.1 Reactor Decommissioning Overview

The University plans to remove all radioactive materials from the NRL facility, demolish the structure, and release the property for unrestricted use. Once released, the University will seek termination of license R-115.

Many of the reactor components and systems are either activated or contaminated (Scientech 2005b). As such, these items need to be segregated from non-radiological components so that they can be disposed of as low-level radioactive waste (LLRW). Building materials, such as parts of the concrete bioshield and floor, and the soil under the reactor are also impacted and need to be removed and disposed of according to their radiological status. The following major decommissioning tasks, which are necessary for site release, are presented below (the sequence will vary):

- Further facility characterization as needed
- Remove loose equipment
- Remove activated materials in the reactor tank
- Drain the reactor tank and associated systems
- Remove the reactor bioshield and segregate waste materials according to radioactivity levels or contaminant concentrations
- Remove primary coolant piping systems, pump, heat exchanger, and nitrogen-16 tanks
- Remove the dry fuel storage tubes
- Remove the water retention system (floor drains, piping, etc.)
- Remove the building exhaust system
- Remove contaminated concrete flooring
- Remove contaminated soil
- Remove asbestos containing materials (ACM)
- Remove uncontaminated building systems (secondary cooling system, plumbing, electrical, heating, ventilation, and air conditioning, etc.)
- Demolish of building structure
- Excavate building footing and foundation
- Ship waste
- Perform the final status survey
- Submit reports to the NRC that demonstrate compliance with the site release requirements and request license termination
- Backfill the excavation to grade

The on-site decommissioning tasks are expected to begin in the fourth quarter of 2007. The on-site activities are then expected to last about 12 months with the completion of the project by the end of 2008.

The final status survey will be developed following the guidance provided in NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" (NRC 2000). Since the complete NRL structure, concrete slabs, and foundations will be removed, the final status survey will cover only the exposed subsurface soils and the surface and subsurface soils surrounding the NRL facility.

1.2.2 Estimated Cost

The decommissioning cost estimate is summarized in Table 1-1. Tasks include the decommissioning operations contractor (DOC) costs as well as subcontractor and other direct costs (ODC). Several elements of the decommissioning cost estimate will pose an indeterminate probability of significant costs increases. The 25% contingency cost included in Table 1-1 covers costs that may result from incomplete design information, unforeseen or unpredictable conditions or uncertainties within the defined project scope. Typically these include external factors such as the cost of waste transportation (i.e., fuel surcharges), waste disposal rates, or waste volumes derived from previously inaccessible and uncharacterized areas.

The cost estimate also includes the disposal of a radium-beryllium (Ra-Be) neutron source that is currently located in the reactor tank. The University is making attempts to arrange for the disposal of this source prior to issuing a decommissioning contract. However, the University wished to include the estimated disposal cost in the total site decommissioning cost.

TABLE 1-1
DECOMMISSIONING COST ESTIMATE

Major Project Activities	Activity Cost[a]
Preparation and approval of site-specific plans and procedures	$110,982
Hazmat removal / asbestos abatement	$179,158
Site mobilization, training and orientation	$55,421[b]
Facility preparation and miscellaneous waste removal	$113,095
Reactor component removal	$29,128
Reactor tank removal and bioshield demolition	$346,546
Primary coolant system removal	$110,408
Dry fuel storage pit removal	$30,661
Floor and tunnel decontamination	$69,238
Facility radiological clearance survey	$43,692
Building & foundation demolition and removal	$124,330
Final status survey (soil sampling and analysis)	$57,479
Site grading and restoration	$83,359
Demobilization[c]	$12,826
Total Decommissioning Activities	**$1,342,172**
Travel and Per Diem Expenses[d]	$309,500
Project Management Site Visits/Audits	$29,680
Consumables and Equipment Rental	$144,732
University Oversight and Licensing	$125,000
Disposal of Ra-Be Neutron Source	$250,000
LLRW Transportation and Disposal	$1,166,394
Estimated Decommissioning Cost	**$3,367,479**
Contingency @ 25%	$841,870
Estimated Cost with Contingency	**$4,209,348**

Notes:
a Includes subcontractor, analytical, and other task-specific specific costs.
b Includes mobilization travel cost.
c Includes demobilization travel costs.
d Does not including mobilization and demobilization travel.

1.2.3 Availability of Funds

In accordance with 10 CFR 50.75 (e)(1)(iv), the University, being a State institution, will appropriate funds to complete the decommissioning of the University NRL when necessary.

1.2.4 Program Quality Assurance

The University will select a qualified DOC to assist in the decommissioning of the NRL facility. The selected contractor will be responsible for developing a Quality Assurance Project Plan (QAPP) appropriate for the decommissioning of the NRL and the final status survey. The QAPP will incorporate standard regulatory and industry measures applicable to: project planning and management; decontamination, dismantling, and demolition; and radiological surveys, sampling, and analysis necessary for decommissioning activities. The QAPP will be reviewed and approved by the Reactor Administrator.

The decommissioning project will incorporate the University's existing quality assurance (QA) program for the transportation of radioactive materials.

The goal of the QAPP is to describe QA activities related to the following:

- Project management
- Training
- Document/procedural control
- Data management
- Recordkeeping
- Radiological surveys
- Sample collection and preparation
- Sample analysis
- Audits and corrective actions

The DOC will provide a QA manager to manage and execute the QA program. The QA manager, who will report to the DOC project manager, will ensure that decommissioning activities, including those subcontracted to other contractors or off-site laboratories, meet the requirements outlined in the QAPP and other QA documents to ensure safe and proper implementation of the decommissioning project.

2.0 DECOMMISSIONING ACTIVITIES

2.1 DECOMMISSIONING ALTERNATIVES

The three alternatives considered for NRL were the no-action alternative (SAFSTOR), complete decontamination and demolition (DECON-A), complete decontamination and release of the intact structure (DECON-B). An entombment option (ENTOMB) was not considered. DECON options are recommended by the NRC for non-power reactors.

In SAFSTOR, the facility would continue to be maintained in a condition that allows it to be safety stored and decontaminated at some time in the future. The NRL has been in a SAFSTOR status since 1998 when the reactor was shut down. The University does not wish to continue with the SAFSTOR option.

In DECON-B, all radioactive materials would be removed from the facility and the facility would be released for unrestricted use. Because the decontamination and removal of contaminated systems is expected to severely impact the existing structure and the University anticipates no future use for the existing structure, DECON-B was not considered as an attractive option for the University.

In DECON-A, all radioactive materials will be removed from the site, the facility structure will be demolished, and the site will be released and restored for unrestricted future use.

2.1.1 Facility Operating History

The NRL facility operating history is thoroughly described in the HSA (Scientech 2005a). The results of the HSA suggest that much of the NRL and its components are known to be or are potentially impacted by radioactive contamination or activation. Activation is caused by neutrons changing the nature of materials such as the transformation of stable cobalt-59, present as an impurity in carbon and stainless steel, concrete, and as the primary metal in the Stellite alloy, to cobalt-60. Through material degradation, such as rusting carbon steel, activation products can be transferred as surface contamination. Sources of contamination may include leakage from sealed radioactive sources, leakage from glove boxes, leaks and spills of radioactive or contaminated liquids, and the emission of radioactive particles during experimental procedures.

Many of the aluminum, stainless steel, and carbon steel reactor components are activated along with a portion of the heavy concrete bioshield and items embedded in the bioshield. The most highly activated component is expected to be the Lazy Susan (rotary specimen rack) because of its proximity to the core and because it contains Stellite and stainless steel components. Activated components also include the beam ports and thermal column components, including the graphite. Known contaminated materials include equipment that was used to manage and transfer specimen containers that were irradiated in the reactor core or the Lazy Susan. Other contaminated materials include concrete that was exposed to reactor primary coolant water in which tritium may have built up over the years. Tritium surveys were not part of the NRL's routine contamination surveys. Routine removable contamination surveys did show, however, that there is no wide-spread contamination of other higher-energy beta or alpha-emitting radionuclides which would likely be present with tritium contamination.

2.1.2 Current Radiological Status of the Facility

The Site Characterization Report (Scientech 2005b) provides a detailed description of the current radiological status of the NRL facility. The scope of the characterization survey, conducted in July 2005, included the complete NRL facility which encompasses the reactor level, the mezzanine level, and the storage level of the reactor building. The characterization also examined the soils surrounding and underneath the reactor building.

The characterization of the NRL facility included measurements for fixed and removable alpha and beta surface activity and removable tritium surface activity. Samples of concrete from the reactor room floor, the Bulk Shielding Tank floor, the reactor bioshield, and several other areas of the facility were also collected and sampled for tritium, total beta activity, and gamma-emitting isotopes. Soil samples were additionally analyzed for tritium.

The characterization effort identified the reactor bioshield, the primary coolant pipe tunnel, and the concrete floor of the reactor room as the primary impacted areas. Large pieces of contaminated and activated equipment identified include the internal reactor components, the nitrogen-16 decay tanks, the primary coolant pipes, and the large glove box. Estimates for the volumes and masses of major radiologically impacted components of the NRL facility are provided in Table 2-1. Table 2-2 includes ancillary items that are not considered as major reactor components. These items are currently in storage or are present in an inactive state and will be managed and disposed of according to existing University procedures or transferred to another one of the University's radioactive materials licenses.

TABLE 2-1
ESTIMATED VOLUMES AND MASSES OF MAJOR RADIOLOGICALLY IMPACTED REACTOR COMPONENTS AND SYSTEMS

Component/System	Material	Volume [a] (Mass)
	Concrete (292,000 kg) [b]	
Reactor bioshield [c]	Heavy concrete	20.3 m^3 (71,700 kg)
Tunnel	Type A Concrete	5 m^3 (11,600 kg)
Bulk storage tank [d]	Heavy Concrete	9.5 m^3 (33,600 kg)
2-foot pad under reactor	Heavy Concrete	22.6 m^3 (80,000 kg)
Reactor room floor (all)[b]	Type A Concrete	39.6 m^3 (92,000 kg)
Beam catchers	Type A Concrete	1.1 m^3 (2,490 kg)
	Reactor Components (2,839 kg)	
Reactor tank	Aluminum	0.08 m^3 (212 kg)
Reactor support	Aluminum	0.012 m^3 (33 kg)
Core assembly	Aluminum	0.05 m^3 (132 kg)
Reflector	Graphite	0.21 m^3 (375 kg)
Reflector shield	Lead	0.06 m^3 (678 kg)
Rotary specimen rack	Aluminum with Stellite	0.015 m^3 (42 kg)
Header spray ring	Aluminum	0.015 m^3 (42 kg)
Reactor bridge and Control rod drive	Steel	0.17 m^3 (1,325 kg)
	Items Imbedded in Bioshield or Floor (10,100 kg)	
Beam port tubes	Aluminum and carbon steel	0.1 m^3 (771 kg) [e]
Beam port plugs	Wood, lead, steel	0.79 m^3 (3,160 kg) [f]
Shadow shields (4)	Carbon Steel	0.42 m^3 (3,270 kg)
Supports, braces, rebar, etc.	Aluminum and carbon steel	0.1 m^3 (771 kg) [e]
Dry fuel storage tubes	Carbon steel	0.09 m^3 (694 kg)
Water confinement system (pipes and 500-gallon tank)	Carbon steel	0.2 m^3 (1,400 kg)
	Primary Coolant System (5,596 kg)	
Primary coolant piping	Aluminum	0.61 m^3 (1,660 kg)
Primary pump	Stainless steel	0.06 m^3 (442 kg)
Heat exchanger	Stainless steel	0.05 m^3 (824 kg) [g]
N-16 tank (3,000 gal)	Stainless steel	0.21 m^3 (1,630 kg)
N-16 tank (1,500 gall)	Stainless steel	0.13 m^3 (1,040 kg)
	Thermal Columns (5,543 kg)	
Graphite blocks	Graphite	3.0 m^3 (5,250 kg)
Liners and curtain	Aluminum and boral	0.1 m^3 (293 kg)

Notes:

a – Estimated material volume, not volume as packaged.

b – Most or all of the reactor room floor concrete, while slightly contaminated with tritium, could possibly be acceptable table for disposal as non-LLRW.

c – The activated portion is approximately 12% of the total bioshield concrete volume.

d – Assumed 1-foot thick on south, east, and west sides.

e – Assumed all carbon steel for mass calculation.

f – Assumed an average density of 4.0 gm/cm^3 for mass calculation.

g – The volume includes only the external shell. The mass of the external shell was doubled to include internal components.

TABLE 2-2
OTHER RADIOACTIVE, CONTAMINATED, OR ACTIVATED ITEMS

Item	Volume (Mass)	Pathway
Large glove box	3 m^3 (800 kg)	Disposal
Small glove box	0.25 m^3 (70 kg)	Disposal
Water filters and resins	55 gal (300 kg) (packaged)	Disposal
Misc. contaminated/activated waste materials	1.5 m^3 (1,500 kg) (packaged)	Disposal
Experimental equipment	1.5 m^3 (2,500 kg) (packaged)	Disposal
Fuel transfer cask	0.11 m^3 (1,300 kg)	Mixed waste Disposal
Sealed radioactive sources	NA	Transfer to University broad-scope license
Cesium-contaminated lead bricks (50)	0.05 m^3 (600 kg)	Mixed-waste Disposal

2.1.3 Release Criteria for Soils

The decommissioning alternative proposed in this Decommissioning Plan includes the dismantlement of the NRL building and the removal of all the concrete slabs, footings, and foundation materials. As such, little attention is given to the minor radiological contaminants (primarily activation products) in the reactor components and structure materials as they will not be of concern in the free-release of the facility. The minor isotopes are also of little concern in assessing radiological dose under working or accident conditions and become an issue only in waste profiling. The final status surveys will be conducted on the excavated foot print of the building and, therefore, only radiological contaminants that may have impacted the soil below the reactor structure are of primary concern.

The results of the facility characterization indicate that the soil under the building may contain residual amounts of tritium and cobalt-60. Characterization of the facility also demonstrated that europium-152 was also present in many activated components along with cobalt-60 in similar concentrations. Therefore, europium-152, while not measured in soil samples during the site characterization, may also be present in the soil as a residual contaminant. Iron-55 is another primary radionuclides of concern identified during the site characterization that may be a potential soil contaminant.

Of the potential radionuclides of concern named in the previous paragraph and those discussed in the Site Characterization Report (Scientech 2005b), only cobalt-60 and europium-152 would have a significant dose impact to future site occupants. This is because these radionuclides emit high-energy gamma radiation that would pose an external radiation hazard. Other radionuclides of concern would be of greater concern to off-site receptors because of the potential groundwater exposure pathway.

For cobalt-60 and europium-152, a site-specific dose assessment was performed to develop DCGLs that would result in a maximum dose to the future site occupant of not greater than 25 millirem per year (mrem/yr). This dose assessment was performed using the RESRAD dose modeling code (Version 6.3) assuming an occupational (industrial) exposure scenario. The details of the dose assessment are provided in Appendix B of this Decommissioning Plan.

For other radionuclides of concern, the DCGLs are set equal to the NRC soil screening values for surface soils (NRC 1998). The screening values are equivalent to 25 mrem/yr as derived using the NRC's DandD screening methodology in NUREG-5512, *Residual Radioactive Contamination for Decommissioning* (NRC 1992). They were derived based on the 90th percentile of the output dose distribution for each specific radionuclide. Behavioral parameters in the DandD model were set at "Standard Man" or at the mean of the distribution for an average human.

Table 2-3 provides the DCGLs for cobalt-60, europium-152, and other primary radionuclides of concern in soil. When applying the DCGLs in Table 2-3, the sum-of-the-fractions rule applies. That is, the sum of the ratios of the radionuclide concentrations to the DCGLs must be less than 1.0.

TABLE 2-3
DCGLs FOR PRIMARY RADIONUCLIDES OF CONCERN IN SOIL

Radionuclide	**Modeled DCGL (pCi/g)**
Cobalt-60	12.9
Europium-152	27.9
Radionuclide	**Screening Value DCGL (pCi/g)**
Tritium (hydrogen-3)	110
Carbon-14	120
Iron-55	10,000
Nickel-63	2,100
Cesium-137	11
Europium-154	8.0

2.1.4 Release Criteria for Surface Contamination

Material released for reuse, recycle, or disposal as clean waste will be shown to be free of detectable surface contamination in accordance with the guidance provided by the NRC in IE Circular 81-07 (NRC 1981). The contamination monitoring using portable survey instruments will be performed with instrumentation and techniques (survey scanning speed, counting times, background radiation levels) necessary to detect 5,000 dpm/100cm^2 total and 1,000 dpm/100cm^2 removable beta/gamma contamination. Instruments should be calibrated with radiation sources having consistent energy spectrum and instrument response with the radionuclides being measured. If alpha contamination is suspected appropriate survey measurements capable of detecting 100 dpm/100cm^2 fixed and 20 dpm/100cm^2 removable alpha activity will be used.

Properly calibrated survey instruments with known efficiencies capable of measuring the radionuclides of concern will be used for these release surveys. Removable contamination wipes may be measured in a liquid scintillation counter (LSC), a wipe/filter counter such as the Ludlum 2929, or equivalent.

For surface tritium contamination, only the removable contamination will be assessed because of the difficulties in measuring total tritium surface contamination directly (ISO 1988). If a removable fraction of 10% is assumed (ISO 1988), analysis for removable tritium must have a detection limit of not more 500 dpm/100cm^2 so that the total (fixed plus removable) required detection limit of 5,000 dpm/100cm^2 is not exceeded Tritium wipes will be measured in a LSC.

2.1.5 Release of Concrete Rubble for Disposal

The demolition of the NRL facility will generate a significant amount of non-activated concrete waste. NUREG-1640, *Radiological Assessments for Clearance of Materials From Nuclear Facilities* (NRC 2003a), provides acceptable volumetric contamination levels for concrete rubble from nuclear facilities that will be used for recycle or disposal in an industrial or municipal solid waste landfill. The Site Characterization Report states that the concrete floor of the reactor room and possibly other non-activated concrete materials in the bioshield and the Bulk Storage tank are volumetrically contaminated with low levels of tritium (Scientech 2005b). This is potentially a large volume of waste (see Table 2-1).

Table 2-4 provides the normalized mass-based effective dose to the critical group identified in NUREG-1640 for tritium and other isotopes of concern in concrete used for recycling or disposal. Tritium is the only radionuclide of concern where the critical pathway scenario is landfill leachate. Table 2-4 also provides the acceptable release level that corresponds to a potential dose rate of 1 mrem/yr to the critical group. This value will be the release criterion for the isotopes of concern in concrete that will be released for disposed at a landfill. This release criterion will only be applied after surface contamination levels have been reduced to acceptable levels as provided in Section 2.1.4.

TABLE 2-4
RELEASE CRITERION TRITIUM IN CONCRETE RUBBLE

Radionuclide	95th Percentile Dose (mrem/yr per pCi/g)	Associated 1 mrem/yr Release Criteria (pCi/g)[a]	Critical Scenario
Tritium	1.1 E-03	9.1 E+02	Leachate-industrial
Iron-55	1.5E-05	6.7E+04	Processing concrete
Cobalt-60	2.0E+00	5.0E-01	Road building
Nickel-63	1.5E-05	6.7E+04	Processing concrete
Europium-152	8.8E-01	1.1E+00	Road Building

Notes:
a – The associated release criteria is the inverse of the 95th percentile dose.

2.2 DECOMMISSIONING TASKS

2.2.1 Activities and Tasks for Decommissioning Preparation

Several activities will be conducted to prepare the NRL for decommissioning. These include the flowing:

General Cleanup – In preparation for decommissioning, non-reactor related equipment and materials located throughout the facility will be collected and surveyed. Radioactive materials will be segregated from clean materials using the criteria described in Section 2.2.3. Clean materials that will not be used to support the decontamination and decommissioning (D&D) project will be released for reuse or disposal according to survey procedures, the release criteria in Section 2.2.4, and the waste management program outlined in Section 3.2. This will include office furniture, electrical equipment, tools, and so forth.

Packaging of Contaminated Items and Equipment – All radioactive materials located on site prior to initiation reactor D&D activities should be segregated and packaged as LLRW. This would include materials such as activated and contaminated materials stored in the southeastern area of the reactor level and the activated graphite removed from the thermal column.

Isolate Inactive Systems – All inactive systems or systems not required by either technical specifications, safety, or other decommissioning activities will be de-energized or drained and isolated. Some systems may be removed to avoid potential contamination during the removal of activated and contaminated items.

Install Temporary Systems – Temporary systems may be needed to support decommissioning. These may include temporary power outlets, portable lighting, temporary ventilation, and air monitoring equipment.

Remove ACM and Other Hazardous Materials – The DOC will coordinate the disposal of all non-radiological hazardous materials through existing University channels. ACM will be removed by licensed asbestos contractors and all appropriate measure will be taken to limit the spread of airborne asbestos. The DOC will provide the asbestos workers with appropriate radiation safety training commensurate with the potential for exposure to radioactive materials.

Characterizations efforts have not identified radioactively-contaminated ACM. However, the DOC will provide health physics support to the asbestos workers to ensure that, if radioactively-contaminated ACM is identified, it will be controlled and segregated from non-contaminated ACM.

2.2.2 Activities and Tasks for Reactor Demolition

Dismantling the reactor components and decontamination and demolition of the reactor bioshield will be conducted using standard industry techniques. These techniques may use tools such as wire saws, high pressure/ultra-high pressure cutters and sprayers, needle guns, scabblers, jackhammers, torches/plasma arc torches, hydraulic cutters, and standard and long-handled hand tools. The specific tools will be designated in approved procedures or work packages. Local containment structures, high-efficiency particulate air (HEPA) filtered ventilation, and temporary shielding will be used as necessary to prevent the spread of contamination and to maintain personnel exposure as low as reasonably achievable (ALARA).

The general reactor demolition activities are described below but may not follow the specific sequence shown for ALARA, safety, or scheduling reasons. Diagrams of the reactor components and bioshield are provided in Appendix A.

Remove Reactor Assembly Components (Figure A-12 and A-18) – As many reactor assembly components as possible should be removed with water in the reactor tank to provide necessary shielding. Long-handled tools and remotely operated equipment may be used to disassemble components such as the reactor grid plates, the rotary specimen rack (Lazy Susan), control rods, and the emergency spray ring. Information that will be helpful in the removal of these components is provided in the reactor mechanical maintenance and operating manual (Gulf General Atomics 1967). The components should be placed in LLRW containers and shielding should be provided to keep doses ALARA.

Remove Reactor Bridge – The reactor bridge is contaminated with fixed and removable contamination and should be removed to allow better access to the reactor core assembly. The control rod drives and other equipment mounted on the bridge should be removed and surveyed for contamination. These activities should be performed with the reactor tank full of water to provide shielding from the activated reactor assembly.

Drain Reactor Tank, Agitate, and Decontaminate – The water in the reactor tank will be partially drained and the water will be managed according to existing procedures. Because there are radioactive sediment-like deposits on the bottom of the reactor tank, the tank should not be completely drained until the water near the bottom of the tank is agitated to suspend as much as the particulate matter as possible. The tank water will be filtered to remove the particulate mater. After passing through a particulate filter, the tank water can be purified further, stored, and sampled according to current procedures that allow for discharge of treated and sampled water to the sanitary sewer system. Once the tank is empty, it will be washed down using a high-pressure wash. The wash water will managed as previously described. To the extent practical, water should be removed from the primary coolant pipes as well. Water filter media and water not meeting the discharge limits will be managed as LLRW. Dose rates should be carefully monitored as the tank is drained.

Remove Reflector Assembly and Reactor Support – With the reactor tank empty, the reflector assembly and core support are exposed. These aluminum components are held together with stainless steel bolts. The core support and reflector assembly can be removed together after the core support is unbolted from the bottom of the tank (the reactor tank will be a confined space for personnel entry). The components should be lifted from the reactor tank using the overhead gantry crane and placed into approved waste containers. Shielding should be used around the waste containers to keep doses ALARA. If cutting is necessary, potential cutting tools include saws and torches. If possible, tools that can be operated from the top of the reactor will be used to keep doses ALARA.

Decontamination of the Reactor Top – With the activated components removed from the reactor tank, the top of the reactor may be decontaminated. This will be done so that the concrete bioshield can be demolished from the top down, beginning with clean, unactivated concrete. Tools such as concrete scabblers may be most effective for this task.

Demolition of the Reactor Bioshield and Bulk Shielding Tank (Figure A-17) – The Site Characterization Report (Scientech 2005b) estimates that the reactor bioshield is activated to a depth of about 3 feet into the concrete from the tank wall and to a height of about 4 feet from the reactor core centerline (Figure A-18). Therefore, concrete beyond this "radius of activation" is potentially radioactively clean; however, contamination may have migrated along cold joints, rebar, and cracks in the concrete. As such, the potentially clean concrete should be removed first by using tools such as diamond wire saws to segment large portions of the concrete or an excavator-mounted pneumatic demolition hammer to rubbleize the concrete. The potentially clean concrete will be segregated and sampled to verify that the material is not contaminated.

Once the potentially clean concrete has been removed, the remainder of the bioshield concrete may be removed, including the walls of the Bulk Shielding Tank. [The walls of the Bulk Shielding Tank may be removed prior to the bioshield demolition to allow more space on the reactor level floor.] When the beam ports and thermal columns are reached (Figures A-20 and A-21), these items will be removed and size reduced to meet LLRW packaging needs. Activated and contaminated concrete will be packaged and managed as LLRW. Items embedded in the bioshield, such as the shadow shields and their support structures, will require additional characterization. Details on the construction of the embedded items can be found in the specification for construction (General Atomic 1959).

Remove the Reactor Tank – Because the reactor tank may become unstable as the bioshield is removed, the tank should be segmented as portions of the tank are exposed. The reactor tank, constructed of ¼-inch thick aluminum, will be segmented using tools such saws, sheers, or torches. The tank segments will be segregated according to whether or not the aluminum is contaminated, activated, or radiologically clean.

2.2.3 Activities and Tasks for Removal of Contaminated Systems

The major contaminated non-reactor components are associated with the primary coolant system. The system components include the coolant water pipes, nitrogen-16 decay tanks (2), the primary pump, and the heat exchanger. The coolant water pipes and the nitrogen-16 tanks are known to be contaminated. The primary pump and heat exchanger are potentially contaminated and should be treated as contaminated unless confirmed to be clean.

The removal activities are described below but may not follow the specific sequence shown for ALARA, safety, or scheduling reasons. Diagrams of the reactor cooling system are provided in Appendix A. Large components that do not meet the release criteria, may be segmented and sized accordingly for packaging in a LLRW container. Large components may also be shipped for disposal according to applicable U.S. Department of Transportation (DOT) regulations and the waste acceptance criteria of the LLRW disposal facility.

Remove the Primary Coolant Water Pipes (Figures A-11, A-12, A-13 and A-14) – The primary coolant water pipes are contained in the pipe tunnel under the reactor and reactor level floor. The outlet pipe terminates at the large (3,000-gallon) nitrogen-decay tank in the vault under the Mechanical Equipment Room. The inlet pipe originates at the heat exchanger. These pipes are likely internally contaminated. The concrete floor may be removed to provide easy access to the pipes. [The concrete floor is potentially contaminated with tritium and should be managed accordingly as described in Section 2.2.4.] To remove loose contamination, the pipes should be flushed with clean water. The water will be managed as described in Section 2.2.2. Once the pipes are drained, they can be surveyed.

Remove the Primary Pump and Heat Exchanger (Figure A-15) – The primary pump and heat exchanger are located in the Mechanical Equipment Room. These items should be disconnected and removed from the Mechanical Equipment Room intact. To fully characterize these items, they will require additional surveys. If internally contaminated, the heat exchangers and pump will be disposed of as LLRW. The pump motor is separate from the pump and should be free of internal contamination.

Remove the Nitrogen-16 Decay Tanks (Figure A-14) – There are two nitrogen-16 decay tanks located in a vault under the Mechanical Equipment Room. One tank has a 3,000-gallon capacity and the other tank has a 1,500-gallon capacity. These tanks should be flushed out to the extent possible before attempts are made to remove them. To remove these tanks, the floor above the tanks and the roof of the Mechanical Equipment Room will need to be removed. Once the tanks can be accessed, they can be disconnected and removed from the vault using a crane. The Mechanical Equipment Room roof may be replaced if necessary to maintain controls of the facility or to keep out rain and snow. However, once the nitrogen tanks and the other equipment are removed from the room, controls may be unnecessary.

Remove the Dry Fuel Storage Tubes (Figure A-22) – There are dry fuel storage pits located in the floor of the truck loading bay. These pits are 1-foot in diameter, 8 feet deep, and lined with 3/8-inch steel with a 1/2-inch bottom plate. Each pit has a 1-foot thick concrete end plug with a 14-inch diameter steel cap. Each pit contains a storage rack that consists of 24 6-foot aluminum tubes. The concrete floor in the loading bay is 8 inches thick.

Remove the Water Retention System (Figure A-23) – The water retention system, which consists of three floor drains, a trench drain, the shower drain, the utility sink drain, associated piping embedded in the concrete floor of the reactor room, and the 500-gallon storage tank located in a small vault under the western side of the reactor room floor north of the Radioactive Materials Storage Cage. The eastern floor drain showed low levels of removable tritium contamination and the entire water retention system should be considered internally contaminated until completely characterized. The water retention system should remain intact through the majority of the D&D project so that it can be utilized to contain spills and/or process water (such as water needed for concrete cutting or coring).

Remove the Building Exhaust System (Figure A-24) – The building exhaust system, which consists of a filter housing on the storage level, a charcoal filter bed, exhaust ducts, and the roof stack and blower, will be removed at the point in the project when it is no longer needed to control the emissions of radioactive materials. It is expected that the exhaust system will be operated to some extent until after the removal of the contaminated concrete as described in Section 2.2.4. No contamination was identified on or in the exhaust system during the site characterization (Scientech 2005b). However, if it is used during D&D tasks, additional characterization will be necessary before it is removed.

2.2.4 Activities and Tasks for the Removal of Contaminated Concrete and Soil

Concrete materials in the reactor room floor, the truck bay floor, and the pipe tunnel are contaminated with various concentrations of tritium. The concrete will be removed using tools such as pneumatic hammers and excavators. A Waste Management Plan will direct the acceptable disposal pathway for contaminated concrete. Concrete should be segregated based on tritium concentration and disposal pathway.

Four concrete beam catchers are concrete pipe sections cast into the footings of the building across from the original reactor beam ports. These beam catchers have a two-foot inner diameter and are set 6 feet into the earth outside the footing. The concrete pipes are 2 to 3 inches thick. The beam catchers were designed to catch the neutron beam coming from the beam tubes and the ends may have been slightly activated as a result. These beam catchers were used to store various radioactive materials and waste.

Characterization data indicate that the soil under the reactor floor may be minimally impacted with tritium, cobalt-60, and possibly other activation products due to historical tank leakage (Scientech 2005b). Once the Reactor Bay floor and tunnel floor have been removed in suspect areas, the exposed soil will be fully characterized. Contaminated soil will be excavated as required. Contaminated soil will be containerized and disposed of according to its waste classification. Based on the extent of soil contamination, the University may require that the DOC install groundwater wells and sample the shallow groundwater for radioisotopes of concern.

2.2.5 Demolition of the Remaining Clean Structure

The DOC will be responsible for the removal of the radioactively-clean NRL building, concrete slabs, and foundation. Release surveys will be conducted to document that the building materials are acceptable for recycle or disposal in a permitted construction and demolition (C&D) debris landfill. The cast-in-place concrete pilings are expected to stay in place and be covered by clean backfill materials following the release of the site.

2.2.6 Schedule

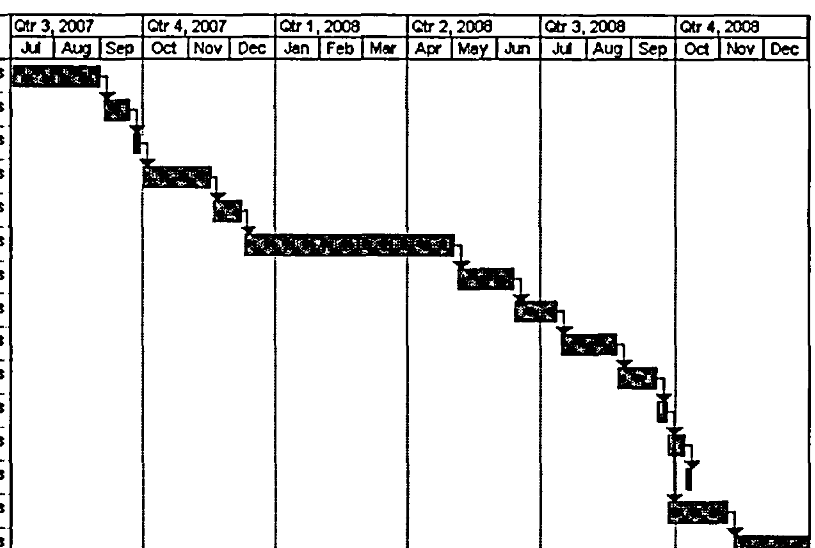
The proposed decommissioning project schedule is provided in Figure 2-1. The schedule, spanning five calendar quarters tentatively beginning with the third quarter of 2007, includes all activities from pre-mobilization project planning to the NRC's review of the Final Status Survey Report and license termination. The project schedule is a floating timeline with an unknown starting date that depends on several uncontrollable factors.

The most time intensive tasks provided in the schedule are the facility preparation and removal of the reactor bioshield. Facility preparation tasks include the removal, packaging, and disposal of all unnecessary equipment (clean and contamination) and facility modifications to allow heavy equipment access to the reactor bay floor. The removal of the bioshield includes the removal and packaging of the Bulk Shielding Tank and bioshield concrete rubble; removal, segmentation, and packaging of the reactor tank; and removal, size reduction, and packaging of all embedded items such as the beam ports, thermal columns, shadow shields.

The schedule is based on a 5 day work week with the number of days listed equal to the number of work days. No time is provided for project down-time or holidays. The schedule does not include the University's bid and procurement activities involved in hiring a decommissioning contractor. The schedule ends at the end of 2008 with license termination; however the length of the regulatory review may take more time than shown.

FIGURE 2-1
PROPOSED DECOMMISSIONING PROJECT SCHEDULE

ID	Task Name	Duration
1	Project Planning	44 days
2	Asbestos Abatement	15 days
3	Full mobilization and training	5 days
4	Facility Preparation	35 days
5	Remove reactor components	15 days
6	Remove bioshield	105 days
7	Remove auxiliary systems	28 days
8	Decontaminate tunnel and floor	22 days
9	Release surveys	28 days
10	Facility demolition	20 days
11	Final status survey/sampling	5 days
12	Site restoration	8 days
13	Demobilization	4 days
14	Final status survey report	30 days
15	NRC review & license termination	38 days



2.3 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

The NRL facility and D&D project is under the supervision of the Reactor Administrator. The Reactor Administrator is responsible for assuring that all decommissioning operations are conducted in a safe manner and within the limits provided by the facility license, the Decommissioning Plan, the Radiation Protection Program, and the provisions of the Nuclear Reactor Committee. The Reactor Administrator will serve as the University Project Manager during the decommissioning with the following duties and responsibilities at a minimum, but not limited to:

- Selecting a decommissioning contractor and overseeing their performance relative to their terms of contract, the decommissioning plan and all subsequent plans and procedures.
- Ensuring that all decommissioning activities comply with applicable regulations and are performed in accordance with all license conditions.
- Approving all plans and procedures required for decommissioning activities.
- Approving minor changes to this decommissioning plan and subsequent plans and procedures (which do not change the original intent of the plan or procedure and do not involve an unreviewed safety question.)
- Communicating with the Nuclear Reactor Committee, decommissioning contractor and the University Administration.
- Communicating with all appropriate regulatory agencies.

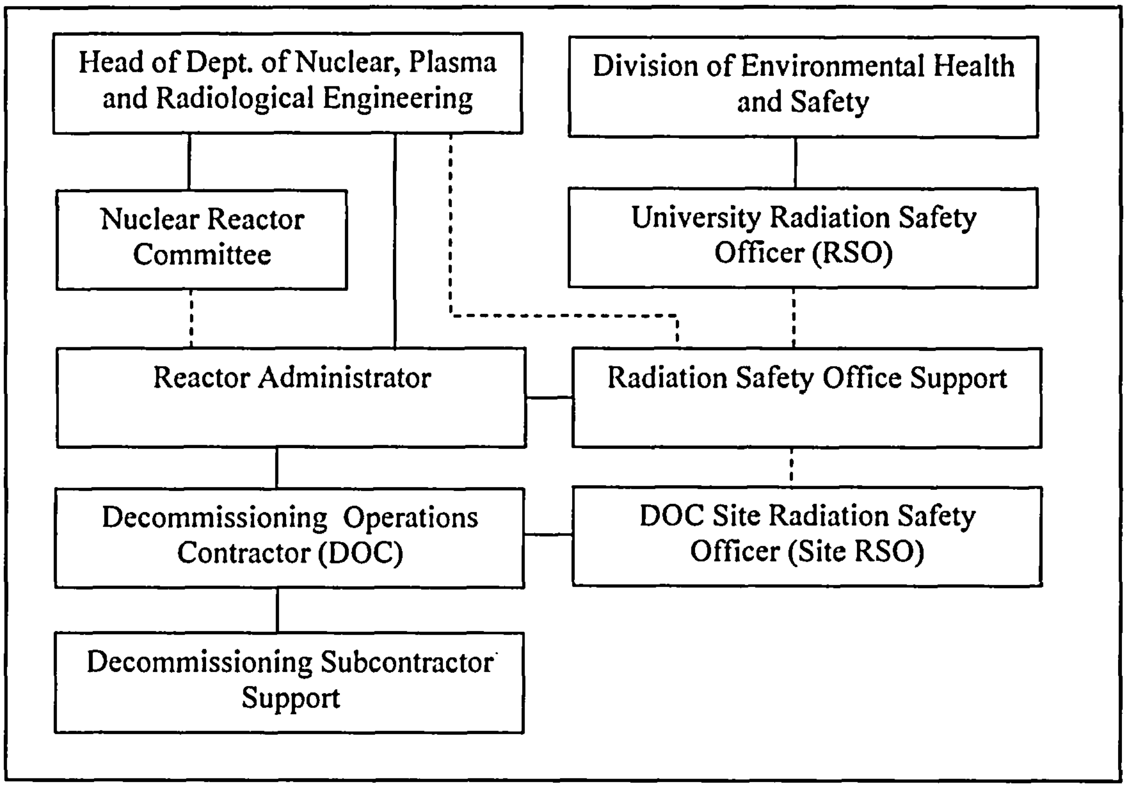
The University Radiation Safety Officer (RSO) shall be responsible for monitoring, planning, and promoting radiological safety at the facility. The University RSO has the responsibility and authority to stop, secure or otherwise control as necessary any operation or activity that poses an unacceptable radiological hazard.

The charter and the rules that describe the function and makeup of the Reactor Committee are provided in the Technical Specifications (University of Illinois 1999). In general, the review function of the Committee includes, but is not limited to, the following:

- Determination that proposed changes in procedures do not involve an unreviewed safety question;
- New procedures and major revisions thereto having safety significance;
- Proposed changes to the technical specifications or license.

The NRL facility management is described by the organizational chart shown in Figure 2-2. The dashed lines indicate reporting paths outside the operational chain of supervision, indicated by solid lines.

FIGURE 2-2
ORGANIZATIONAL CHART



2.4 TRAINING PROGRAM

2.4.1 General Site Training

A general training program will be designed and implemented to provide orientation to project personnel and meet the requirements of 10 CFR 19, *Notices, Instructions, and Reports to Workers: Inspection and Investigations.* General site training will be required for all personnel assigned on a regular basis to the D&D project. General site training will include:

- Project orientation, security, and access control
- Introduction to radiation protection
- Quality assurance
- Industrial safety
- Emergency procedures
- Packaging and transport of radioactive materials

The following are examples of additional training that may be required:

- Radiation Worker Training – will meet the requirements identified in the DOC's Radiation Protection Plan (see Section 2.4.2).
- Hazardous Waste Operations and Emergency Response (HAZWOPER) training – will be required for personnel engaged in hazardous substance removal or other activities that potentially expose them to hazardous substances and health hazards.

- Respirator Training and Fit Testing – will be performed according to the DOC's Respiratory Protection Program.
- Hazard Communication Training – will be provided to all personnel exposed to hazardous or potentially hazardous materials.
- Hearing Conservation Training – will be provided on the effects of noise on hearing and the purpose, advantages, disadvantages, and attenuation of various types of hearing protective devices.
- Permit-Required Confined Space Entry Training – will be required for personnel entering confined spaces
- Lockout/Tagout Hazardous Energy Control Training – for hazardous energy control.
- Trenching and Excavation Training – for the purpose of determining the safety and stability of excavations.

For specific tasks that require state licensing or other special qualifications, the qualifications will be reviewed by the DOC project manager or site safety officer. Additional radiation safety training will be provided to these contractors by the site RSO as necessary.

2.4.2 Radiation Worker Training

The majority of the NRL facility D&D operations will be performed by the DOC and its subcontractors. As such, the DOC will be responsible for the radiation worker training of its employees and verifying that subcontractors are also adequately trained in radiation safety commensurate with their work activities in accordance with the requirements of 10 CFR 19. The DOC Site RSO will be responsible for on-site radiation safety training of workers and verifying pervious training and qualification. The DOC's radiation safety training program will be administered by a Certified Health Physicist (CHP) who will approve all training materials and the designation of the Site RSO. The University RSO may also provide additional training or verification of support staff training prior to providing dose monitoring badges such as thermoluminescent dosimeters (TLD).

The minimum radiation safety training provided to any worker will include, but is not limited to the following subjects:

- Principles of radiation protection
- Radiation monitoring techniques
- Radiation monitoring instrumentation
- Emergency procedures
- Radiation hazards and controls
- Concepts of radiation and contamination
- Provisions of 10 CFR 19 and 20
- NRC license conditions and limitations
- Reporting requirements for workers
- Biological effects of radiation
- Radiation control zone procedures
- Radiation Work Permits (RWP)

A written exam will be required to demonstrate proficiency with the radiation worker training topics. Radiation worker training will also include a practical factors demonstration and evaluation. This evaluation will include a review of the following

- Proper procedures for donning and removing protective clothing and equipment
- The ability of the worker to read and interpret self-reading and/or electronic dosimeters (if used)
- Proper procedures for entering and exiting a controlled area, including proper frisking techniques

Persons who have document equivalent radiation worker training from another site or employer may be waived from taking the training but must take the written and practical factors examinations.

2.5 CONTRACTOR ASSISTANCE

The University will select a qualified contractor to perform all or parts of the NRL facility D&D project. In selecting the contractor, the University will produce a request for proposal, which will define the qualifications and experience necessary for prospective DOCs and subcontractors. Prior history and performance of the prospective contractor on non-power reactor decommissioning projects will be used to help the University select a qualified contractor to perform the facility D&D. The DOC will also be licensed by the NRC to provide decommissioning services.

The selected DOC will manage the physical aspects of their portions of the decommissioning work including QA, health physics, safety, waste processing, and waste packaging and shipping. However, the University will continue to maintain overall responsibility for health and safety, compliance with regulations, and applicable license conditions.

2.6 D&D DOCUMENTS AND GUIDES

This decommissioning plan was prepared using the guidance and format specified in Chapter 17 of NUREG-1537 (NRC 1996). The radiological criteria for license termination to allow unrestricted use will be as set forth in 10 CFR 20, Subpart E. The decommissioning project will also be administered according to the applicable section of the following regulations and regulatory guidance documents:

<u>Code of Federal Regulations</u>

10 CFR 19	"Notices, Instructions and Reports to Workers; Inspections"
10 CFR 20	"Standards for Protection Against Radiation"
10 CFR 30	"Rules of General Applicability to Domestic Licensing of Byproduct Material"
10 CFR 50	"Domestic Licensing of Production and Utilization Facilities"
10 CFR 51	"Licensing and Regulatory Policy and Procedures for Environmental Protection"
10 CFR 71	"Packaging of Radioactive Materials for Transport and Transportation of Radioactive Materials Under Certain Conditions"
29 CFR 1910	"Occupational Safety and Health Standards"
29 CFR 1926	"Occupation Safety and Health Standards for Construction"
49 CFR 170-199	"Department of Transportation Hazardous Materials Regulations"

NRC Regulatory Guides

1.86	"Termination of Operating Licenses for Nuclear Reactors"
1.187	"Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments"
8.2	"Guide for Administrative Practices in Radiation Monitoring"
8.7	"Occupational Radiation Exposure Records Systems"
8.9	"Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program"
8.10	"Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable"
8.13	"Instruction Concerning Prenatal Radiation Exposure"
8.15	"Acceptable Programs for Respiratory Protection"

NRC Guidance Documents (NUREG)

1505	"A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys"
1507	"Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions"
1549	"Using Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination, Draft"
1575	"Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)"
1640	"Radiological Assessments for Clearance of Materials From Nuclear Facilities"
1757	"Technology, Safety, and Cost of Decommissioning Reference Nuclear Research and Test Reactors"

Additional project-specific documents will be developed by the DOC and/or the University prior to starting the D&D project. Such documents may include:

- Radiation Protection and ALARA Plan
- Site Health and Safety Plan
- Quality Assurance Project Plan
- Waste Management Plan
- Final Status Survey Plan
- Specific Task Plans

3.0 PROTECTION OF WORKERS AND THE PUBLIC

3.1 RADIATION PROTECTION

The D&D project Radiation Protection Program (Program) will be administered by University RSO with additional more detailed radiation protections plans and procedures related to facility D&D provided by the DOC. The day-to-day Program implementation will be the responsibility of the DOC Site RSO. The University RSO, the DOC Site RSO and DOC health physics staff will be responsible for implementing ALARA principles; providing radiation worker training; establishing administrative-level occupational and public dose limits; monitoring personnel for occupational exposures; controlling exposures; providing and maintaining radiation monitoring equipment; performing radiation surveys and monitoring; and maintaining records and generating reports as necessary to comply with regulatory and licensing requirements.

3.1.1 Ensuring As Low As Reasonably Achievable Radiation Exposures

The DOC will prepare a Radiation Protection and ALARA Plan (Plan) that will incorporate provisions for minimizing occupational and public radiation exposures. The Plan will describe specific administrative and engineering controls that will be in place during specific D&D project activities. Examples of administrative and engineering controls include limiting access to certain areas, dry-run (mock-up) training, use of remote-handling devices, incorporation of temporary shielding, construction of containment structures, controlling ventilation, and use of specialized protective equipment and respiratory protection.

The Plan will include a description on the methods for evaluating control measures to ensure that implementing the measure will result in an overall risk reduction and not a transfer of the risk. The ALARA evaluation will also include a cost justification and a justification in the context of the overall task or project objectives.

3.1.2 Health Physics Program

The project Health Physics Program will be implemented under the authority of the University RSO with the assistance of the DOC Site RSO. The Health Physics Program will satisfy the following commitments that should be established by the Radiation Protection Program:

- Implement the procedures defined in the Radiation Protection and ALARA Plan.
- Ensure radiological safety of the public, occupationally exposed personnel, and the environment.
- Monitor radiation levels and radioactive materials.
- Control the distribution and release of radioactive materials.
- Maintain potential exposures to the public and occupational radiation exposure to individual within administrative limits and the regulatory limits of 10 CFR 20 and ALARA.

3.1.3 Dose Estimates

The primary doses expected to be received by D&D project workers will be from external exposures to activated materials containing high concentrations of cobalt-60 and europium-152 with little dose expected from internal exposures. The total estimated DDE exposure to complete the D&D project, 8.5 person-rem, accounts for external exposures only. External doses will be monitored using whole-body and extremity TLDs or optically-stimulated luminescent (OSL) dosimeters and possibly electronic dosimeters. Air sampling will be performed to assess the potential for airborne contaminants and internal doses will be monitored if they are expected to exceed 10% of the annual dose limits specified in 10 CFR 20. However, the committed effective dose equivalent (CEDE), the sum of the external and internal doses, is expected to be equal to the DDE.

A task-by-task breakdown of this dose estimate is provided in Table 3-1. Dose estimates are based on the nature of the work involved in each task, the expected number of people assigned to each task, and the estimated task duration as shown on the overall project schedule provided in Section 2.2.6. The maximum possible dose to a single individual is estimated to be about 1.5 rem if the individual were to participate fully in each of the tasks listed in Table 3-1. However, it is not likely that one individual would implement all high-dose tasks; therefore, the 1.5 rem maximum dose is considered a conservative estimate.

The maximum DDE (whole body dose) measured during site characterization activities for 2.5 weeks in July 2005 converted to an hourly dose rate of 0.367 mrem/hr was used to determine the dose estimates provided above. This dose was assumed for low-to-average-dose D&D tasks. Ten times the dose rate was assumed for higher-dose D&D tasks (Tasks 7, 10, 12, and 13 in Table 3-1). To be additionally conservative, the calculations do not account for the decrease in dose in the later D&D tasks after many of the high-activity radioactive materials are removed from the facility. 60-hour work weeks are assumed for the dose estimate calculations.

This estimate is provided for planning purposes only. Detailed dose estimates and exposure controls will be developed in accordance with the requirements of the Radiation Protection and ALARA Plan.

TABLE 3-1
PROJECT DOSE ESTIMATE

Task Number	Task Name	Time (weeks/people)	Estimated Dose (person-rem)
1[a]	Project planning	12/2	0.00
2	General cleanup, prep work, verification surveys	2/4	0.18
3	Packaging contaminated items and equipment	1/2	0.04
4	Isolate inactive systems	1/2	0.04
5	Install temporary systems	2/2	0.09
6	Remove ACM and other hazardous materials	2/4	0.18
7[b]	Remove reactor assembly components	1/6	1.32
8	Remove reactor bridge	0.5/2	0.02
9	Decontaminate reactor tank	0.5/2	0.02
10[b]	Remove reflector assembly and reactor support	1/6	1.32
11	Decontaminate the reactor top	0.5/2	0.02
12[b]	Demolition of the bioshield	2/6	2.64
13[b]	Removal of the reactor tank	1/6	1.32
14	Removal of the primary coolant water pipes	1/6	0.13
15	Removal of the primary pump and heat exchanger	1/6	0.13
16	Removal of the N-16 tanks	2/6	0.26
17	Removal of dry fuel storage tubes	0.5/2	0.02
18	Removal of water retention system	1/4	0.09
19	Removal of contaminated concrete and soil	4/6	0.53
20	Removal of building exhaust system	1/4	0.09
21[a]	Final demolition and excavation	4/6	0.00
22[a]	Final status survey and sampling	2/4	0.00
TOTAL			**8.5 person-rem**

Note: a – No measurable dose expected
b– Higher dose tasks

3.2 RADIOACTIVE WASTE MANAGEMENT

The DOC will implement a Waste Management Plan at the NRL facility D&D project. The Waste Management Plan will be submitted to the Reactor Administrator for review prior to the start of work. The Waste Management Plan will include detailed guidance for the characterization, sampling, classification, segregation, handling, packaging, manifesting, transporting and disposal of all waste categories.

Waste generated during the reactor D&D project will be characterized and segregated on site according to separate categories for removal and disposal. These categories may include: uncontaminated waste acceptable for land disposal or reuse, uncontaminated demolition wastes suitable for land disposal (C&D wastes) or recycle, and Class A LLRW. Additionally, mixed wastes and non-radiological hazardous waste will be segregated from LLRW. Based on the site characterization, Class B and C LLRW are not expected at the NRL facility (Scientech 2005b).

3.2.1 Fuel Removal

The fuel has been removed from the NRL reactor and is no longer stored on-site.

3.2.2 Uncontaminated Wastes For Disposal or Reuse

Uncontaminated wastes will consist primarily of support equipment and building demolition debris. Waste equipment will come from offices, storage areas, work areas, and the control room. These wastes will include desks, chairs, storage shelves and cabinet, and electronic control equipment. These items will be released by the DOC using radiological surveys and the surface contamination release criteria. These waste streams are suitable for disposal at a local solid waste disposal facility or reuse by the University.

Non-radioactive hazardous waste will be managed through the University's Division of Research Safety.

3.2.3 Construction and Demolition Waste

Clean C&D waste is expected to include the structural steel, concrete blocks from the exterior walls, and other roofing and floor materials. C&D waste will be released by the DOC according to release criteria specified in Section 2.1.5 and 2.1.6. C&D waste from the University is currently transported to Brickyard Disposal and Recycling in Danville, IL. The Brickyard facility is operated by the City of Danville. Macks Twin City Recycling of Urbana currently receives all the universities scrap metal products.

3.2.4 Radioactive Waste Processing

The NRL facility D&D project will generate solid LLRW, mixed waste (i.e., contaminated lead and contaminated ACM), hazardous waste (i.e., ACM and oils and fluids drained from equipment), and potentially liquid LLRW (i.e., primary coolant water and decontamination liquids). These wastes will be handled, stored, and disposed of according to applicable state and federal regulations. The DOC will coordinate with the waste disposal site(s) regarding the site's waste acceptance criteria and pre-shipment processing requirements.

Waste processing may include volume reduction through compaction or segmentation, neutralization, stabilization, or solidification. Due to the limited size of the facility and work area, concrete rubbleization beyond that required for demolition will likely not take place on-site. Complying with written procedures, standard work practices, and operating with the limits of the University's, DOC's, or subcontractor's NRC licenses will ensure safe waste processing operations. The decisions as to the type and degree of waste processing will primarily be based on economics that weigh the costs of additional handling and processing compared to transferring the material off-site for treatment and/or disposal.

After the characterization surveys and sampling are complete, wastes will be wrapped, bagged, and/or containerized and staged in the appropriate designated area. Items and containers will be properly labeled as Radioactive Material and the label will indicate the external dose rate from the material. Radioactive wastes will be stored in properly secured radioactive materials storage areas. Logs will be maintained for materials placed in disposal and shipping containers.

3.2.5 Class A and Mixed Radioactive Waste Disposal

Prior to disposal, all waste streams will be properly characterized according to the requirements of the disposal facility. This characterization will include qualification of primary radionuclides of concern as well as hard-to-detect radionuclides. Additionally, those radionuclides that have specific limits for Class A waste will be directly quantified or estimated based on ratios to concentrations of other radionuclides.

All waste will be shipped to an acceptable waste disposal site in accordance with applicable NRC and DOT regulations regarding waste packaging, labeling, and placarding. It is expected that Energy*Solutions*, LLC (formerly Envirocare of Utah, LLC) will receive the Class A D&D wastes at its LLRW disposal site in Clive, Utah. Each LLRW shipment will be accompanied by a shipping manifest as specified in Section I of Appendix F to 10 CFR 20, "Requirements for Low-Level Waste Transfer for Disposal at Land Facilities and Manifests." The waste will be manifested consistent with its classification. Only licensed transporters will be used to transport wastes from the NRL facility.

Mixed wastes may be shipped to a licensed processing facility or directly to a licensed land disposal facility depending on the nature of the waste and the treatment options available.

3.3 GENERAL INDUSTRIAL SAFETY PROGRAM

DOC industrial safety and hygiene personnel, such as Certified Safety Professionals or Certified Industrial Hygienists, along with project management personnel, will be responsible for ensuring that the D&D project complies with all applicable federal safety requirements and general safe work practices. The DOC will prepare a Site Health and Safety Plan (SHASP) as well as a Fire Protection Plan to document safety requirements and accident response procedures.

All DOC personnel working on the D&D project will receive health and safety training in order to recognize and understand potential hazards and risks. Training requirements for DOC subcontractors will be determined by the DOC Site Health and Safety Officer (SHSO) based on the specific task the subcontractor is performing.

3.4 SITE HEALTH AND SAFETY PLAN

The SHASP will be submitted to University personnel for review and approval. The SHASP will direct site activities necessary for ensuring that the NRL facility D&D project meets occupational safety and health requirements for protection of project personnel. The functional responsibility of the SHASP will be to ensure compliance with the Occupational Safety and Health Act (OSHA) of 1973. Illinois is not a state plan state and there no additional state occupation safety and health requirements. The SHASP is implemented on-site by the SHSO.

As a minimum, the SHASP will include the following:

- Hazards assessment
- General site safety procedures

- A requirement for a daily site safety meeting
- Site inspection procedures
- Emergency response procedures
- Emergency contact telephone numbers
- Material Safety Data Sheets for hazardous materials present on-site
- Training requirements for specific activities such as permit-required confined space entry or hot work
- Local emergency medical information

3.4.1 Fire Safety Plan

The DOC will develop a Fire Safety Plan that will be reviewed and approved by the University. While the NRL facility is constructed of mostly inflammable materials such as metal and concrete, some D&D activities, such as hot cutting, have a potential to ignite dry solid materials such as personal protective equipment (PPE), rags, and wipes. Some flammable materials, such as gasoline or cutting torch fuels may also be present on site during D&D operations. During such activities where the potential exists for accidental ignition a fire watch will be posted. Proper storage and use of flammable and ignitable materials, the use of portable fire extinguisher, and external fire department support will be described in the Fire Safety Plan.

3.5 RADIOLOGICAL ACCIDENT ANALYSES

There is a potential for radiological accidents during the NRL facility D&D project resulting from the uncontrolled release of radioactive materials to the work area or the environment. These releases are most likely associated with the management of contaminated liquids in the reactor tank, the primary coolant piping, the heat exchanger, and the nitrogen-16 decay tanks. Uncontrolled releases of airborne contamination could also occur during the demolition of the reactor bioshield and segmentation of activated and/or contaminated reactor components such as the reactor tank and the beam tubes. An uncontrolled release of radioactive material could also occur during a transportation accident.

The accidental dropping of an activated reactor component was also considered as a potential accident. However, because the more highly activated components are located under water, the surface contamination on these parts would not be sufficiently high to release significant quantities of radioactive materials during such an incident. Such an incident would mostly likely result in additional unplanned external exposures.

A fire is another possible source of an uncontrolled release of radioactive materials. However, the majority of the combustibles that will be present on site will be clean materials. Potentially contaminated combustibles will include dry active waste such as personal protective clothing and rags and towels used for site cleanup and decontamination. The radioactivity contained in these materials would not be high enough to result in a significant release of during such an incident.

There are no fissile materials located on site that could result in a criticality incident.

The consequence levels discussed in the following paragraphs are described in the U.S. Department of Energy (DOE) Standard DOE-STD-1120-2005, "Integration of Environment, Safety, and Health Into Facility Disposition Activities" (DOE 2005).

3.5.1 Release of Contaminated Liquid

An uncontrolled release of radioactively contaminated liquids could result in the contamination of workers, the NRL facility, or the environment. Liquids containing radioactive suspended solids containing activation products (primarily cobalt-60) are present in the reactor tank, the primary coolant water pipes, the nitrogen-16 decay tanks, and possibly the heat exchanger. These liquids will be drained or pumped out during the D&D project and filtered to remove the suspended radioactive contaminants.

Accidents could occur during the draining or pumping activities. Hoses could burst or come free from pumps resulting in an uncontrolled release. To mitigate the extent of such releases, process involving contaminated liquids will only be operated with personnel present and personnel will watch for leaks and spills and respond by shutting down the activity. The safe shut down process will not allow for additional water to leak from the system. A spill kit will be readily available to respond to any incidents.

While the concentrations of radioactive materials in the liquids is not known for certain and will vary, the dose consequence is expected to be low. The nitrogen-16 tanks likely contain water with the highest concentration of contaminants and the current dose rate around the nitrogen-16 decay tanks is less than 5 mrem/hr. A MicroShield analysis of the dose rate and the source suggest that the residual contamination in the nitrogen-16 tank is on the order of 0.01 to 0.02 microcuries per milliliter (uCi/ml). This assumes that the residual contamination is not suspended throughout the volume of the tank (settled to the bottom) and is therefore a maximum potential concentration.

An uncontrolled release of the contaminated water may result in only incidental ingestion, short term dermal contact, and external exposures. The oral ingestion annual limit on intake (ALI) in Appendix B of 10 CFR 20 for cobalt-60 is 500 uCi (the lowest ALI of the contaminants of concern). The ALI corresponds to CEDE of 5 mrem. To approach the oral ingestion ALI, more than 25 liters of contaminated water (0.02 uCi/ml cobalt-60) would need to be ingested. External exposures would also be far less than the current dose measured dose rate in an accident scenario because the activity would be diluted over a large area. A plausible accident scenario may result in the ingestion of several milliliters of contaminated water and exposure to the material for an 8-hour period the resulting occupational CEDE would then be about 40.2 mrem. Therefore, the resulting dose in an accident involving the release of contaminated liquids would be far less than 1 roentgen-equivalent man (rem) to off-site receptors and 25 rem to on-site workers. As such, safety management operations (standard engineering and administrative controls) are sufficient for protecting against such accidents.

3.5.2 Release of Airborne Contamination

An uncontrolled release of airborne radioactivity could occur during cutting and demolition activities involving contaminated or activated materials. Such activities may take place inside temporary containment structures equipped with HEPA filter ventilation systems. The failure of the containment structure could result in the release of airborne radioactive materials into the NRL facility. If the negative pressure is still maintained in the NRL at the time of such an incident, the facility air filter system would prevent release to the environment. If the air flow system in the NRL facility is not operating at the time of such an incident, airborne radioactive material could be released directly to the environment. Alarming continuous air monitors (CAM) will be used in the work areas to warn against the release of airborne radioactivity.

Temporary containment systems with HEPA filter systems will likely vent to the NRL building or tie into existing building ventilation. A failure in the HEPA filter system could result in the uncontrolled release of airborne radioactive materials. CAM will be used to monitor effluent air. If allowable effluent criteria are exceeded, the CAM will alarm and operations inside the containment structure will immediately stop.

Europium-152 has the most-limiting inhalation ALI of the contaminants of concern and the derived air concentration (DAC) is 1E-8 uCi/ml. The DAC concentration is the air concentration that results in 1 ALI, or 5 rem to the exposed individual in a 2,000-hour work year. The highest measured europium-152 concentration in the bioshield concrete was 9E-3 microcuries per gram (uCi/g) (Scientech 2005b). If concrete dust at its worst case concentration were to become airborne as a result of an uncontrolled release, the breathing air would be limited to a respirable particulate loading of 1.1 ug/ml of air (or 1.1 mg/l) before the DAC was exceeded. Given that the interior free volume of reactor room is about 70,000 cubic feet (University of Illinois 1999) (or about 2,000,000 cubic liters), about 2.2 kg of the most contaminated concrete would have to become airborne to reach the DAC level. The ALI of 5 rem would not be approached until a worker was exposed to this airborne concentration for 2,000 hours. Because the DAC level is based on an exposure duration of a year, the uncontrolled release of air at the DAC level to the air outside the facility would have minimal dose consequence due to the short duration of such an accidental release.

While the actual concentrations of airborne radioactive materials are unknown at this time, as demonstrated in the previous paragraph, the dose consequence of an uncontrolled release is expected to be low (< 1 rem off-site impact and < 25 rem to on-site workers). As such, safety management operations (standard engineering and administrative controls) are sufficient for protecting against such accidents.

3.5.3 Transportation Accidents

Various forms and quantities of radioactive waste will be shipped from the NRL facility during the D&D project. The dose consequence from transportation accidents could be higher than the contamination accident scenarios described in Section 3.5.1 and 3.5.2 because high-activity reactor components could be involved. As such, there is a potential for a moderate dose consequence of between 1 and 25 rem for the public following a transportation accident. However, adherence to NRC and DOT radioactive material packaging and transportation requirements is considered a sufficient control measure for mitigating transportation-related incidents.

4.0 PROPOSED FINAL STATUS SURVEY PLAN

4.1 SURVEY AND SAMPLING APPROACH

The NRL structure, concrete slabs, and foundation will be removed prior to site release with only the cast-in-place pilings possibly remaining. As such, the final status survey (FSS) will cover only the exposed soils within the footprint of the structure, the pilings, and the surface and subsurface soils surrounding the NRL facility.

The FSS will be developed following the guidance provided in NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" (NRC 2000) to demonstrate compliance with the release criteria provided in Section 2.1.3. The MARSSIM process emphasizes the use of data quality objectives (DQO), proper classification of survey areas (survey units), a statistically-based survey and sampling plan, and an adequate quality assurance/quality control (QA/QC) program.

The FSS will be performed in accordance with a Final Status Survey Plan (FSSP) by trained DOC technicians experienced in performing FSSs. The technicians will follow written procedures covering surveys and sampling, sample collection and handling, chain-of-custody, and recordkeeping. The FSSP will define sampling locations, analysis required, and survey types. The FSSP will also direct surveys or sampling to meet any additional release criteria set forth by the University or the State of Illinois.

The FSS will include surface walk-over gamma surveys using sodium-iodide (NaI) gamma scintillation detectors. Surface and subsurface soil samples will be collected using a random-start grid pattern. Soil samples will be analyzed for contaminants of concern using standard analytical methods including liquid scintillation counting for hard-to-detect beta-emitting radionuclides and gamma spectroscopy for gamma-emitting radionuclides.

4.2 DATA QUALITY OBJECTIVES

The object of the FSS is to demonstrate that the radiological conditions of the NRL site satisfy the decommissioning criteria provided in Section 2.1. The DQOs in the MARSSIM survey approach for surface soils will provide a 95% confidence level in demonstrating that the site meets the criteria. Therefore, Type I and Type II decision errors will be 5-percent. These decision error rates are used in deterring the number of samples necessary from each survey unit and the background reference areas as well as in the final nonparametric statistical test used to evaluate contaminant concentrations in the survey units against release criteria.

DQOs, which will be fully described in the FSSP, will also include limits on the sensitivities of survey and analytical methods. Typical sensitivities for walkover surveys are less than or equal to 100% of the DCGL and sample analytical techniques are typically less than 50% of the DCGL. Data quality indicators such as precision, accuracy, and completeness will also be evaluated according to MARSSIM protocols.

As stated in Section 2.1.4, the QAPP will incorporate standard regulatory and industry measures applicable to the FSS. The QAPP will be reviewed and approved by the Reactor Administrator.

4.3 IDENTIFICATION AND CLASSIFICATION OF SURVEY UNITS

Survey units will be classified based on contamination potential according to the methods described in MARSSIM. MARSSIM defines Class 1, Class 2, and Class 3 areas. Class 1 survey units have the highest potential for residual radioactive contamination greater than the DCGLs while Class 3 survey units have the lowest potential.

Based on the classification of a survey unit, its size is limited. It is expected that the footprint of the NRL facility will be designated as a Class 1 area. MARSSIM limits the size of a Class 1 land area survey unit to 2,000 square meters. Therefore, the footprint of the NRL facility many include only one Class 1 survey Unit. The land area surrounding the Class 1 survey unit will likely be designated as a single Class 2 or Class 3 survey unit (limited to 10,000 square meters).

4.4 FINAL STATUS SURVEY REPORT

The Final Status Survey Report (FSSR) will be prepared to present the findings of the FSS, including all FSS data and data analysis. The FSSR will be provided to the NRC in support of the license termination request.

5.0 TECHNICAL SPECIFICATIONS

The NRL facility currently operates under technical specifications that are included as Appendix A of NRC License R-115. These technical specifications are in place to insure the safe operation of the reactor facility. However, most of the technical specifications do not apply to the reactor when it is not in operation. Other technical specifications that apply to non-operating conditions have been amended since reactor shutdown. If additional changes to the technical specifications are necessary prior to D&D operations, the University will request that changes be approved by the NRC with a license amendment.

6.0 PHYSICAL SECURITY PLAN

Under regulations enacted by the DOT in 2004, those responsible for the transportation of hazardous materials, including Class 7 radioactive materials, must receive security training and, in some instances, prepare Security Plans to direct security measures for shipment of radioactive materials.

Based on the nature of the LLRW that will be shipped from the NRL D&D project site, a Security Plan will be developed according to the requirements of 49 CFR 172.800. The Security Plan will cover the control of radioactive materials on-site and in transport. The plan will also address the security training requirements for on-site personnel in 49 CFR 172.702.

The Security Plan will include an assessment of the possible storage and transportation security risks for radioactive materials and the appropriate measures necessary to address the assessed risks. Specific measures put into place by the Security Plan may vary commensurate with the level of threat at a particular time. As such, a Security Plan may require changes over the course of a long-term project. The following are the minimum components of the Security Plan:

- *Personnel Security* – Measures to confirm information provided by job applicants, full-time and temporary, hired for positions that involve access to and handling RAM and/or LLRW covered by the Security Plan.
- *Unauthorized Access* – Measures to address the assessed risk that unauthorized persons may gain access to the RAM of LLRW covered by the Security Plan or transport conveyances being prepared for transportation of the RAM of LLRW covered by the Security Plan.
- *En Route Security* – Measures to address the assessed security risks of shipments of RAM of LLRW covered by the Security Plan en route from origin to destination, including storage.

7.0 EMERGENCY PLAN

The University has an Emergency Plan for responding to emergencies at the NRL facility. The Emergency Plan involves the response of the University police department and local fire and emergency medical services. The plan covers events involving the potential or actual release of radioactivity and provides measures for evacuation, reentry, recovery, and medical support. The D&D project will initially adopt the Emergency Plan as written. Substantive changes to the plan will be reviewed and approved by the Nuclear Reactor Committee and the Reactor Administrator. Minor changes to the plan that do not change the original intent of the plan do not require the approval of the Nuclear Reactor Committee but do require the approval by the Reactor Administrator.

8.0 ENVIRONMENTAL REPORT

The Environmental Report (ER) (Envirocare of Utah 2005) was prepared in December 2005 and was submitted to the NRC along with the submittal of this Decommissioning Plan. The ER was prepared in accordance with the guidance provided in Chapter 6.0 of the NRC Office of Nuclear Material and Safety and Safeguards' (NMSS) NUREG-1748, *Environmental Review Guidance for Licensing Actions Associated with NMSS Programs* (NRC 2003b). This ER is designed to be used by the NRC in conducting its environmental assessment in accordance with the National Environmental Policy Act (NEPA) of 1969. NEPA requires Federal agencies, as part of their decision-making process, to consider the environmental impacts of actions under their jurisdiction. The NRC's NEPA requirements are provided in 10 CFR 51.

9.0 CHANGES TO THE DECOMMISSIONING PLAN

The Decommissioning Plan will be approved by the NRC and incorporated as a license amendment. Minor changes to the Decommissioning Plan which do not change the original intent of the Plan and which do not involve an unreviewed safety question may be approved by the Reactor Administrator.

If a significant change to the Decommissioning Plan is required the Nuclear Reactor Committee will apply the test identified in 10 CFR 50.59 (effective date March 2001) as it applies to non-power reactors in decommissioning. Should the Committee determine that the change is significant and could pose a significant increase in potential worker, public, or environmental impacts, NRC approval will be obtained prior to implementing the change. Guidance on implementing the requirements 10 CFR 50.59 is provided in the following documents:

- NRC Regulatory Guide 1.187 "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments"
- Nuclear Energy Institute (NEI) Guidance NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, September 2000
- NRC Inspection Guidance (Part 9900)

All changes to the Decommissioning Plan will be documented and records of changes will be maintained until license termination. All changes to the Plan will be described in the FSSR.

10.0 REFERENCES

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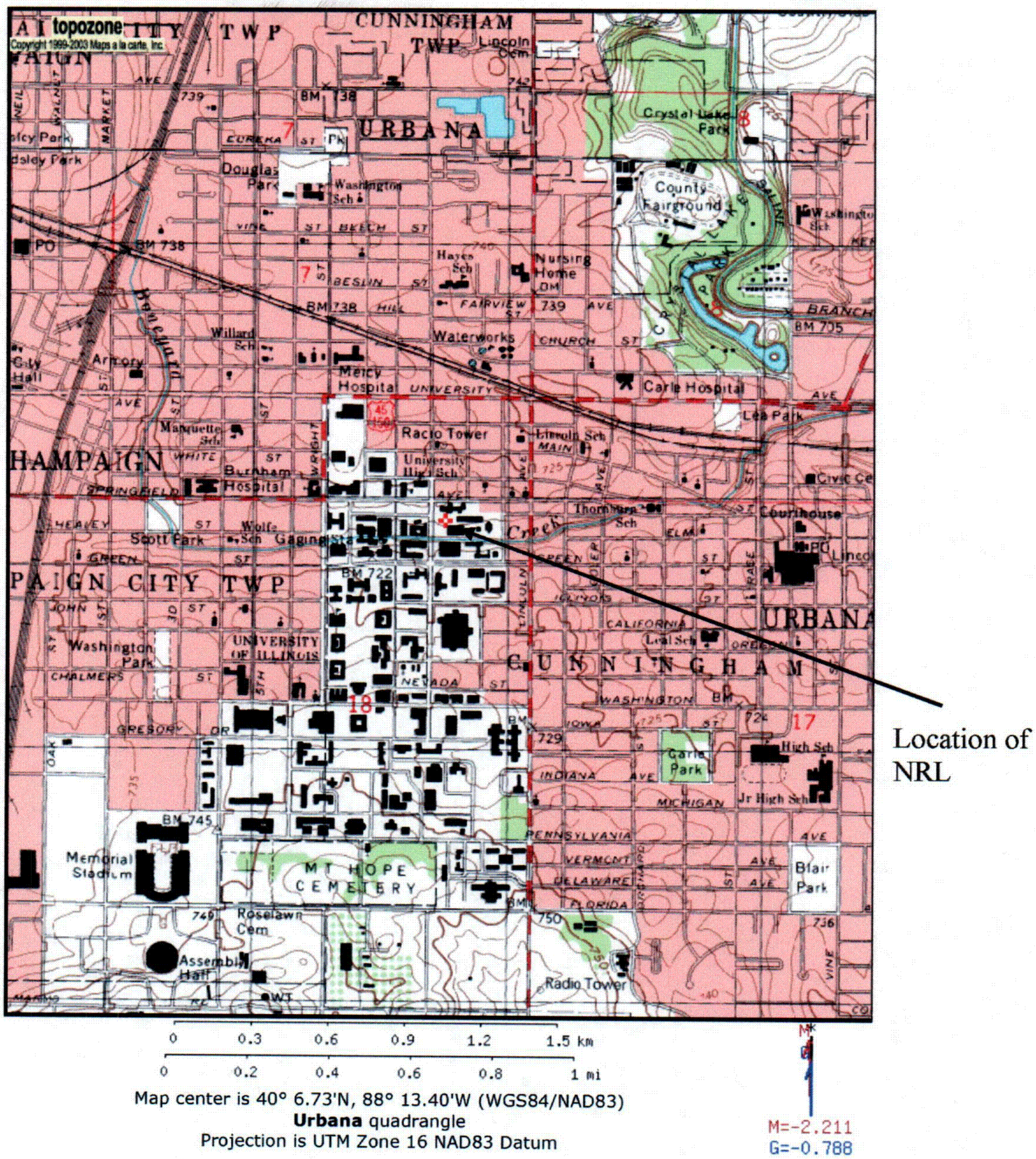
APPENDIX A

FIGURES

NUCLEAR RESEARCH LABORATORY
UNIVERSITY OF ILLINOIS AT URBANA-CHAMPAIGN

TopoZone - The Web's Topographic Map

Page 1 of 1



http://www.topozone.com/print.asp?z=16&n=4440928.00011339&e=395745.999999324&... 10/6/2005

Figure A-1
Location of NRL Facility

C07

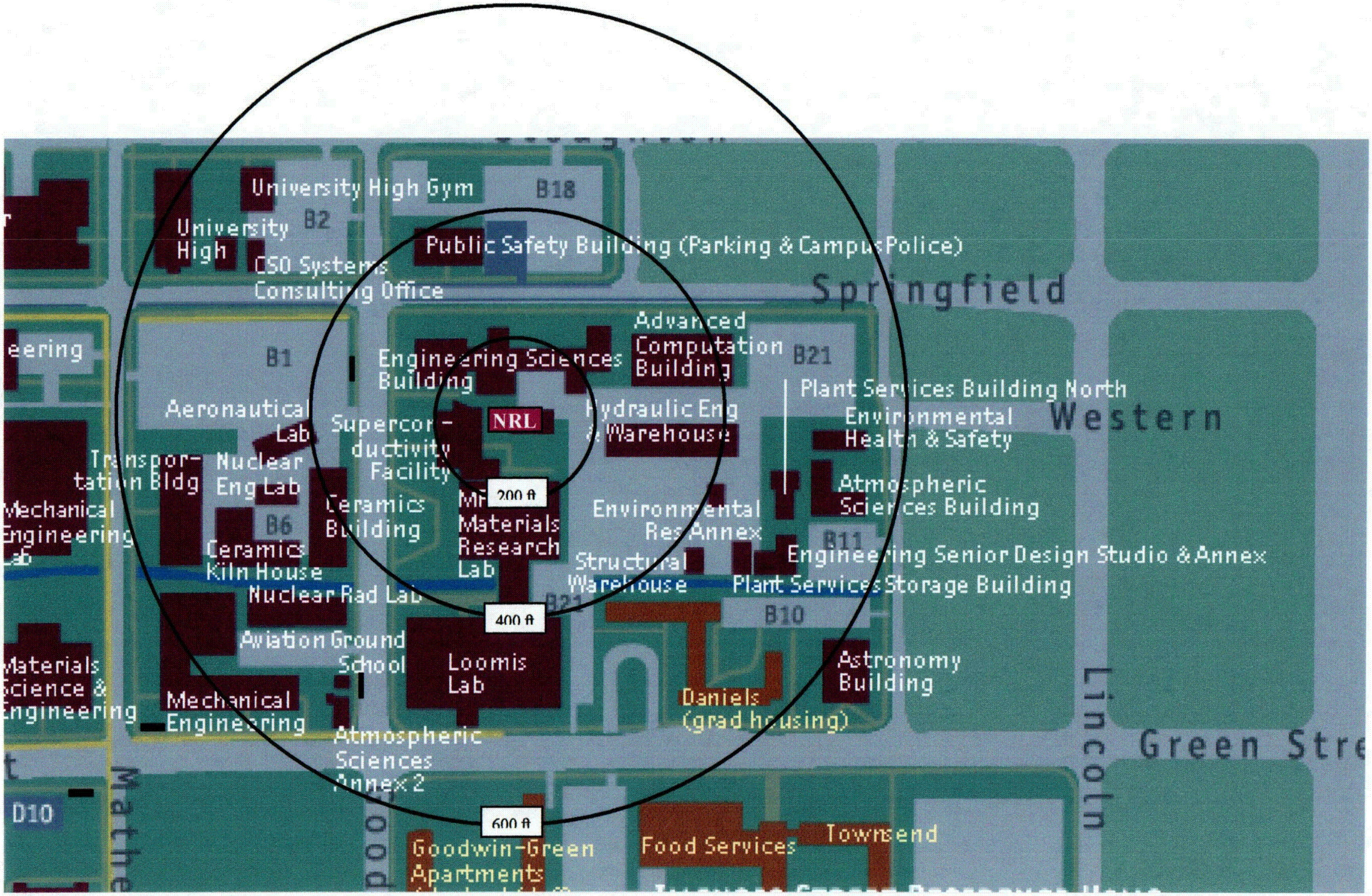


Figure A-2
Campus Map Showing the Location of the NRL Facility

C08

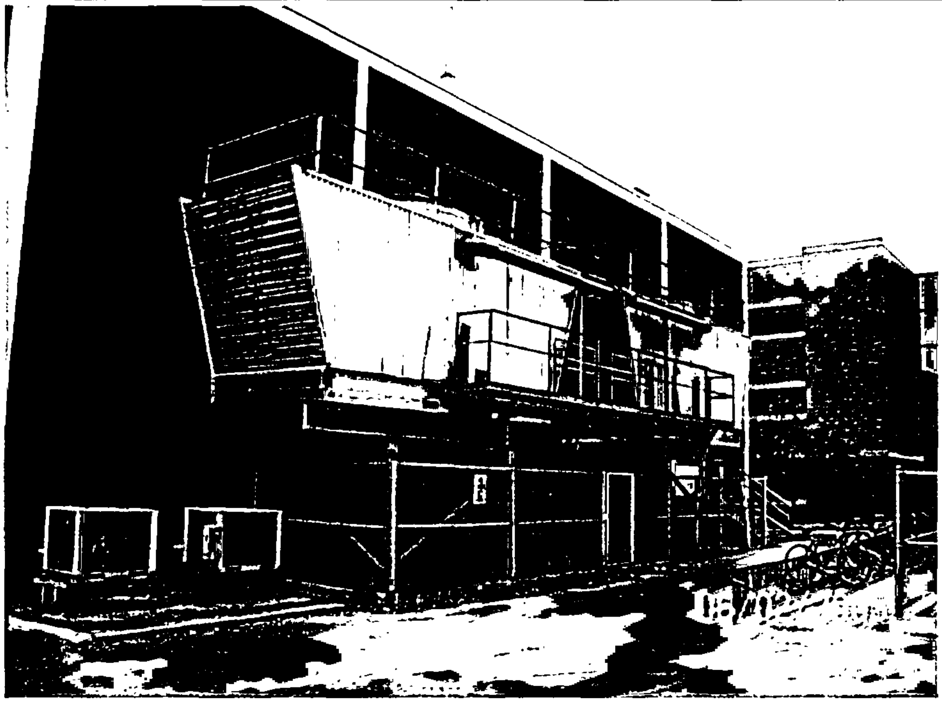


Figure A-3
North Side of NRL Building with Cooling Towers

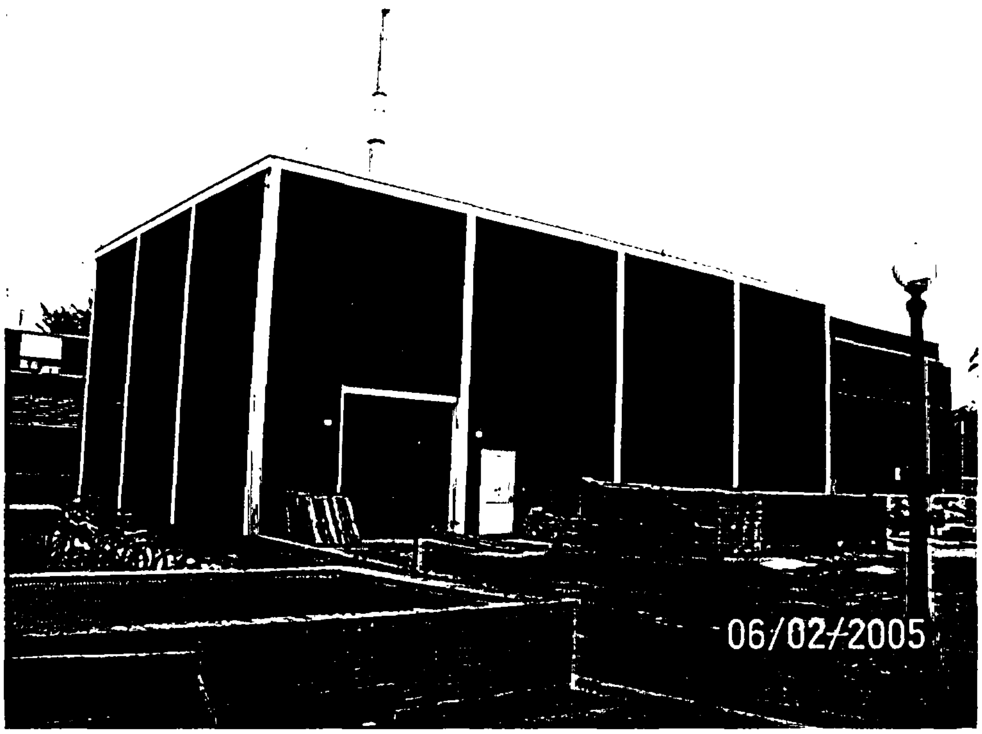


Figure A-4
South Side of NRL Building

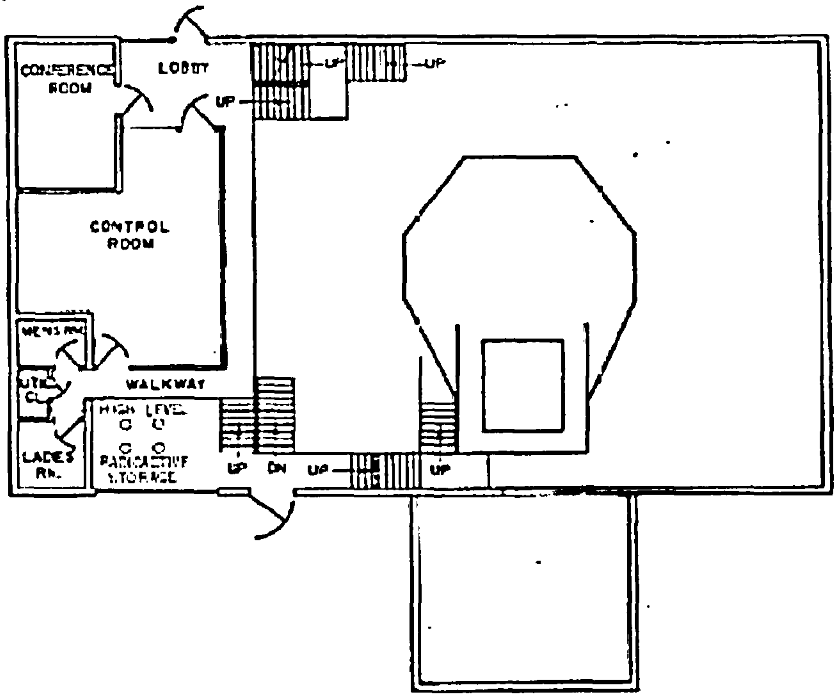


Figure A-5
Mezzanine Level Floor Plan

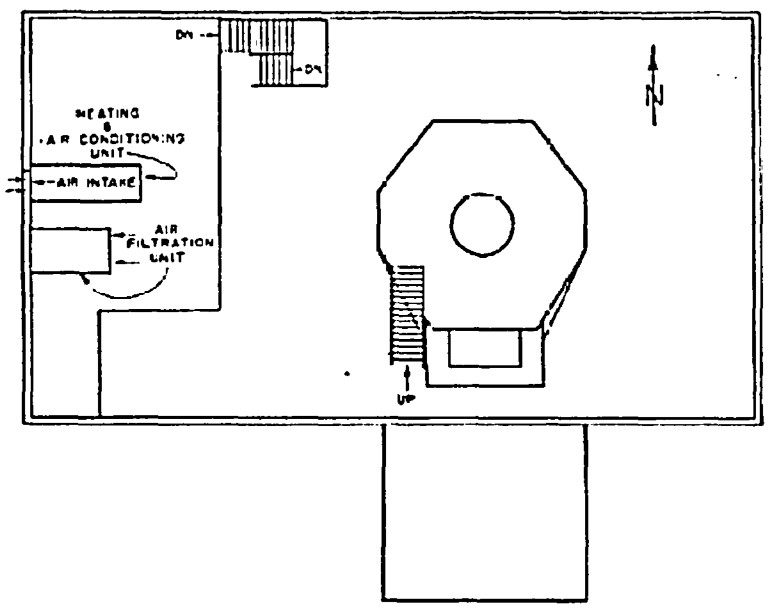


Figure A-6
Storage Level Floor Plan

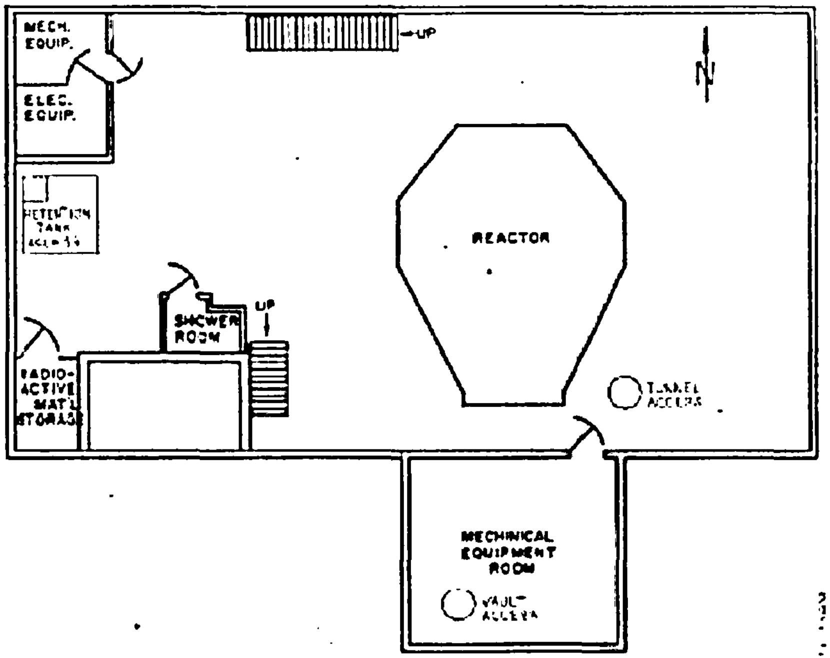


Figure A-7
Reactor Level Floor Plan

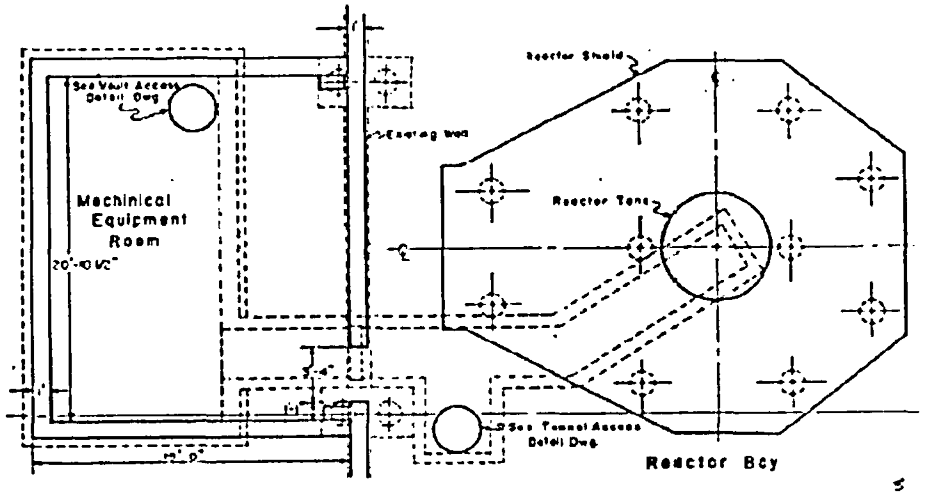


Figure A-8
Reactor Level Tunnel and Vault

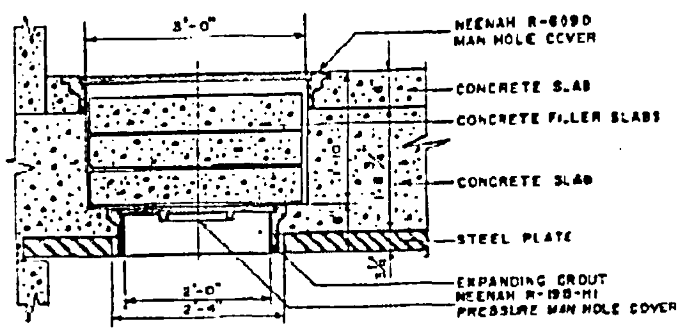


Figure A-9
Vault Access Details

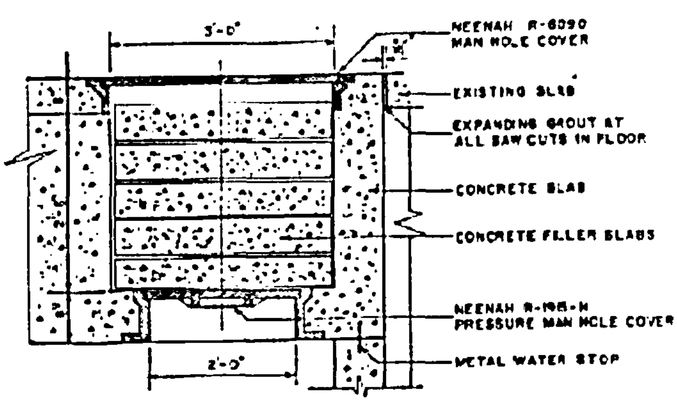


Figure A-10
Tunnel Access Details

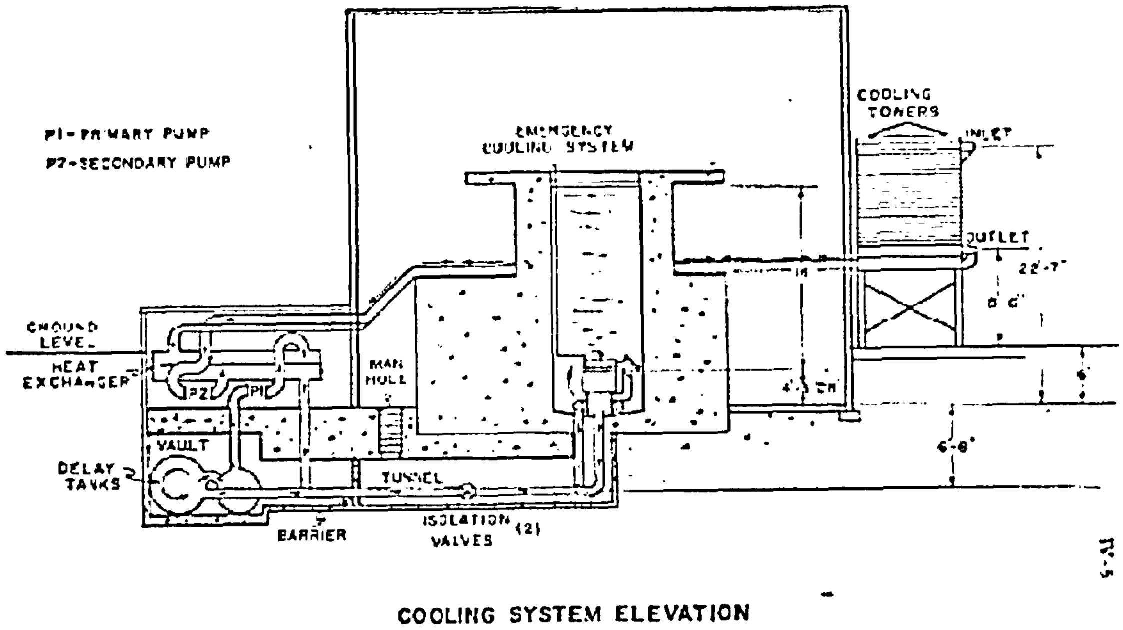


Figure A-11
Cooling System (East) Elevation of NRL Facility

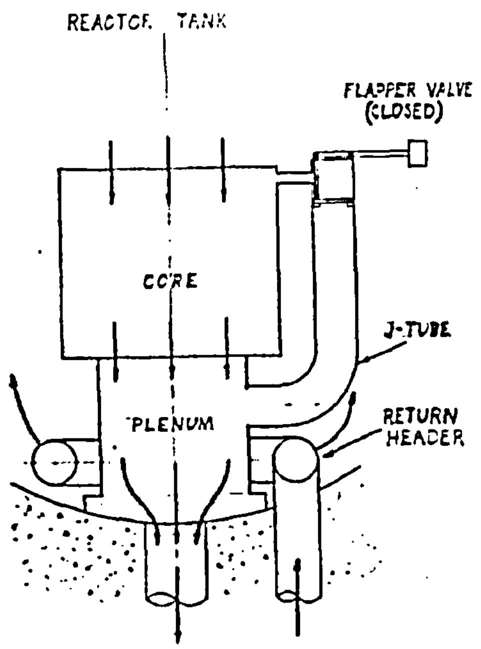


Figure A-12
Forced Convection Flow of Primary Coolant from Reactor Tank

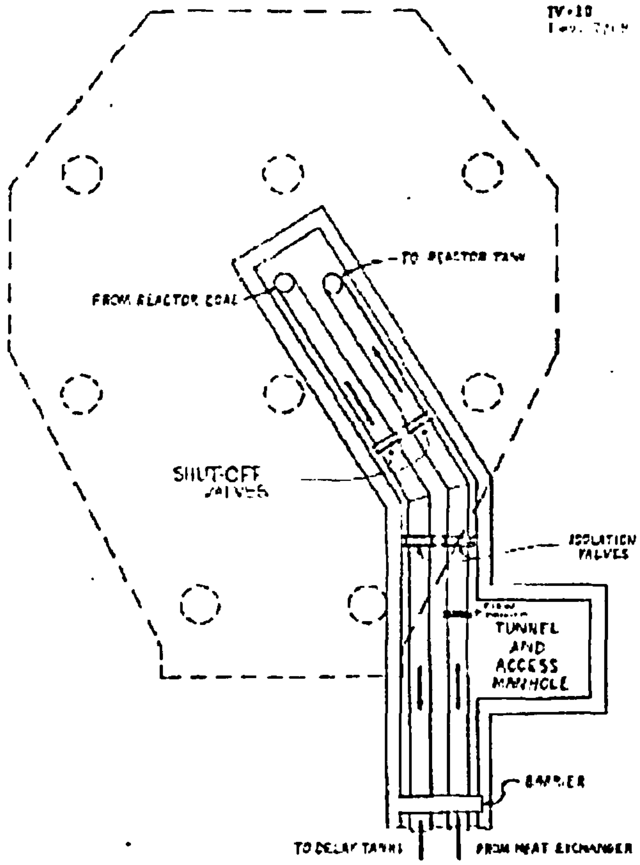


Figure A-13
Primary Coolant Water Pipes in Tunnel

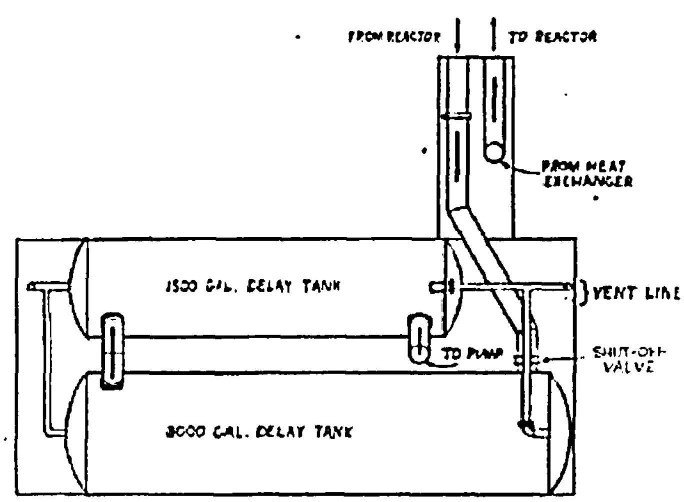


Figure A-14
Nitrogen-16 Decay Tanks

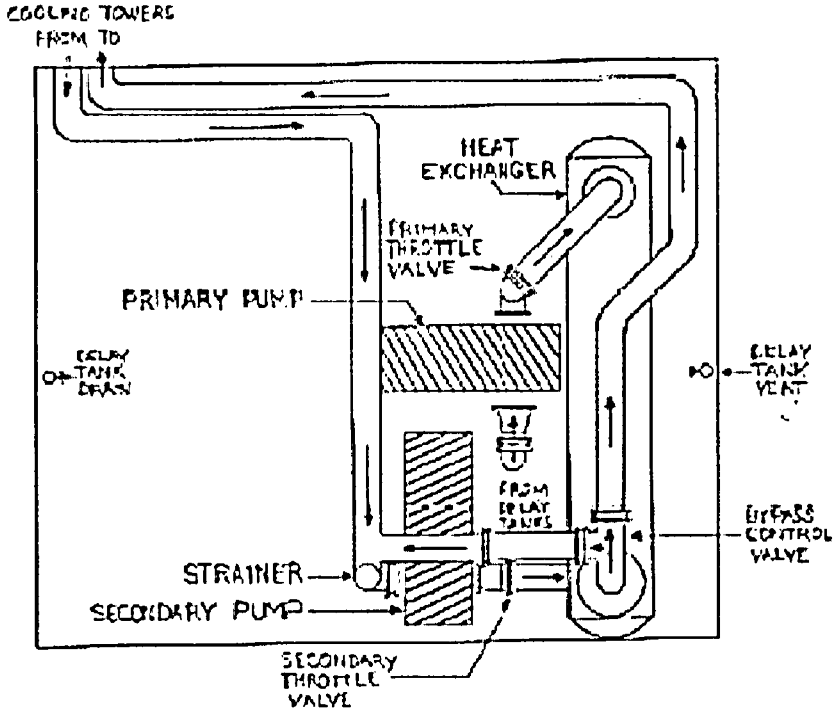


Figure A-15
Arrangement of Primary and Secondary Coolant System Components in the Mechanical Equipment Room

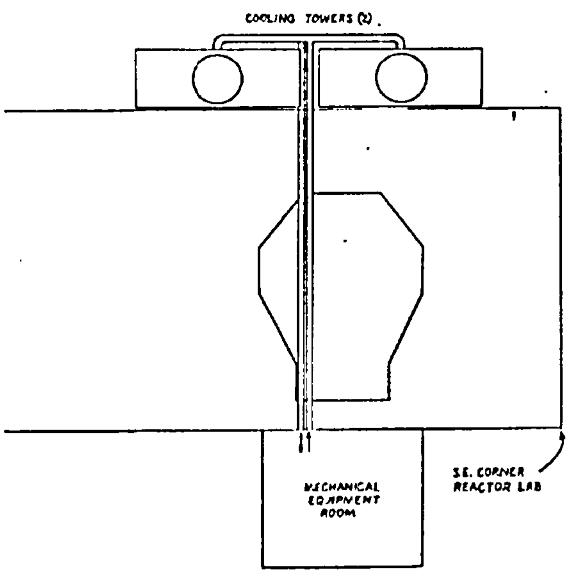


Figure A-16
Flow of Secondary Coolant to Cooling Towers

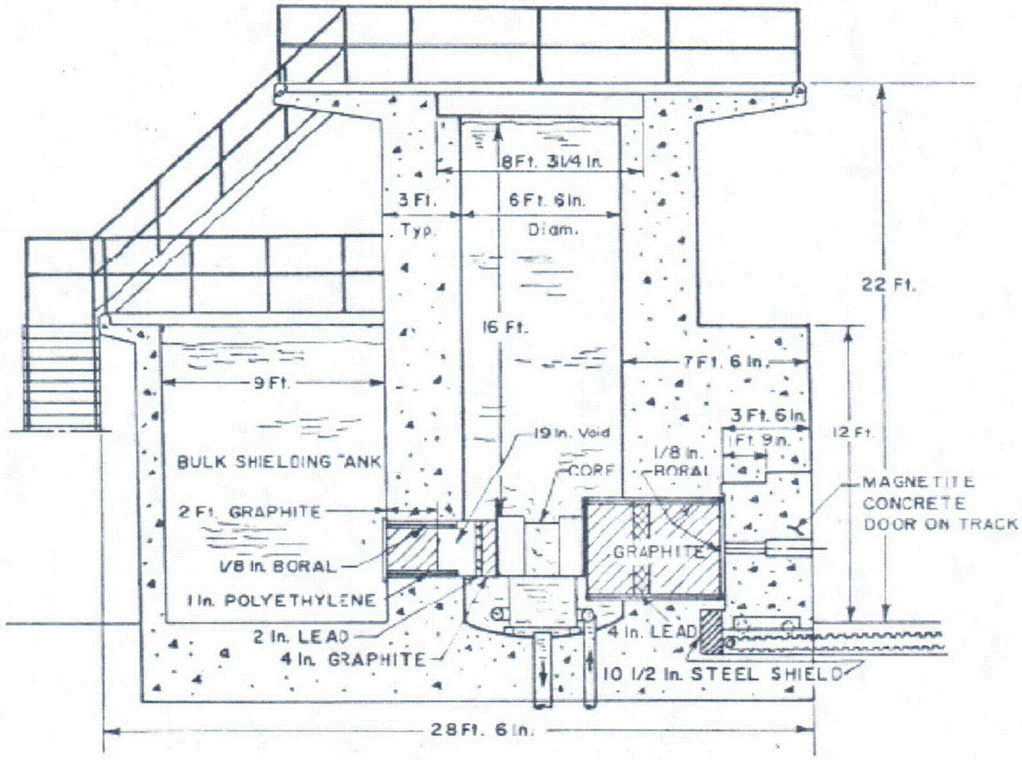


Figure A-17
Reactor Section through Bulk Shielding Tank and Thermal Column

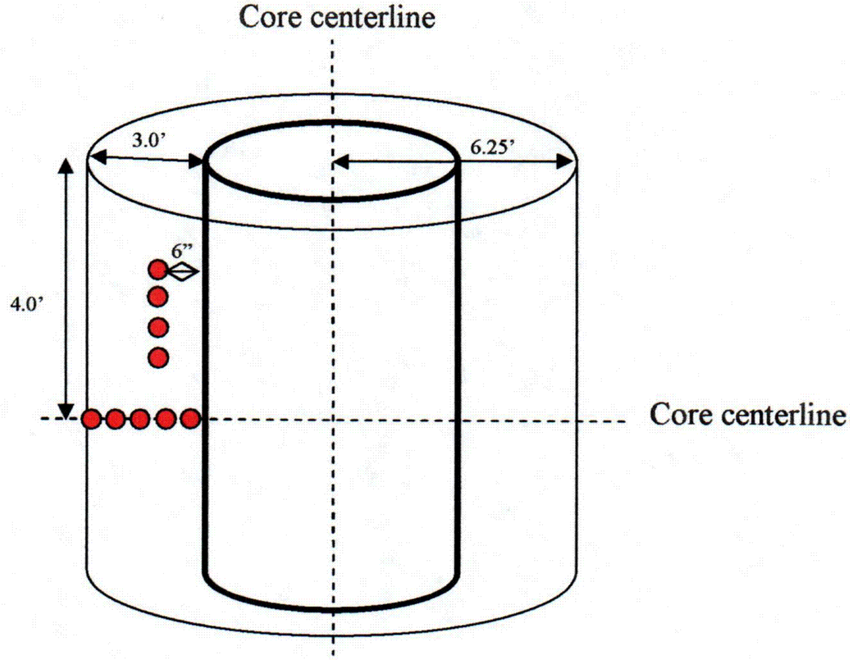


Figure A-18
Estimated Configuration of Activated Portion of the Bioshield

C09

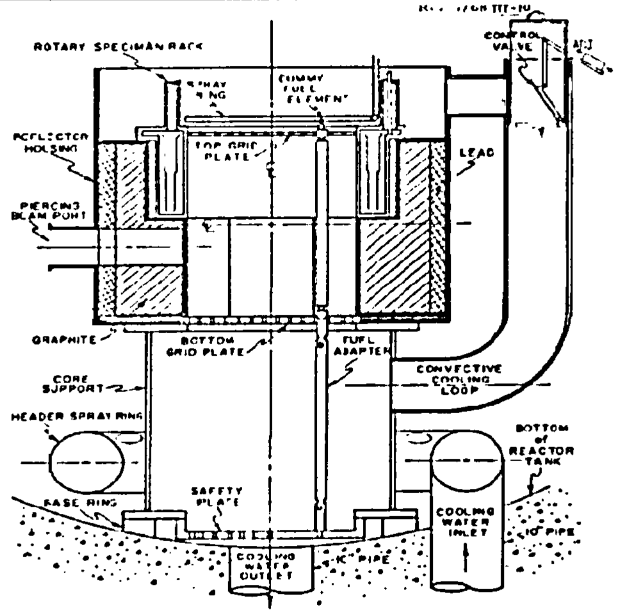


Figure A-19
Vertical Section of Reflector and Core Assembly

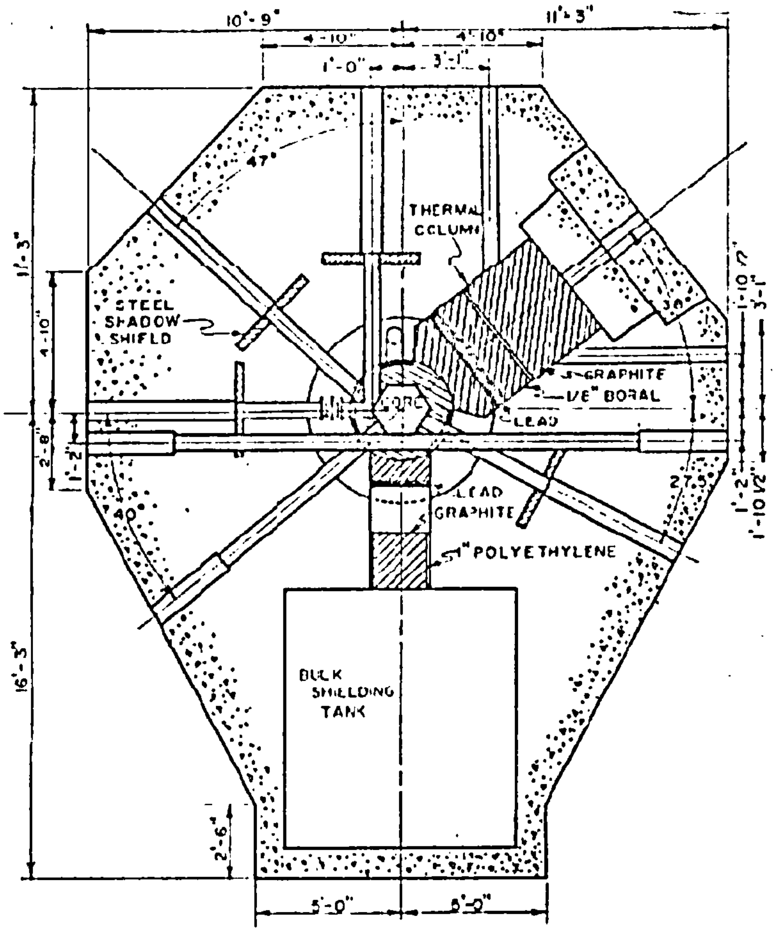


Figure A-20
Reactor Section at Beam Port Level

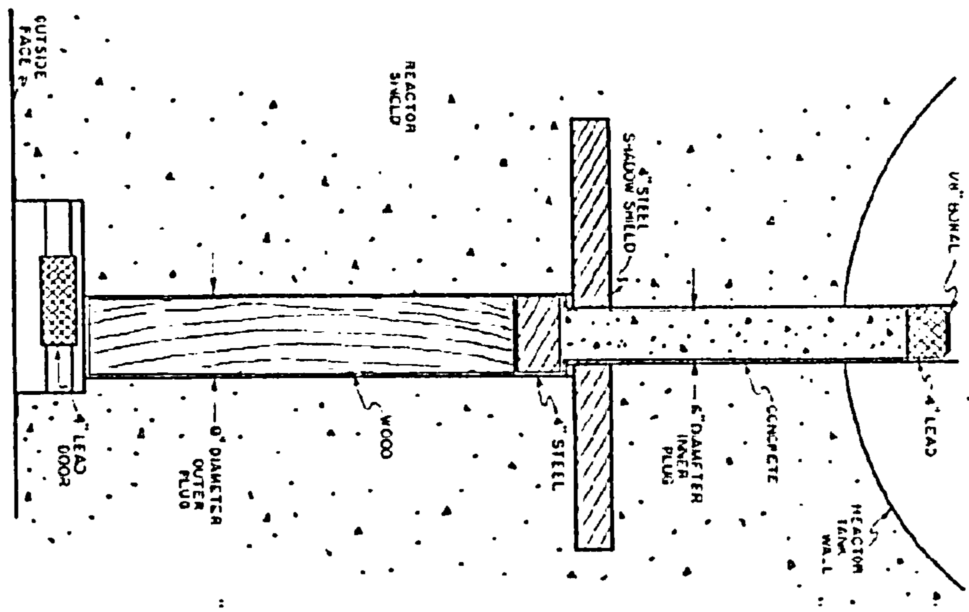


Figure A-21
Horizontal Section of a Beam Tube

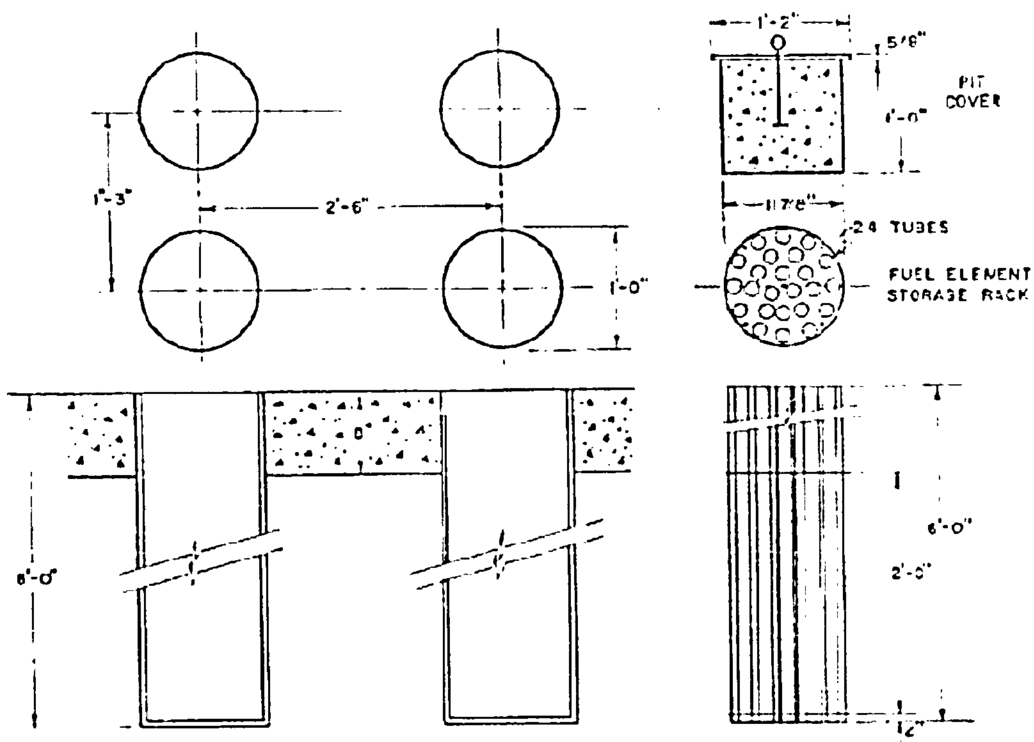


Figure A-22
Dry Fuel Storage Pits

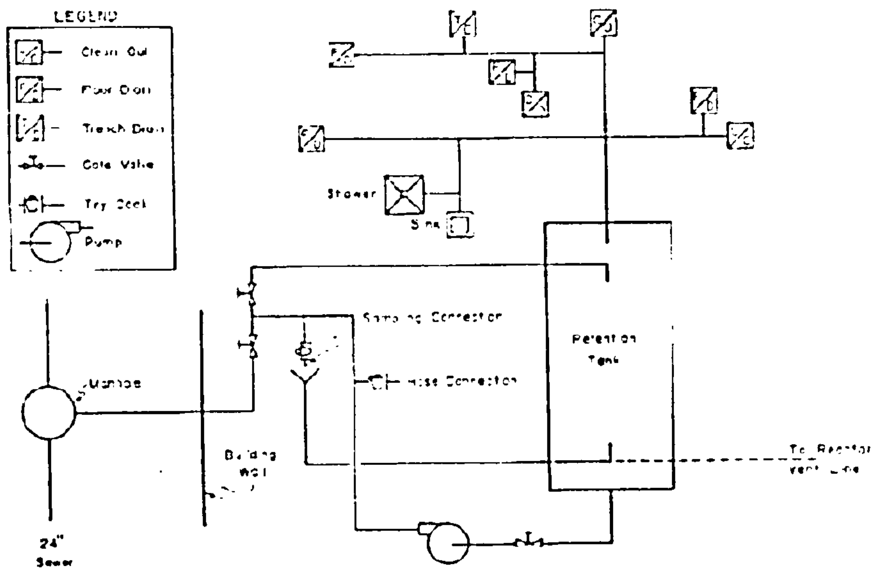


Figure A-23
Water Retention System

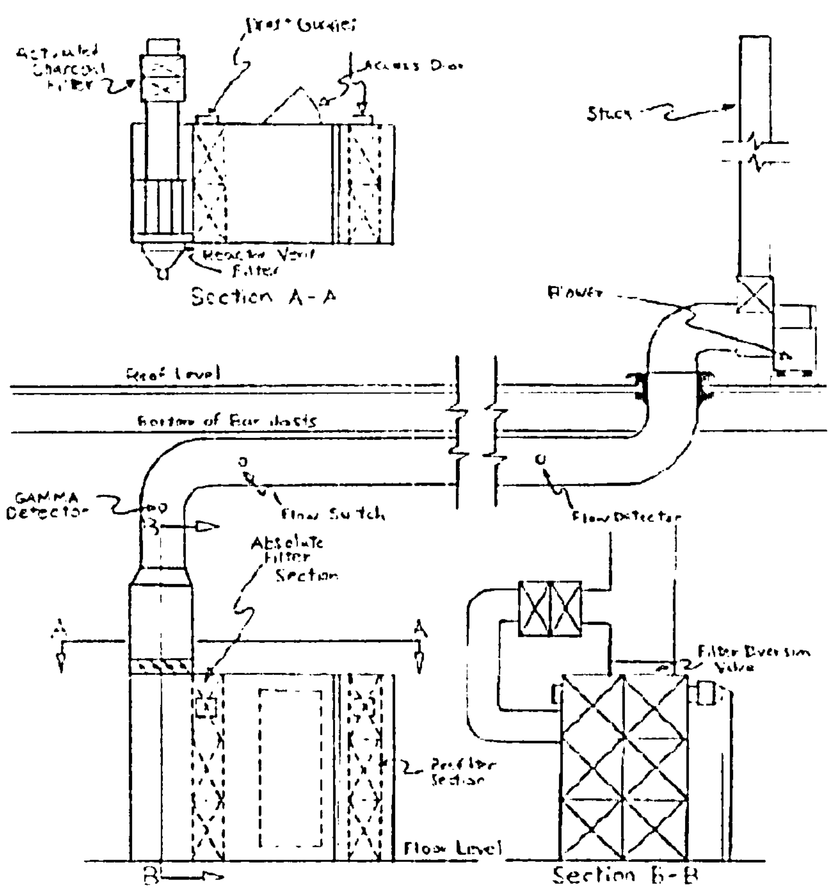


Figure A-24
Building Exhaust System

APPENDIX B

DOSE ASSESSMENT FOR THE DEVELOPMENT OF DCGLS

NUCLEAR RESEARCH LABORATORY
UNIVERSITY OF ILLINOIS AT URBANA-CHAMPAIGN

University of Illinois Decommissioning Plan
Dose Assessment for the Development of DCGLs

The decommissioning of the University of Illinois research reactor will involve the demolition of the reactor facility and the free-release of the building foot-print. As such, release criteria are needed for contaminants of concern that may have impacted the soil under or surrounding the reactor. The Decommissioning Plan proposes to use both site-specific derived concentration guideline levels (DCGL) and published NRC soil screening values for surface soils (NRC 2002) as release criteria. The site characterization (Scientech 2005) identified the following as the primary radionuclides of concern: hydrogen-3 (tritium), carbon-14, iron-55, cobalt-60, and europium-152 while nickel-63, cesium-137, and europium-154 were determined to be less likely environmental contaminates of concern.

The potential reuse options for the reactor site are limited due to its size and location. The site is small, with a building foot print of 80 by 45 feet [334.5 square meters (m^2)], and is surrounded by academic buildings to the north, south, and west and a parking area to the east. Because of its location, only an industrial use scenario was assumed for potential future on-site populations.

To evaluate the exposures to potential contaminants of concern, RESRAD Version 6.3 was used to model the industrial exposure scenario. To construct the dose model, all exposure pathways except external exposure, soil ingestion, and inhalation were turned off. Because soil ingestion, inhalation, and external exposure are the only pathways, the list of contaminants of concern for on-site exposures is limited to primarily gamma-emitting and alpha-emitting radionuclides. Because there are no alpha-emitting contaminants of concern, only the gamma-emitting cobalt-60 and europium-152 were considered as on-site contaminants of concern. As such, site specific DCGLs were only developed for these two radionuclides while the soil screening criteria were used for the other radionuclides. The radionuclides for which the soil screening criteria are applied show insignificant dose consequence in an industrial exposure scenario using RESRAD.

For the on-site industrial exposure model, the area of contamination was assumed to be equal to the footprint of the building, 334.5 m^2. The thickness of the contaminated layer was assumed to equal 15 centimeters (cm). Following the decommissioning project, the building footprint, which is below grade, will be backfilled with clean soil. However, for the exposure scenario, it is assumed that the residually contaminated soil is later excavated and brought back to the ground surface and a building is placed on top of the excavated soil. Therefore, no clean soil cover is assumed for the contaminated area.

The Table B-1 provides many of the site specific and default exposure parameters used in the RESRAD dose assessment.

TABLE B-1
RESRAD PARAMETERS

Parameter	Value
Transport Factors	
Time since material placement	0 years (default)
Solubility limit	0 mol/L (default)
Leach Rate	0 /year (default)
Contaminated Zone Parameters	
Area	334.5 m^2
Thickness	0.15 m
Cover Depth	0 m [a]
Density of contaminated zone	1.5 g/cm^3 (default)
Erosion rate	0.0 m/year [b]
Occupancy Factors	
Inhalation rate	8,4000 m^3/year (default)
Exposure duration	30 years (default)
Indoor dust filtration factor	0.4 (default)
External gamma shielding factor	0.7 (default)
Indoor time fraction	0.18 (8 hrs/day 200 days/year)
Outdoor time fraction	0.046 (2 hrs/day 200 days/year)

Notes:

a The contaminated soil is directly below the building with no cover. Shielding from the building floor is accounted for in the Occupancy Factors.

b A zero erosion rate is assumed for the soil below the building.

10 CFR 20, Subpart E requires that an NRC-licensee demonstrate that the potential maximum dose from residual contamination to a future occupant of the licensed facility be less than 25 millirem per year (mrem/yr) and as low as reasonably achievable (ALARA). The ALARA requirement will be met by the successful completion of the Decommissioning Plan and removal of all activated and contaminated materials from the site at a significant cost as provided in the Decommissioning Plan. An ALARA analysis conducted according to Volume 2 of NUREG-1757 (NRC 2002) shows that, because of the high decommissioning costs, the DCGLs are orders of magnitude below the ALARA concentrations.

Table B-2 provides the cobalt-60 and europium-152 DCGLs, the soil concentrations that result in a maximum of 25 mrem/yr dose to the exposed individual as determined in the RESRAD model. The RESRAD output files are provided after this appendix.

TABLE B-2
DERIVED CONCENTRATION GUIDELINE LEVELS

Radionuclide	Concentration (pCi/g)	Maximum Dose (mrem/yr)	Time of Maximum Dose (years)
Cobalt-60	12.9	24.9	0
Europium-152 Gadolinium-152	27.9	25.0	0

For radionuclides of concern other than cobalt-60 and europium-152, the DCGLs are set equal to the NRC soil screening values for surface soils (NRC 2002) as provided in Table B-3. The screening values are equivalent to 25 mrem/yr as derived using the NRC's DandD dose modeling code and the screening methodology in NUREG-5512 (NRC 1992).

TABLE B-3
SOIL SCREENING CRITERIA

Radionuclide	Screening Value DCGL (pCi/g)
Tritium (hydrogen-3)	110
Carbon-14	120
Iron-55	10,000
Nickel-63	2,100
Cesium-137	11
Europium-154	8.0

Each of the DCGLs and the soil screening criteria correspond to a dose to the exposed individual of 25 mrem/yr. As such, if multiple contaminants are identified in the soil being released as part of the final status survey, the sum of the fractions rule will apply. Furthermore, the minimum detectable concentrations (MDC) of the analytical methods should be such at the sum of the fractions of the MDCs to the DCGLs are not greater than 1. Some isotopes may, however, later be eliminated as potential residual soil contaminants when additional soil characterization is performed at the time of the facility decommissioning. Limited soil data indicate only the potential for elevated tritium and cobalt-60 in the soil under the reactor bay floor.

REFERENCES

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NRC. 2002. "Consolidated NMSS Decommissioning Guidance, Decommissioning Process for Materials Licenses." Volume 2. NUREG-1757. September.

Scientech, LLC. 2005b. "Site Characterization Report – Nuclear Research Laboratory, University of Illinois at Champaign-Urbana." Document No. 82A9571. October 6.

Table of Contents

·rt I: Mixture Sums and Single Radionuclide Guidelines

Dose Conversion Factor (and Related) Parameter Summary
File: FGR 13 MORBIDITY

Menu	Parameter	Current Value	Base Case*	Parameter Name
B-1	Dose conversion factors for inhalation, mrem/pCi:			
B-1	Co-60	2.190E-04	2.190E-04	DCF2(1)
D-1	Dose conversion factors for ingestion, mrem/pCi:			
D-1	Co-60	2.690E-05	2.690E-05	DCF3(1)
D-34	Food transfer factors:			
D-34	Co-60 , plant/soil concentration ratio, dimensionless	8.000E-02	8.000E-02	RTF(1,1)
D-34	Co-60 , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	2.000E-02	2.000E-02	RTF(1,2)
D-34	Co-60 , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	2.000E-03	2.000E-03	RTF(1,3)
D-5	Bioaccumulation factors, fresh water, L/kg:			
D-5	Co-60 , fish	3.000E+02	3.000E+02	BIOFAC(1,1)
D-5	Co-60 , crustacea and mollusks	2.000E+02	2.000E+02	BIOFAC(1,2)

*Base Case means Default.Lib w/o Associate Nuclide contributions.

Site-Specific Parameter Summary

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R011	Area of contaminated zone (m**2)	3.345E+02	1.000E+04	---	AREA
R011	Thickness of contaminated zone (m)	1.500E-01	2.000E+00	---	THICK0
R011	Length parallel to aquifer flow (m)	not used	1.000E+02	---	LCZPAQ
R011	Basic radiation dose limit (mrem/yr)	2.500E+01	3.000E+01	---	BRDL
R011	Time since placement of material (yr)	0.000E+00	0.000E+00	---	TI
R011	Times for calculations (yr)	1.000E+00	1.000E+00	---	T(2)
R011	Times for calculations (yr)	3.000E+00	3.000E+00	---	T(3)
R011	Times for calculations (yr)	1.000E+01	1.000E+01	---	T(4)
R011	Times for calculations (yr)	3.000E+01	3.000E+01	---	T(5)
R011	Times for calculations (yr)	1.000E+02	1.000E+02	---	T(6)
R011	Times for calculations (yr)	3.000E+02	3.000E+02	---	T(7)
R011	Times for calculations (yr)	1.000E+03	1.000E+03	---	T(8)
R011	Times for calculations (yr)	not used	0.000E+00	---	T(9)
R011	Times for calculations (yr)	not used	0.000E+00	---	T(10)
R012	Initial principal radionuclide (pCi/g): Co-60	1.290E+01	0.000E+00	---	S1(1)
R012	Concentration in groundwater (pCi/L): Co-60	not used	0.000E+00	---	W1(1)
R013	Cover depth (m)	0.000E+00	0.000E+00	---	COVER0
R013	Density of cover material (g/cm**3)	not used	1.500E+00	---	DENSCV
R013	Cover depth erosion rate (m/yr)	not used	1.000E-03	---	VCV
R013	Density of contaminated zone (g/cm**3)	1.500E+00	1.500E+00	---	DENSCZ
R013	Contaminated zone erosion rate (m/yr)	0.000E+00	1.000E-03	---	VCZ
R013	Contaminated zone total porosity	4.000E-01	4.000E-01	---	TPCZ
	Contaminated zone field capacity	2.000E-01	2.000E-01	---	FCCZ
R013	Contaminated zone hydraulic conductivity (m/yr)	1.000E+01	1.000E+01	---	HCCZ
R013	Contaminated zone b parameter	5.300E+00	5.300E+00	---	BCZ
R013	Average annual wind speed (m/sec)	2.000E+00	2.000E+00	---	WIND
R013	Humidity in air (g/m**3)	not used	8.000E+00	---	HUMID
R013	Evapotranspiration coefficient	5.000E-01	5.000E-01	---	EVAPTR
R013	Precipitation (m/yr)	1.000E+00	1.000E+00	---	PRECIP
R013	Irrigation (m/yr)	2.000E-01	2.000E-01	---	RI
R013	Irrigation mode	overhead	overhead	---	IDITCH
R013	Runoff coefficient	2.000E-01	2.000E-01	---	RUNOFF
R013	Watershed area for nearby stream or pond (m**2)	not used	1.000E+06	---	WAREA
R013	Accuracy for water/soil computations	not used	1.000E-03	---	EPS
R014	Density of saturated zone (g/cm**3)	not used	1.500E+00	---	DENSAQ
R014	Saturated zone total porosity	not used	4.000E-01	---	TPSZ
R014	Saturated zone effective porosity	not used	2.000E-01	---	EPSZ
R014	Saturated zone field capacity	not used	2.000E-01	---	FCSZ
R014	Saturated zone hydraulic conductivity (m/yr)	not used	1.000E+02	---	HCSZ
R014	Saturated zone hydraulic gradient	not used	2.000E-02	---	HGWT
R014	Saturated zone b parameter	not used	5.300E+00	---	BSZ
R014	Water table drop rate (m/yr)	not used	1.000E-03	---	VWT
R014	Well pump intake depth (m below water table)	not used	1.000E+01	---	DWIBWT
R014	Model: Nondispersion (ND) or Mass-Balance (MB)	not used	ND	---	MODEL
R014	Well pumping rate (m**3/yr)	not used	2.500E+02	---	UW
R015	Number of unsaturated zone strata	not used	1	---	NS

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R015	Unsat. zone 1, thickness (m)	not used	4.000E+00	---	H(1)
R015	Unsat. zone 1, soil density (g/cm**3)	not used	1.500E+00	---	DENSUZ(1)
R015	Unsat. zone 1, total porosity	not used	4.000E-01	---	TPUZ(1)
R015	Unsat. zone 1, effective porosity	not used	2.000E-01	---	EPUZ(1)
R015	Unsat. zone 1, field capacity	not used	2.000E-01	---	FCUZ(1)
R015	Unsat. zone 1, soil-specific b parameter	not used	5.300E+00	---	BUZ(1)
R015	Unsat. zone 1, hydraulic conductivity (m/yr)	not used	1.000E+01	---	HCUZ(1)
R016	Distribution coefficients for Co-60				
R016	Contaminated zone (cm**3/g)	1.000E+03	1.000E+03	---	DCNUCC(1)
R016	Unsaturated zone 1 (cm**3/g)	not used	1.000E+03	---	DCNUCU(1,1)
R016	Saturated zone (cm**3/g)	not used	1.000E+03	---	DCNUCS(1)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.222E-03	ALEACH(1)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(1)
R017	Inhalation rate (m**3/yr)	8.400E+03	8.400E+03	---	INHALR
R017	Mass loading for inhalation (g/m**3)	1.000E-04	1.000E-04	---	MLINH
R017	Exposure duration	3.000E+01	3.000E+01	---	ED
R017	Shielding factor, inhalation	4.000E-01	4.000E-01	---	SHF3
R017	Shielding factor, external gamma	7.000E-01	7.000E-01	---	SHF1
R017	Fraction of time spent indoors	1.800E-01	5.000E-01	---	FIND
R017	Fraction of time spent outdoors (on site)	4.600E-02	2.500E-01	---	FOTD
R017	Shape factor flag, external gamma	1.000E+00	1.000E+00	>0 shows circular AREA.	FS
R017	Radii of shape factor array (used if FS = -1):				
R017	Outer annular radius (m), ring 1:	not used	5.000E+01	---	RAD_SHAPE(1)
R017	Outer annular radius (m), ring 2:	not used	7.071E+01	---	RAD_SHAPE(2)
R017	Outer annular radius (m), ring 3:	not used	0.000E+00	---	RAD_SHAPE(3)
R017	Outer annular radius (m), ring 4:	not used	0.000E+00	---	RAD_SHAPE(4)
R017	Outer annular radius (m), ring 5:	not used	0.000E+00	---	RAD_SHAPE(5)
R017	Outer annular radius (m), ring 6:	not used	0.000E+00	---	RAD_SHAPE(6)
R017	Outer annular radius (m), ring 7:	not used	0.000E+00	---	RAD_SHAPE(7)
R017	Outer annular radius (m), ring 8:	not used	0.000E+00	---	RAD_SHAPE(8)
R017	Outer annular radius (m), ring 9:	not used	0.000E+00	---	RAD_SHAPE(9)
R017	Outer annular radius (m), ring 10:	not used	0.000E+00	---	RAD_SHAPE(10)
R017	Outer annular radius (m), ring 11:	not used	0.000E+00	---	RAD_SHAPE(11)
R017	Outer annular radius (m), ring 12:	not used	0.000E+00	---	RAD_SHAPE(12)
R017	Fractions of annular areas within AREA:				
R017	Ring 1	not used	1.000E+00	---	FRACA(1)
R017	Ring 2	not used	2.732E-01	---	FRACA(2)
R017	Ring 3	not used	0.000E+00	---	FRACA(3)
R017	Ring 4	not used	0.000E+00	---	FRACA(4)
R017	Ring 5	not used	0.000E+00	---	FRACA(5)
R017	Ring 6	not used	0.000E+00	---	FRACA(6)
R017	Ring 7	not used	0.000E+00	---	FRACA(7)
R017	Ring 8	not used	0.000E+00	---	FRACA(8)
R017	Ring 9	not used	0.000E+00	---	FRACA(9)
R017	Ring 10	not used	0.000E+00	---	FRACA(10)
R017	Ring 11	not used	0.000E+00	---	FRACA(11)
R017	Ring 12	not used	0.000E+00	---	FRACA(12)

Site-Specific Parameter Summary (continued)

	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R018	Fruits, vegetables and grain consumption (kg/yr)	not used	1.600E+02	---	DIET(1)
R018	Leafy vegetable consumption (kg/yr)	not used	1.400E+01	---	DIET(2)
R018	Milk consumption (L/yr)	not used	9.200E+01	---	DIET(3)
R018	Meat and poultry consumption (kg/yr)	not used	6.300E+01	---	DIET(4)
R018	Fish consumption (kg/yr)	not used	5.400E+00	---	DIET(5)
R018	Other seafood consumption (kg/yr)	not used	9.000E-01	---	DIET(6)
R018	Soil ingestion rate (g/yr)	3.650E+01	3.650E+01	---	SOIL
R018	Drinking water intake (L/yr)	not used	5.100E+02	---	DWI
R018	Contamination fraction of drinking water	not used	1.000E+00	---	FDW
R018	Contamination fraction of household water	not used	1.000E+00	---	FHHW
R018	Contamination fraction of livestock water	not used	1.000E+00	---	FLW
R018	Contamination fraction of irrigation water	not used	1.000E+00	---	FIRW
R018	Contamination fraction of aquatic food	not used	5.000E-01	---	FR9
R018	Contamination fraction of plant food	not used	-1	---	FPLANT
R018	Contamination fraction of meat	not used	-1	---	FMEAT
R018	Contamination fraction of milk	not used	-1	---	FMILK
R019	Livestock fodder intake for meat (kg/day)	not used	6.800E+01	---	LFI5
R019	Livestock fodder intake for milk (kg/day)	not used	5.500E+01	---	LFI6
R019	Livestock water intake for meat (L/day)	not used	5.000E+01	---	LWI5
R019	Livestock water intake for milk (L/day)	not used	1.600E+02	---	LWI6
R019	Livestock soil intake (kg/day)	not used	5.000E-01	---	LSI
R019	Mass loading for foliar deposition (g/m**3)	not used	1.000E-04	---	MLFD
R019	Depth of soil mixing layer (m)	1.500E-01	1.500E-01	---	DM
R019	Depth of roots (m)	not used	9.000E-01	---	DROOT
[illegible]	Drinking water fraction from ground water	not used	1.000E+00	---	FGWDW
R019	Household water fraction from ground water	not used	1.000E+00	---	FGWHH
R019	Livestock water fraction from ground water	not used	1.000E+00	---	FGWLW
R019	Irrigation fraction from ground water	not used	1.000E+00	---	FGWIR
R19B	Wet weight crop yield for Non-Leafy (kg/m**2)	not used	7.000E-01	---	YV(1)
R19B	Wet weight crop yield for Leafy (kg/m**2)	not used	1.500E+00	---	YV(2)
R19B	Wet weight crop yield for Fodder (kg/m**2)	not used	1.100E+00	---	YV(3)
R19B	Growing Season for Non-Leafy (years)	not used	1.700E-01	---	TE(1)
R19B	Growing Season for Leafy (years)	not used	2.500E-01	---	TE(2)
R19B	Growing Season for Fodder (years)	not used	8.000E-02	---	TE(3)
R19B	Translocation Factor for Non-Leafy	not used	1.000E-01	---	TIV(1)
R19B	Translocation Factor for Leafy	not used	1.000E+00	---	TIV(2)
R19B	Translocation Factor for Fodder	not used	1.000E+00	---	TIV(3)
R19B	Dry Foliar Interception Fraction for Non-Leafy	not used	2.500E-01	---	RDRY(1)
R19B	Dry Foliar Interception Fraction for Leafy	not used	2.500E-01	---	RDRY(2)
R19B	Dry Foliar Interception Fraction for Fodder	not used	2.500E-01	---	RDRY(3)
R19B	Wet Foliar Interception Fraction for Non-Leafy	not used	2.500E-01	---	RWET(1)
R19B	Wet Foliar Interception Fraction for Leafy	not used	2.500E-01	---	RWET(2)
R19B	Wet Foliar Interception Fraction for Fodder	not used	2.500E-01	---	RWET(3)
R19B	Weathering Removal Constant for Vegetation	not used	2.000E+01	---	WLAM
C14	C-12 concentration in water (g/cm**3)	not used	2.000E-05	---	C12WTR
C14	C-12 concentration in contaminated soil (g/g)	not used	3.000E-02	---	C12CZ
C14	Fraction of vegetation carbon from soil	not used	2.000E-02	---	CSOIL
C14	Fraction of vegetation carbon from air	not used	9.800E-01	---	CAIR

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
C14	C-14 evasion layer thickness in soil (m)	not used	3.000E-01	---	DMC
C14	C-14 evasion flux rate from soil (1/sec)	not used	7.000E-07	---	EVSN
C14	C-12 evasion flux rate from soil (1/sec)	not used	1.000E-10	---	REVSN
C14	Fraction of grain in beef cattle feed	not used	8.000E-01	---	AVFG4
C14	Fraction of grain in milk cow feed	not used	2.000E-01	---	AVFG5
C14	DCF correction factor for gaseous forms of C14	not used	0.000E+00	---	CO2F
STOR	Storage times of contaminated foodstuffs (days):				
STOR	Fruits, non-leafy vegetables, and grain	1.400E+01	1.400E+01	---	STOR_T(1)
STOR	Leafy vegetables	1.000E+00	1.000E+00	---	STOR_T(2)
STOR	Milk	1.000E+00	1.000E+00	---	STOR_T(3)
STOR	Meat and poultry	2.000E+01	2.000E+01	---	STOR_T(4)
STOR	Fish	7.000E+00	7.000E+00	---	STOR_T(5)
STOR	Crustacea and mollusks	7.000E+00	7.000E+00	---	STOR_T(6)
STOR	Well water	1.000E+00	1.000E+00	---	STOR_T(7)
STOR	Surface water	1.000E+00	1.000E+00	---	STOR_T(8)
STOR	Livestock fodder	4.500E+01	4.500E+01	---	STOR_T(9)
R021	Thickness of building foundation (m)	not used	1.500E-01	---	FLOOR1
R021	Bulk density of building foundation (g/cm**3)	not used	2.400E+00	---	DENSFL
R021	Total porosity of the cover material	not used	4.000E-01	---	TPCV
R021	Total porosity of the building foundation	not used	1.000E-01	---	TPFL
R021	Volumetric water content of the cover material	not used	5.000E-02	---	PH2OCV
R021	Volumetric water content of the foundation	not used	3.000E-02	---	PH2OFL
R021	Diffusion coefficient for radon gas (m/sec):				
R021	in cover material	not used	2.000E-06	---	DIFCV
R021	in foundation material	not used	3.000E-07	---	DIFFL
R021	in contaminated zone soil	not used	2.000E-06	---	DIFCZ
R021	Radon vertical dimension of mixing (m)	not used	2.000E+00	---	HMIX
R021	Average building air exchange rate (1/hr)	not used	5.000E-01	---	REXG
R021	Height of the building (room) (m)	not used	2.500E+00	---	HRM
R021	Building interior area factor	not used	0.000E+00	---	FAI
R021	Building depth below ground surface (m)	not used	-1.000E+00	---	DMFL
R021	Emanating power of Rn-222 gas	not used	2.500E-01	---	EMANA(1)
R021	Emanating power of Rn-220 gas	not used	1.500E-01	---	EMANA(2)
TITL	Number of graphical time points	32	---	---	NPTS
TITL	Maximum number of integration points for dose	17	---	---	LYMAX
TITL	Maximum number of integration points for risk	257	---	---	KYMAX

Summary of Pathway Selections

Pathway	User Selection
1 -- external gamma	active
2 -- inhalation (w/o radon)	active
3 -- plant ingestion	suppressed
4 -- meat ingestion	suppressed
5 -- milk ingestion	suppressed
6 -- aquatic foods	suppressed
7 -- drinking water	suppressed
8 -- soil ingestion	active
9 -- radon	suppressed
Find peak pathway doses	suppressed

Contaminated Zone Dimensions		Initial Soil Concentrations, pCi/g	
Area:	334.50 square meters	Co-60	1.290E+01
Thickness:	0.15 meters		
Cover Depth:	0.00 meters		

Total Dose TDOSE(t), mrem/yr
Basic Radiation Dose Limit = 2.500E+01 mrem/yr
Total Mixture Sum M(t) = Fraction of Basic Dose Limit Received at Time (t)

t (years):	0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
TDOSE(t):	2.493E+01	2.181E+01	1.669E+01	6.547E+00	4.514E-01	3.884E-05	9.422E-17	0.000E+00
M(t):	9.974E-01	8.725E-01	6.678E-01	2.619E-01	1.806E-02	1.554E-06	3.769E-18	0.000E+00

Maximum TDOSE(t): 2.493E+01 mrem/yr at t = 0.000E+00 years

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 0.000E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	2.493E+01	1.0000	3.114E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.962E-04	0.000[illegible]
Total	2.493E+01	1.0000	3.114E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.962E-04	0.000[illegible]

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 0.000E+00 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.493E+01	1.000[illegible]
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.493E+01	1.000[illegible]

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	2.181E+01	1.0000	2.724E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	7.841E-04	0.000
Total	2.181E+01	1.0000	2.724E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	7.841E-04	0.000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+00 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.181E+01	1.000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.181E+01	1.000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract
Co-60	1.669E+01	1.0000	2.085E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.001E-04	0.000
Total	1.669E+01	1.0000	2.085E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.001E-04	0.000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+00 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.669E+01	1.000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.669E+01	1.000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+01 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	6.547E+00	1.0000	8.177E-06	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.353E-04	0.0000
Total	6.547E+00	1.0000	8.177E-06	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.353E-04	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+01 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.547E+00	1.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.547E+00	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+01 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	4.514E-01	1.0000	5.637E-07	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.622E-05	0.000(
Total	4.514E-01	1.0000	5.637E-07	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.622E-05	0.000(

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+01 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.514E-01	1.000(
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.514E-01	1.000(

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+02 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	3.884E-05	1.0000	4.850E-11	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.396E-09	0.0000
Total	3.884E-05	1.0000	4.850E-11	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.396E-09	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+02 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.884E-05	1.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.884E-05	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+02 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	9.422E-17	1.0000	1.177E-22	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.387E-21	0.0000
Total	9.422E-17	1.0000	1.177E-22	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.387E-21	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+02 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.422E-17	1.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.422E-17	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+03 years

Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground mrem/yr	Ground fract.	Inhalation mrem/yr	Inhalation fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	Soil mrem/yr	Soil fract.
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+03 years

Water Dependent Pathways

Radio-Nuclide	Water mrem/yr	Water fract.	Fish mrem/yr	Fish fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	All Pathways* mrem/yr	All Pathways* fract.
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

*Sum of all water independent and dependent pathways.

Dose/Source Ratios Summed Over All Pathways
Parent and Progeny Principal Radionuclide Contributions Indicated

Parent (i)	Product (j)	Thread Fraction	DSR(j,t) At Time in Years (mrem/yr)/(pCi/g) 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
Co-60	Co-60	1.000E+00	1.933E+00	1.691E+00	1.294E+00	5.075E-01	3.499E-02	3.011E-06	7.304E-18	0.000E+00

The DSR includes contributions from associated (half-life ≤ 180 days) daughters.

Single Radionuclide Soil Guidelines G(i,t) in pCi/g
Basic Radiation Dose Limit = 2.500E+01 mrem/yr

Nuclide (i)	t= 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
Co-60	1.293E+01	1.478E+01	1.932E+01	4.926E+01	7.145E+02	8.304E+06	*1.132E+15	*1.132E+15

*At specific activity limit

Summed Dose/Source Ratios DSR(i,t) in (mrem/yr)/(pCi/g)
and Single Radionuclide Soil Guidelines G(i,t) in pCi/g
at tmin = time of minimum single radionuclide soil guideline
and at tmax = time of maximum total dose = 0.000E+00 years

Nuclide (i)	Initial (pCi/g)	tmin (years)	DSR(i,tmin)	G(i,tmin) (pCi/g)	DSR(i,tmax)	G(i,tmax) (pCi/g)
Co-60	1.290E+01	0.000E+00	1.933E+00	1.293E+01	1.933E+00	1.293E+01

Individual Nuclide Dose Summed Over All Pathways
Parent Nuclide and Branch Fraction Indicated

Nuclide (j)	Parent (i)	THF(i)	DOSE(j,t), mrem/yr t= 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
Co-60	Co-60	1.000E+00	2.493E+01	2.181E+01	1.669E+01	6.547E+00	4.514E-01	3.884E-05	9.422E-17	0.000E+00

THF(i) is the thread fraction of the parent nuclide.

Individual Nuclide Soil Concentration
Parent Nuclide and Branch Fraction Indicated

Nuclide (j)	Parent (i)	THF(i)	S(j,t), pCi/g t= 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
Co-60	Co-60	1.000E+00	1.290E+01	1.129E+01	8.637E+00	3.397E+00	2.335E-01	2.009E-05	4.875E-17	0.000E+00

THF(i) is the thread fraction of the parent nuclide.

RESCALC.EXE execution time = 6.69 seconds

Table of Contents

Part I: Mixture Sums and Single Radionuclide Guidelines

Dose Conversion Factor (and Related) Parameter Summary
File: FGR 13 MORBIDITY

Menu	Parameter	Current Value	Base Case*	Parameter Name
B-1	Dose conversion factors for inhalation, mrem/pCi:			
B-1	Eu-152	2.210E-04	2.210E-04	DCF2(1)
B-1	Gd-152	2.430E-01	2.430E-01	DCF2(3)
D-1	Dose conversion factors for ingestion, mrem/pCi:			
D-1	Eu-152	6.480E-06	6.480E-06	DCF3(1)
D-1	Gd-152	1.610E-04	1.610E-04	DCF3(3)
D-34	Food transfer factors:			
D-34	Eu-152 , plant/soil concentration ratio, dimensionless	2.500E-03	2.500E-03	RTF(1,1)
D-34	Eu-152 , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	2.000E-03	2.000E-03	RTF(1,2)
D-34	Eu-152 , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	5.000E-05	5.000E-05	RTF(1,3)
D-34				
D-34	Gd-152 , plant/soil concentration ratio, dimensionless	2.500E-03	2.500E-03	RTF(3,1)
D-34	Gd-152 , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	2.000E-03	2.000E-03	RTF(3,2)
D-34	Gd-152 , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	2.000E-05	2.000E-05	RTF(3,3)
D-5	Bioaccumulation factors, fresh water, L/kg:			
D-5	Eu-152 , fish	5.000E+01	5.000E+01	BIOFAC(1,1)
D-5	Eu-152 , crustacea and mollusks	1.000E+03	1.000E+03	BIOFAC(1,2)
D-5				
D-5	Gd-152 , fish	2.500E+01	2.500E+01	BIOFAC(3,1)
D-5	Gd-152 , crustacea and mollusks	1.000E+03	1.000E+03	BIOFAC(3,2)

*Base Case means Default.Lib w/o Associate Nuclide contributions.

Site-Specific Parameter Summary

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R011	Area of contaminated zone (m**2)	3.345E+02	1.000E+04	---	AREA
R011	Thickness of contaminated zone (m)	1.500E-01	2.000E+00	---	THICK0
R011	Length parallel to aquifer flow (m)	not used	1.000E+02	---	LCZPAQ
R011	Basic radiation dose limit (mrem/yr)	2.500E+01	3.000E+01	---	BRDL
R011	Time since placement of material (yr)	0.000E+00	0.000E+00	---	TI
R011	Times for calculations (yr)	1.000E+00	1.000E+00	---	T(2)
R011	Times for calculations (yr)	3.000E+00	3.000E+00	---	T(3)
R011	Times for calculations (yr)	1.000E+01	1.000E+01	---	T(4)
R011	Times for calculations (yr)	3.000E+01	3.000E+01	---	T(5)
R011	Times for calculations (yr)	1.000E+02	1.000E+02	---	T(6)
R011	Times for calculations (yr)	3.000E+02	3.000E+02	---	T(7)
R011	Times for calculations (yr)	1.000E+03	1.000E+03	---	T(8)
R011	Times for calculations (yr)	not used	0.000E+00	---	T(9)
R011	Times for calculations (yr)	not used	0.000E+00	---	T(10)
R012	Initial principal radionuclide (pCi/g): Eu-152	2.790E+01	0.000E+00	---	S1(1)
R012	Initial principal radionuclide (pCi/g): Gd-152	2.790E+01	0.000E+00	---	S1(3)
R012	Concentration in groundwater (pCi/L): Eu-152	not used	0.000E+00	---	W1(1)
R012	Concentration in groundwater (pCi/L): Gd-152	not used	0.000E+00	---	W1(3)
R013	Cover depth (m)	0.000E+00	0.000E+00	---	COVER0
R013	Density of cover material (g/cm**3)	not used	1.500E+00	---	DENSCV
R013	Cover depth erosion rate (m/yr)	not used	1.000E-03	---	VCV
R013	Density of contaminated zone (g/cm**3)	1.500E+00	1.500E+00	---	DENSCZ
R013	Contaminated zone erosion rate (m/yr)	0.000E+00	1.000E-03	---	VCZ
R013	Contaminated zone total porosity	4.000E-01	4.000E-01	---	TPCZ
R013	Contaminated zone field capacity	2.000E-01	2.000E-01	---	FCCZ
R013	Contaminated zone hydraulic conductivity (m/yr)	1.000E+01	1.000E+01	---	HCCZ
R013	Contaminated zone b parameter	5.300E+00	5.300E+00	---	BCZ
R013	Average annual wind speed (m/sec)	2.000E+00	2.000E+00	---	WIND
R013	Humidity in air (g/m**3)	not used	8.000E+00	---	HUMID
R013	Evapotranspiration coefficient	5.000E-01	5.000E-01	---	EVAPTR
R013	Precipitation (m/yr)	1.000E+00	1.000E+00	---	PRECIP
R013	Irrigation (m/yr)	2.000E-01	2.000E-01	---	RI
R013	Irrigation mode	overhead	overhead	---	IDITCH
R013	Runoff coefficient	2.000E-01	2.000E-01	---	RUNOFF
R013	Watershed area for nearby stream or pond (m**2)	not used	1.000E+06	---	WAREA
R013	Accuracy for water/soil computations	not used	1.000E-03	---	EPS
R014	Density of saturated zone (g/cm**3)	not used	1.500E+00	---	DENSAQ
R014	Saturated zone total porosity	not used	4.000E-01	---	TPSZ
R014	Saturated zone effective porosity	not used	2.000E-01	---	EPSZ
R014	Saturated zone field capacity	not used	2.000E-01	---	FCSZ
R014	Saturated zone hydraulic conductivity (m/yr)	not used	1.000E+02	---	HCSZ
R014	Saturated zone hydraulic gradient	not used	2.000E-02	---	HGWT
R014	Saturated zone b parameter	not used	5.300E+00	---	BSZ
R014	Water table drop rate (m/yr)	not used	1.000E-03	---	VWT
R014	Well pump intake depth (m below water table)	not used	1.000E+01	---	DWIBWT
R014	Model: Nondispersion (ND) or Mass-Balance (MB)	not used	ND	---	MODEL
R014	Well pumping rate (m**3/yr)	not used	2.500E+02	---	UW

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R015	Number of unsaturated zone strata	not used	1	---	NS
R015	Unsat. zone 1, thickness (m)	not used	4.000E+00	---	H(1)
R015	Unsat. zone 1, soil density (g/cm**3)	not used	1.500E+00	---	DENSUZ(1)
R015	Unsat. zone 1, total porosity	not used	4.000E-01	---	TPUZ(1)
R015	Unsat. zone 1, effective porosity	not used	2.000E-01	---	EPUZ(1)
R015	Unsat. zone 1, field capacity	not used	2.000E-01	---	FCUZ(1)
R015	Unsat. zone 1, soil-specific b parameter	not used	5.300E+00	---	BUZ(1)
R015	Unsat. zone 1, hydraulic conductivity (m/yr)	not used	1.000E+01	---	HCUZ(1)
R016	Distribution coefficients for Eu-152				
R016	Contaminated zone (cm**3/g)	-1.000E+00	-1.000E+00	8.249E+02	DCNUCC(1)
R016	Unsaturated zone 1 (cm**3/g)	not used	-1.000E+00	---	DCNUCU(1,1)
R016	Saturated zone (cm**3/g)	not used	-1.000E+00	---	DCNUCS(1)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.693E-03	ALEACH(1)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(1)
R016	Distribution coefficients for Gd-152				
R016	Contaminated zone (cm**3/g)	-1.000E+00	-1.000E+00	8.249E+02	DCNUCC(3)
R016	Unsaturated zone 1 (cm**3/g)	not used	-1.000E+00	---	DCNUCU(3,1)
R016	Saturated zone (cm**3/g)	not used	-1.000E+00	---	DCNUCS(3)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	2.693E-03	ALEACH(3)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(3)
R017	Inhalation rate (m**3/yr)	8.400E+03	8.400E+03	---	INHALR
R017	Mass loading for inhalation (g/m**3)	1.000E-04	1.000E-04	---	MLINH
R017	Exposure duration	3.000E+01	3.000E+01	---	ED
R017	Shielding factor, inhalation	4.000E-01	4.000E-01	---	SHF3
R017	Shielding factor, external gamma	7.000E-01	7.000E-01	---	SHF1
R017	Fraction of time spent indoors	1.800E-01	5.000E-01	---	FIND
R017	Fraction of time spent outdoors (on site)	4.600E-02	2.500E-01	---	FOTD
R017	Shape factor flag, external gamma	1.000E+00	1.000E+00	>0 shows circular AREA.	FS
R017	Radii of shape factor array (used if FS = -1):				
R017	Outer annular radius (m), ring 1:	not used	5.000E+01	---	RAD_SHAPE(1)
R017	Outer annular radius (m), ring 2:	not used	7.071E+01	---	RAD_SHAPE(2)
R017	Outer annular radius (m), ring 3:	not used	0.000E+00	---	RAD_SHAPE(3)
R017	Outer annular radius (m), ring 4:	not used	0.000E+00	---	RAD_SHAPE(4)
R017	Outer annular radius (m), ring 5:	not used	0.000E+00	---	RAD_SHAPE(5)
R017	Outer annular radius (m), ring 6:	not used	0.000E+00	---	RAD_SHAPE(6)
R017	Outer annular radius (m), ring 7:	not used	0.000E+00	---	RAD_SHAPE(7)
R017	Outer annular radius (m), ring 8:	not used	0.000E+00	---	RAD_SHAPE(8)
R017	Outer annular radius (m), ring 9:	not used	0.000E+00	---	RAD_SHAPE(9)
R017	Outer annular radius (m), ring 10:	not used	0.000E+00	---	RAD_SHAPE(10)
R017	Outer annular radius (m), ring 11:	not used	0.000E+00	---	RAD_SHAPE(11)
R017	Outer annular radius (m), ring 12:	not used	0.000E+00	---	RAD_SHAPE(12)

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R017	Fractions of annular areas within AREA:				
R017	Ring 1	not used	1.000E+00	---	FRACA(1)
R017	Ring 2	not used	2.732E-01	---	FRACA(2)
R017	Ring 3	not used	0.000E+00	---	FRACA(3)
R017	Ring 4	not used	0.000E+00	---	FRACA(4)
R017	Ring 5	not used	0.000E+00	---	FRACA(5)
R017	Ring 6	not used	0.000E+00	---	FRACA(6)
R017	Ring 7	not used	0.000E+00	---	FRACA(7)
R017	Ring 8	not used	0.000E+00	---	FRACA(8)
R017	Ring 9	not used	0.000E+00	---	FRACA(9)
R017	Ring 10	not used	0.000E+00	---	FRACA(10)
R017	Ring 11	not used	0.000E+00	---	FRACA(11)
R017	Ring 12	not used	0.000E+00	---	FRACA(12)
R018	Fruits, vegetables and grain consumption (kg/yr)	not used	1.600E+02	---	DIET(1)
R018	Leafy vegetable consumption (kg/yr)	not used	1.400E+01	---	DIET(2)
R018	Milk consumption (L/yr)	not used	9.200E+01	---	DIET(3)
R018	Meat and poultry consumption (kg/yr)	not used	6.300E+01	---	DIET(4)
R018	Fish consumption (kg/yr)	not used	5.400E+00	---	DIET(5)
R018	Other seafood consumption (kg/yr)	not used	9.000E-01	---	DIET(6)
R018	Soil ingestion rate (g/yr)	3.650E+01	3.650E+01	---	SOIL
R018	Drinking water intake (L/yr)	not used	5.100E+02	---	DWI
R018	Contamination fraction of drinking water	not used	1.000E+00	---	FDW
R018	Contamination fraction of household water	not used	1.000E+00	---	FHHW
R018	Contamination fraction of livestock water	not used	1.000E+00	---	FLW
R018	Contamination fraction of irrigation water	not used	1.000E+00	---	FIRW
R018	Contamination fraction of aquatic food	not used	5.000E-01	---	FR9
R018	Contamination fraction of plant food	not used	-1	---	FPLANT
R018	Contamination fraction of meat	not used	-1	---	FMEAT
R018	Contamination fraction of milk	not used	-1	---	FMILK
R019	Livestock fodder intake for meat (kg/day)	not used	6.800E+01	---	LFI5
R019	Livestock fodder intake for milk (kg/day)	not used	5.500E+01	---	LFI6
R019	Livestock water intake for meat (L/day)	not used	5.000E+01	---	LWI5
R019	Livestock water intake for milk (L/day)	not used	1.600E+02	---	LWI6
R019	Livestock soil intake (kg/day)	not used	5.000E-01	---	LSI
R019	Mass loading for foliar deposition (g/m**3)	not used	1.000E-04	---	MLFD
R019	Depth of soil mixing layer (m)	1.500E-01	1.500E-01	---	DM
R019	Depth of roots (m)	not used	9.000E-01	---	DROOT
R019	Drinking water fraction from ground water	not used	1.000E+00	---	FGWDW
R019	Household water fraction from ground water	not used	1.000E+00	---	FGWHH
R019	Livestock water fraction from ground water	not used	1.000E+00	---	FGWLW
R019	Irrigation fraction from ground water	not used	1.000E+00	---	FGWIR
R19B	Wet weight crop yield for Non-Leafy (kg/m**2)	not used	7.000E-01	---	YV(1)
R19B	Wet weight crop yield for Leafy (kg/m**2)	not used	1.500E+00	---	YV(2)
R19B	Wet weight crop yield for Fodder (kg/m**2)	not used	1.100E+00	---	YV(3)
R19B	Growing Season for Non-Leafy (years)	not used	1.700E-01	---	TE(1)
R19B	Growing Season for Leafy (years)	not used	2.500E-01	---	TE(2)
R19B	Growing Season for Fodder (years)	not used	8.000E-02	---	TE(3)
R19B	Translocation Factor for Non-Leafy	not used	1.000E-01	---	TIV(1)

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R19B	Translocation Factor for Leafy	not used	1.000E+00	---	TIV(2)
R19B	Translocation Factor for Fodder	not used	1.000E+00	---	TIV(3)
R19B	Dry Foliar Interception Fraction for Non-Leafy	not used	2.500E-01	---	RDRY(1)
R19B	Dry Foliar Interception Fraction for Leafy	not used	2.500E-01	---	RDRY(2)
R19B	Dry Foliar Interception Fraction for Fodder	not used	2.500E-01	---	RDRY(3)
R19B	Wet Foliar Interception Fraction for Non-Leafy	not used	2.500E-01	---	RWET(1)
R19B	Wet Foliar Interception Fraction for Leafy	not used	2.500E-01	---	RWET(2)
R19B	Wet Foliar Interception Fraction for Fodder	not used	2.500E-01	---	RWET(3)
R19B	Weathering Removal Constant for Vegetation	not used	2.000E+01	---	WLAM
C14	C-12 concentration in water (g/cm**3)	not used	2.000E-05	---	C12WTR
C14	C-12 concentration in contaminated soil (g/g)	not used	3.000E-02	---	C12CZ
C14	Fraction of vegetation carbon from soil	not used	2.000E-02	---	CSOIL
C14	Fraction of vegetation carbon from air	not used	9.800E-01	---	CAIR
C14	C-14 evasion layer thickness in soil (m)	not used	3.000E-01	---	DMC
C14	C-14 evasion flux rate from soil (1/sec)	not used	7.000E-07	---	EVSN
C14	C-12 evasion flux rate from soil (1/sec)	not used	1.000E-10	---	REVSN
C14	Fraction of grain in beef cattle feed	not used	8.000E-01	---	AVFG4
C14	Fraction of grain in milk cow feed	not used	2.000E-01	---	AVFG5
C14	DCF correction factor for gaseous forms of C14	not used	0.000E+00	---	CO2F
STOR	Storage times of contaminated foodstuffs (days):				
STOR	Fruits, non-leafy vegetables, and grain	1.400E+01	1.400E+01	---	STOR_T(1)
STOR	Leafy vegetables	1.000E+00	1.000E+00	---	STOR_T(2)
STOR	Milk	1.000E+00	1.000E+00	---	STOR_T(3)
STOR	Meat and poultry	2.000E+01	2.000E+01	---	STOR_T(4)
STOR	Fish	7.000E+00	7.000E+00	---	STOR_T(5)
STOR	Crustacea and mollusks	7.000E+00	7.000E+00	---	STOR_T(6)
STOR	Well water	1.000E+00	1.000E+00	---	STOR_T(7)
STOR	Surface water	1.000E+00	1.000E+00	---	STOR_T(8)
STOR	Livestock fodder	4.500E+01	4.500E+01	---	STOR_T(9)
R021	Thickness of building foundation (m)	not used	1.500E-01	---	FLOOR1
R021	Bulk density of building foundation (g/cm**3)	not used	2.400E+00	---	DENSFL
R021	Total porosity of the cover material	not used	4.000E-01	---	TPCV
R021	Total porosity of the building foundation	not used	1.000E-01	---	TPFL
R021	Volumetric water content of the cover material	not used	5.000E-02	---	PH2CCV
R021	Volumetric water content of the foundation	not used	3.000E-02	---	PH2OFL
R021	Diffusion coefficient for radon gas (m/sec):				
R021	in cover material	not used	2.000E-06	---	DIFCV
R021	in foundation material	not used	3.000E-07	---	DIFFL
R021	in contaminated zone soil	not used	2.000E-06	---	DIFCZ
R021	Radon vertical dimension of mixing (m)	not used	2.000E+00	---	HMIX
R021	Average building air exchange rate (1/hr)	not used	5.000E-01	---	REXG
R021	Height of the building (room) (m)	not used	2.500E+00	---	HRM
R021	Building interior area factor	not used	0.000E+00	---	FAI
R021	Building depth below ground surface (m)	not used	-1.000E+00	---	DMFL
R021	Emanating power of Rn-222 gas	not used	2.500E-01	---	EMANA(1)
R021	Emanating power of Rn-220 gas	not used	1.500E-01	---	EMANA(2)
TITL	Number of graphical time points	32	---	---	NPTS
TITL	Maximum number of integration points for dose	17	---	---	LYMAX

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
TITL	Maximum number of integration points for risk	257	---	---	KYMAX

Summary of Pathway Selections

Pathway	User Selection
1 -- external gamma	active
2 -- inhalation (w/o radon)	active
3 -- plant ingestion	suppressed
4 -- meat ingestion	suppressed
5 -- milk ingestion	suppressed
6 -- aquatic foods	suppressed
7 -- drinking water	suppressed
8 -- soil ingestion	active
9 -- radon	suppressed
Find peak pathway doses	suppressed

Contaminated Zone Dimensions		Initial Soil Concentrations, pCi/g	
Area:	334.50 square meters	Eu-152	2.790E+01
Thickness:	0.15 meters	Gd-152	2.790E+01
Cover Depth:	0.00 meters		

Total Dose TDOSE(t), mrem/yr
Basic Radiation Dose Limit = 2.500E+01 mrem/yr
Total Mixture Sum M(t) = Fraction of Basic Dose Limit Received at Time (t)

t (years):	0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
TDOSE(t):	2.505E+01	2.372E+01	2.127E+01	1.453E+01	4.922E+00	1.755E-01	4.106E-02	6.231E-03
M(t):	1.002E+00	9.488E-01	8.508E-01	5.813E-01	1.969E-01	7.021E-03	1.642E-03	2.493E-04

Maximum TDOSE(t): 2.505E+01 mrem/yr at t = 0.000E+00 years

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 0.000E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	2.496E+01	0.9963	7.066E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.855E-04	0.0000
Gd-152	0.000E+00	0.0000	7.973E-02	0.0032	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.238E-02	0.000[illegible]
Total	2.496E+01	0.9963	7.980E-02	0.0032	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.286E-02	0.000[illegible]

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 0.000E+00 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.496E+01	0.996[illegible]
Gd-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.211E-02	0.003[illegible]
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.505E+01	1.000[illegible]

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	2.363E+01	0.9961	6.690E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.596E-04	0.000
Gd-152	0.000E+00	0.0000	7.952E-02	0.0034	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.234E-02	0.000
Total	2.363E+01	0.9961	7.958E-02	0.0034	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.280E-02	0.000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+00 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract
Eu-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.363E+01	0.996
Gd-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.186E-02	0.003
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.372E+01	1.000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground mrem/yr	Ground fract.	Inhalation mrem/yr	Inhalation fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	Soil mrem/yr	Soil fract.
Eu-152	2.118E+01	0.9957	5.997E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.120E-04	0.000(
Gd-152	0.000E+00	0.0000	7.909E-02	0.0037	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.228E-02	0.000(
Total	2.118E+01	0.9957	7.915E-02	0.0037	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.269E-02	0.000(

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+00 years

Water Dependent Pathways

Radio-Nuclide	Water mrem/yr	Water fract.	Fish mrem/yr	Fish fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	All Pathways* mrem/yr	All Pathways* fract.
Eu-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.118E+01	0.995
Gd-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.137E-02	0.004
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.127E+01	1.000(

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+01 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	1.444E+01	0.9938	4.089E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.809E-04	0.000
Gd-152	0.000E+00	0.0000	7.761E-02	0.0053	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.205E-02	0.000
Total	1.444E+01	0.9938	7.765E-02	0.0053	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.233E-02	0.000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+01 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.444E+01	0.993
Gd-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.966E-02	0.006
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.453E+01	1.000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+01 years

Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground mrem/yr	Ground fract.	Inhalation mrem/yr	Inhalation fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	Soil mrem/yr	Soil fract.
Eu-152	4.837E+00	0.9827	1.370E-05	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.410E-05	0.000
Gd-152	0.000E+00	0.0000	7.354E-02	0.0149	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.142E-02	0.002
Total	4.837E+00	0.9827	7.356E-02	0.0149	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.151E-02	0.002

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+01 years

Water Dependent Pathways

Radio-Nuclide	Water mrem/yr	Water fract.	Fish mrem/yr	Fish fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	All Pathways* mrem/yr	All Pathways* fract.
Eu-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.837E+00	0.982
Gd-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.496E-02	0.017
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.922E+00	1.000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+02 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	1.052E-01	0.5991	2.978E-07	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.046E-06	0.000(
Gd-152	0.000E+00	0.0000	6.091E-02	0.3470	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.455E-03	0.053!
Total	1.052E-01	0.5991	6.091E-02	0.3470	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	9.457E-03	0.053!

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+02 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.052E-01	0.599:
Gd-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	7.036E-02	0.400(
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.755E-01	1.000(

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+02 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	1.868E-06	0.0000	5.291E-12	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.634E-11	0.000
Gd-152	0.000E+00	0.0000	3.554E-02	0.8656	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.517E-03	0.134
Total	1.868E-06	0.0000	3.554E-02	0.8656	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.517E-03	0.134

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 3.000E+02 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.868E-06	0.000
Gd-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.106E-02	1.000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	4.106E-02	1.000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+03 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Eu-152	4.411E-23	0.0000	1.859E-16	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.886E-17	0.000(
Gd-152	0.000E+00	0.0000	5.394E-03	0.8656	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.374E-04	0.134
Total	4.411E-23	0.0000	5.394E-03	0.8656	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.374E-04	0.134

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+03 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract
Eu-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.147E-16	0.000(
Gd-152	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.231E-03	1.000(
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.231E-03	1.000(

*Sum of all water independent and dependent pathways.

Dose/Source Ratios Summed Over All Pathways
Parent and Progeny Principal Radionuclide Contributions Indicated

Parent (i)	Product (j)	Thread Fraction	DSR(j,t) At Time in Years (mrem/yr)/(pCi/g) 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
Eu-152	Eu-152	7.208E-01	6.447E-01	6.104E-01	5.472E-01	3.731E-01	1.250E-01	2.717E-03	4.826E-08	1.140E-24
Eu-152	Eu-152	2.792E-01	2.497E-01	2.364E-01	2.119E-01	1.445E-01	4.841E-02	1.052E-03	1.869E-08	4.414E-25
Eu-152	Gd-152	2.792E-01	2.906E-18	8.501E-18	1.877E-17	4.659E-17	8.345E-17	8.644E-17	5.071E-17	7.697E-18
Eu-152	∑DSR(j)		2.497E-01	2.364E-01	2.119E-01	1.445E-01	4.841E-02	1.052E-03	1.869E-08	7.697E-18
Gd-152	Gd-152	1.000E+00	3.301E-03	3.293E-03	3.275E-03	3.214E-03	3.045E-03	2.522E-03	1.472E-03	2.234E-04

The DSR includes contributions from associated (half-life ≤ 180 days) daughters.

Single Radionuclide Soil Guidelines G(i,t) in pCi/g
Basic Radiation Dose Limit = 2.500E+01 mrem/yr

Nuclide (i)	t= 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
Eu-152	2.795E+01	2.952E+01	3.293E+01	4.830E+01	1.442E+02	6.632E+03	3.734E+08	*1.765E+14
Gd-152	*2.179E+01	*2.179E+01	*2.179E+01	*2.179E+01	*2.179E+01	*2.179E+01	*2.179E+01	*2.179E+01

*At specific activity limit

Summed Dose/Source Ratios DSR(i,t) in (mrem/yr)/(pCi/g)
and Single Radionuclide Soil Guidelines G(i,t) in pCi/g
at tmin = time of minimum single radionuclide soil guideline
and at tmax = time of maximum total dose = 0.000E+00 years

Nuclide (i)	Initial (pCi/g)	tmin (years)	DSR(i,tmin)	G(i,tmin) (pCi/g)	DSR(i,tmax)	G(i,tmax) (pCi/g)
Eu-152	2.790E+01	0.000E+00	8.945E-01	2.795E+01	8.945E-01	2.795E+01
Gd-152	2.790E+01	0.000E+00	3.301E-03	*2.179E+01	3.301E-03	*2.179E+01

*At specific activity limit

Individual Nuclide Dose Summed Over All Pathways
Parent Nuclide and Branch Fraction Indicated

Nuclide (j)	Parent (i)	THF(i)	DOSE(j,t), mrem/yr t= 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
Eu-152	Eu-152	7.208E-01	1.799E+01	1.703E+01	1.527E+01	1.041E+01	3.487E+00	7.581E-02	1.346E-06	3.180E-23
Eu-152	Eu-152	2.792E-01	6.968E+00	6.597E+00	5.913E+00	4.032E+00	1.351E+00	2.936E-02	5.216E-07	1.232E-23
Eu-152	∑DOSE(j)		2.496E+01	2.363E+01	2.118E+01	1.444E+01	4.837E+00	1.052E-01	1.868E-06	4.411E-23
Gd-152	Eu-152	2.792E-01	8.108E-17	2.372E-16	5.236E-16	1.300E-15	2.328E-15	2.412E-15	1.415E-15	2.147E-16
Gd-152	Gd-152	1.000E+00	9.211E-02	9.186E-02	9.137E-02	8.966E-02	8.496E-02	7.036E-02	4.106E-02	6.231E-03
Gd-152	∑DOSE(j)		9.211E-02	9.186E-02	9.137E-02	8.966E-02	8.496E-02	7.036E-02	4.106E-02	6.231E-03

THF(i) is the thread fraction of the parent nuclide.

Individual Nuclide Soil Concentration
Parent Nuclide and Branch Fraction Indicated

Nuclide (j)	Parent (i)	THF(i)	S(j,t), pCi/g t= 0.000E+00	1.000E+00	3.000E+00	1.000E+01	3.000E+01	1.000E+02	3.000E+02	1.000E+03
Eu-152	Eu-152	7.208E-01	2.011E+01	1.904E+01	1.707E+01	1.164E+01	3.898E+00	8.475E-02	1.505E-06	3.555E-23
Eu-152	Eu-152	2.792E-01	7.790E+00	7.375E+00	6.611E+00	4.508E+00	1.510E+00	3.283E-02	5.831E-07	1.377E-23
Eu-152	∑S(j):		2.790E+01	2.642E+01	2.368E+01	1.615E+01	5.408E+00	1.176E-01	2.088E-06	4.932E-23
Gd-152	Eu-152	2.792E-01	0.000E+00	4.859E-14	1.378E-13	3.795E-13	7.005E-13	7.304E-13	4.286E-13	6.505E-14
Gd-152	Gd-152	1.000E+00	2.790E+01	2.782E+01	2.768E+01	2.716E+01	2.573E+01	2.131E+01	1.244E+01	1.888E+00
Gd-152	∑S(j):		2.790E+01	2.782E+01	2.768E+01	2.716E+01	2.573E+01	2.131E+01	1.244E+01	1.888E+00

THF(i) is the thread fraction of the parent nuclide.

RESCALC.EXE execution time = 4.36 seconds