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Reactor Administrator: Richard L. Holm

August 20, 2007
Docket No. 50-151
License No. R-115

Tom McLaughlin
U.S. Nuclear Regulatory Commission
MS T-8F5
Washington, DC 20555

Dear Sir,

SUBJECT: Resubmittal of Decommissioning Plan and Final Status Survey Plan

I am enclosing a copy of the revised decommissioning plan for the University of Illinois Advanced TRIGA reactor facility. The revised plan is submitted in its entirety. I am also enclosing the Final Status Survey plan per our discussion.

If there are any questions please do not hesitate to contact me.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 20, 2007.

Sincerely,

Richard L. Holm
Reactor Administrator

c: File

DECOMMISSIONING PLAN

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

Prepared by:
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March 2006

Revised 8/15/07 – R. Holm

Project Application	***Prepared By***	***Date***
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APPROVALS:

Title	***Signature***	***Date***
Project Manager	Kevin E. Taylor, PE, CHP	3/2/06
Operations Manager	Lee G. Penney	3/2/06
Field Services RSO	Kenneth M. Kasper, CIH CHP	3/2/06

REVISION LOG

Revision Number	Affected Pages	CRA Number	Approval
1	14,16,17,22	-----	Holm

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

TABLE OF CONTENTS

FIGURES

TABLES

APPENDICES

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.

A comprehensive background radioactivity determination (soil, concrete and metals) will be made prior to the analysis for release of materials from the site.

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, although not found 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.

The NRC default screening DCGLs and EPA MOU consultation triggers will be utilized for soil release. Table 2-3 provides the DCGLs for the 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	Screening Value DCGL (pCi/g)
Cobalt-60	3.8
Europium-152	6.9
Tritium (hydrogen-3)	110
Carbon-14	12
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). Records indicate leakage from the tank as far back as the mid-1960's. The tunnel containing the primary coolant piping provided a catch basin for the leakage from under the primary tank. The tunnel has a small sump in the floor directly under the entrance. This sump would collect the leakage and trigger an alarm when full. During the early 1990's it was discovered that there was a crack in the concrete in the sump that was allowing water to go in/out of the sump. The crack was sealed and the issue was resolved. The primary source of leak water is believed to be from the Bulk Shielding Tank, not the primary tank, following a path along the thermal column and beamports. The leak water then either went onto the bay floor or into the tunnel. When the Bulk Shielding Tank was emptied after removal of the fuel all leak paths dried up. 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

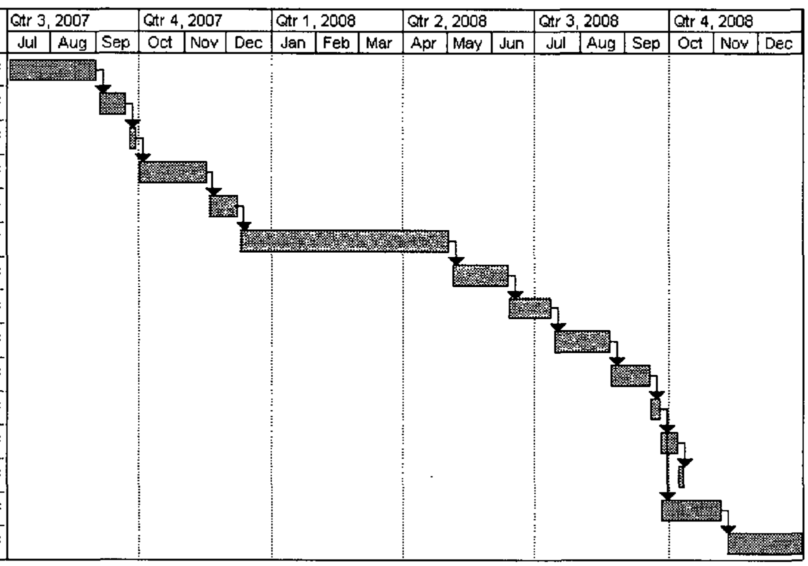
The proposed decommissioning project schedule is provided in Figure 2-1. The schedule, spanning five calendar quarters tentatively beginning with the third quarter of 2007, includes all activities from pre-mobilization project planning to the NRC's review of the Final Status Survey Report and license 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.

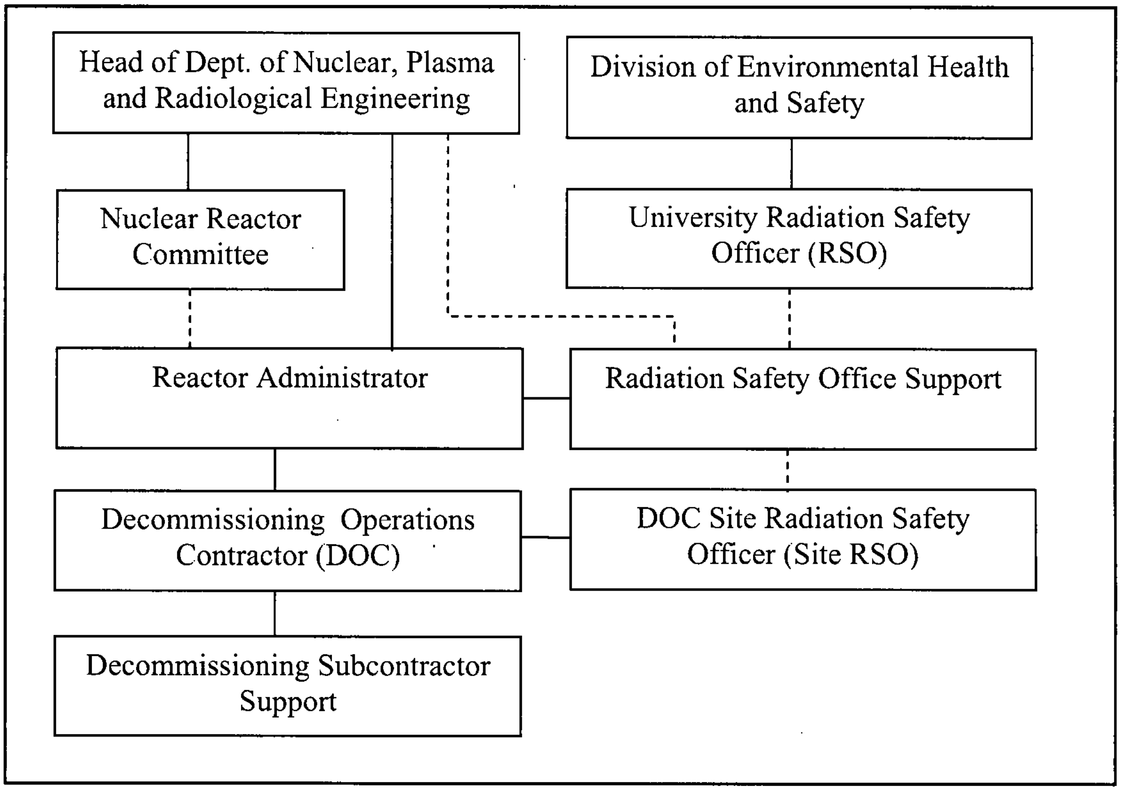
The University Radiation Safety Officer (RSO) shall be responsible for monitoring, planning, and promoting radiological safety at the facility. The University RSO has the responsibility and authority to stop, secure or otherwise control as necessary any operation or activity that poses an unacceptable 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 mrem off-site impact and < 25 mrem 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 mrem 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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ENERGYSOLUTIONS

CS-RS-PN-003

Final Status Survey Plan (FSSP) for the Nuclear Research Laboratory University of Illinois

Revision 1

U. S. Nuclear Regulatory Commission
Facility Operating License No. R-115

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August 2007

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- [ ] New
- [ ] Title Change
- [X] Revision
- [ ] Rewrite
- [ ] Cancellation

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1	8-12, 18	NA	K. Kasper, L. Penney

Table of Contents

Tables and Figures

ABBREVIATIONS/ACRONYMS

2x2 NaI	2-inch by 2-inch sodium iodide detector
cpm	counts per minute
DCGL	Derived Concentration Guideline Level
DP	Decommissioning Plan
DQO	Data quality objective
dpm	disintegrations per minute
Energy*Solutions*	Energy*Solutions*, LLC
FSS	Final status survey
FSSP	Final Status Survey Plan
GPS	Global Positioning Satellite
LBGR	Lower bound of the gray region
LLRW	Low-level radioactive waste
MARSSIM	Multi Agency Radiation Site Survey and Investigation Manual, NUREG-1575
MDC	Minimum Detectable Concentration
MDCR	Minimum Detectable Count Rate
mrem	Millirem
NRC	U.S. Nuclear Regulatory Commission
NRL	Nuclear Research Laboratory
pCi/g	Picocuries per gram
QAPP	Quality Assurance Program Plan
SADA	Spatial Analysis and Decision Assistance
ScanMDC	Scanning minimum detectable concentrations
TRIGA	Teaching Research Isotope General Atomic
University	University of Illinois at Urbana-Champaign
WRS	Wilcoxon Rank Sum

1. PURPOSE AND SCOPE

1.1 Purpose

This Final Status Survey Plan (FSSP) was prepared to support the termination of U.S. Nuclear Regulatory Commission (NRC) Facility Operating License R-115 which covered the operation of the University of Illinois' (University) Nuclear Research Laboratory (NRL) and Advanced Teaching Research Isotope General Atomic (TRIGA) Mark II nuclear research reactor. This FSSP describes the activities that the University's decommissioning contractor should perform to demonstrate that residual radioactivity in the footprint of the NRL following demolition meets the derived concentration guideline levels (DCGL). The DCGLs will be approved by the NRC along with their acceptance of the facility Decommissioning Plan (DP) (Reference 8.1). The overall decommissioning approach is described in the DP.

1.2 Scope

The NRL final status survey (FSS) will incorporate on-site radiological survey techniques as well as off-site laboratory analysis of soil samples. On-site techniques will include walk-over gamma radiation surveys and direct radiation measurements. An off-site laboratory will be used to analyze all soil samples from designated location.

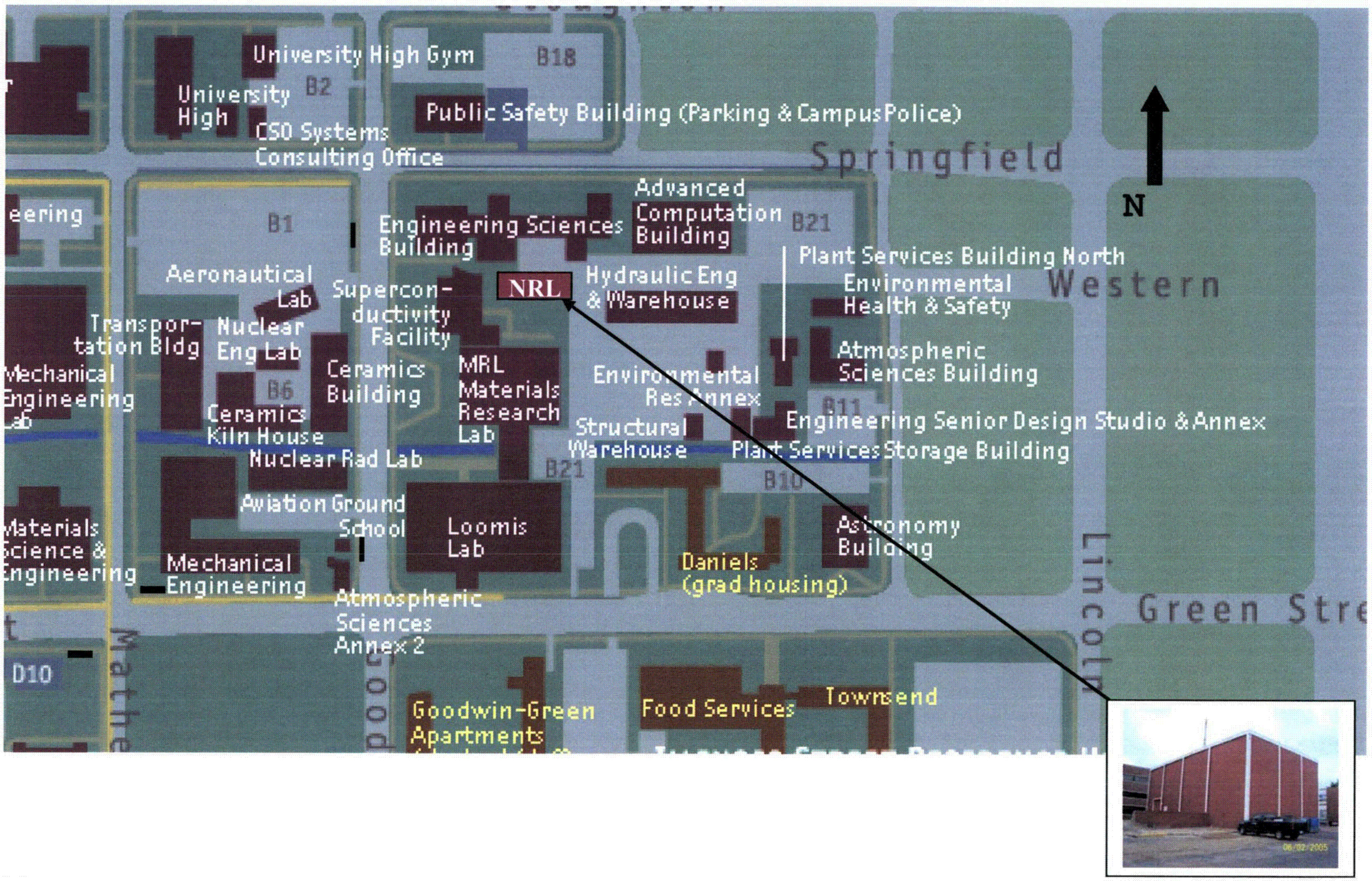
The guidance provided in NRC Guidance (NUREG)-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" (Reference 8.2) was used in designing survey and sampling efforts described in this FSSP to demonstrate compliance with the DCGLs.

Decommissioning operations may require modifications to this FSSP which may include, but are not limited to: adjustments of the boundaries of a survey unit, changes in the locations of survey points, the addition of survey units, or substitution of survey instruments. Modifications to this plan altering the intent or purpose of the FSS or affecting the overall quality of survey data shall be approved by the University. A license amendment is not required for issuing a revision to this FSSP.

2. SITE DESCRIPTION

The NRL building is a steel frame concrete block building that is approximately 80 feet east-west by 45 feet north-south located on the main University campus. The building is located between Green St. and Springfield St. as shown in Figure 2-1. 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 foundation, which is laid on concrete pile caps, supports the walls.

The DP calls for the NRL to be completely demolished and all systems removed. 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. The resulting survey units will consist of the building foot print and immediate surrounding areas.



ENERGYSOLUTIONS	**FIGURE 2-1** **LOCATION NRL FACILITY**

3. APPLYING THE RELEASE CRITERIA

Upon approval of the facility DP, the NRC will approve the site DCGLs for soil as provided in Table 3-1. The DCGLs are the screening levels for radionuclide concentrations in soils in picocuries per gram (pCi/g) as provided in the NRC's supplemental guidance (Reference 8.3) to the License Termination Rule (Reference 8.4). When applying the DCGLs in Table 3-1, 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 3-1
DCGLs for Primary Radionuclides of Concern in Soil

Radionuclide	Screening Value DCGL (pCi/g)
Gamma-Emitting Radionuclides	
Cobalt-60	3.8
Europium-152	6.9
Cesium-137	11
Europium-154	8.0
Hard-To-Detect Radionuclides	
Tritium (hydrogen-3)	110
Carbon-14	12
Iron-55	10,000
Nickel-63	2,100

The release criteria for surface contamination on remaining building materials, such as concrete or steel footings and foundation, are provided in Table 3-2. These release criteria are from the NRC's Regulatory Guide 1.86.

Table 3-2
Release Criteria for Surface Contamination

	Average[a]	Maximum[b]	Removable
Net beta-gamma activity in dpm/100 cm^2	5,000	15,000	1,000

Notes: [a]Averaged over not more than 1 square meter
[b]Applicable to an area of not more than 100 square centimeters

The average and maximum activity values apply to the total beta-gamma activity from the gamma-emitting radionuclides in Table 3-1 and carbon-14. The detector efficiency for carbon-14 should be used to conservatively estimate the total activity. The removable contamination limit applies to the hard-to-detect isotopes as well as those detectable with direct measurements.

3.1 Surveys and Sampling

The NRL footprint will be a single Class 1 survey unit. This will include the excavation floor and the side walls. The area surrounding the excavation will be a single Class 2 survey unit. The classifications are based on the potential for radioactive materials or contamination to be present in the survey unit following remediation according to MARSSIM protocols (Reference 8.2).

Survey units designated as Class 1 survey units, those most likely to contain residual contamination above the release criteria, will be no greater than 2,000 m^2 in area. Each of these survey units will receive, at a minimum, a 100% walkover survey with gamma scintillation detectors [e.g., a 2-inch by 2-inch sodium iodide (2x2 NaI) detector] and discrete sampling. Each survey unit will have 28 sample points as described in Section 3.2. The number of samples was determined using the MARSSIM protocols as presented in the following section. The sample locations will be determined using a random-start square grid pattern.

Survey units designated as Class 2 survey units, those less likely to contain residual contamination above the release criteria, will be no greater than 10,000 m^2 in area. Each of these survey units will receive a 25% walkover survey and discrete sampling. The Class 2 survey unit will also have 28 sample points as developed in the previous paragraph. The number of samples was determined using the MARSSIM protocols. The sample locations will be determined using a random-start square grid pattern.

Area exposure rates will be taken at one meter above ground surfaces at the location of each sample point using an exposure rate meter measuring in units of microRoentgen per hour (uR/hr). The purpose of the measurements will be to provide a comparison of the post-decontamination exposure levels to the general site background.

3.2 Number of Samples

MARSSIM (Reference 8.2) provides the following equations for determining the number of samples:

$$N = \frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{4(SignP - 0.5)^2}$$

Where:

N = Number of combined samples in the survey unit and the reference area (rounded up to the nearest integer value)

Z = Z-statistic based on selected α and β error rates forum in Table 5.2 in MARSSIM.

α = Acceptable Type I (false positive) error rate.

β = Acceptable Type II (false negative) error rate.

*Sign*P = Probability of a random measurement from the survey unit exceeding a random measurement from the reference area by less than the DCGL; *Sign*P, found in Table 5.1 of MARSSIM, is based on the relative shift Δ/σ.

- Δ is the shift which is typically defined as the DCGL minus the lower bound of the gray region (LBGR).
- σ is the estimated standard deviation of data from the survey unit.

To determine the standard deviation (σ) for cobalt-60, the gamma spectroscopy data for the soil samples presented in the Site Characterization Report (Reference 8.5) with concentrations below the screening criteria of 3.8 pCi/g were used (Samples: NMNT-10, NMNT-15, NMNT-45. NMNT-46, NMNT-88, Bkgnd Soil, Unk 1, and Unk 2). Soil samples with no cobalt-60 reported were assigned a concentration of 3.8 pCi/g. The resulting cobalt-60 σ for the 8 applicable soil samples is 1.39 pCi/g. No europium-152 peaks were identified in any of the soil samples. Therefore, the σ for europium-152 is estimated as the σ of the one-half the minimum detectable activities (MDA) for the same 8 samples used for the cobalt-60 σ calculation. The resulting europium-152 σ for the 8 applicable soil samples is 1.92 pCi/g. The tritium σ for soil was calculated using the tritium concentrations provided by off-site laboratory soil analysis (Reference 8.5). The resulting tritium σ is 0.37 pCi/g. Other non-gamma emitting isotopes of concern provided in Table 3-1 were not analyzed for in soil samples during the site characterization. The other gamma-emitting isotopes were not identified during gamma spectroscopy analysis. With multiple contaminants, the standard deviations are normalized using the unity rule as follows:

$$\sigma = \sqrt{\left(\frac{0.37}{110}\right)^2_{H3} + \left(\frac{1.39}{3.8}\right)^2_{Co60} + \left(\frac{1.92}{6.9}\right)^2_{Eu152}} = 0.460$$

Furthermore, the DCGL is normalized to 1.0.

Therefore, with a DCGL of 1.0, if the LBGR is estimated as one-half of the DCGL (or 0.5), this means that Δ is 0.5. With σ equal to 0.460, the relative shift Δ/σ then equals 1.09. Looking up the next lowest value of the relative shift in Table 5.1 of MARSSIM, Sign P is identified as 0.841. The Sign P statistic is used because the gamma-emitting isotopes of concern are generally not detectable in background samples. (Samples analyzed during the site characterization were analyzed inside the reactor building where there is a measurable cobalt-60 background.). The hard-to-detect isotopes in Table 3-1 are not a concern with respect to determining the number of samples because their screening criteria are substantially higher than measurable background levels (see Section 3.5).

Typically, 5% is an acceptable error rate for both Type I and Type II errors. Therefore, α and β are both equal to 0.05 and the Z-statistics from Table 5.2 of MARSSIM is 1.645.

Substituting *Sign*P and the Z-statistics into the previous equation for N gives:

$$\frac{(1.645 + 1.645)^2}{4(0.841 - 0.5)^2} = 23.27$$

Then, according to MARSSIM protocols, the value for N is increased by 20% and rounded to the nearest integer.

$$N = 23.27 \text{ x } 1.2 = 28$$

Therefore, the recommended number of samples in each survey unit is 28.

3.3 Instruments and Detection Limits

3.3.1 Walkover Surveys

The FSS will consist of walkover surveys with gamma scintillation detectors, and soil sampling with off-site analysis. The instruments proposed for use during the FSS and their applications are provided in Table 3-3. If necessary, the decommissioning contractor may substitute comparable instruments.

All instruments will be calibrated using NIST-traceable standards. Instruments will be checked at the beginning of each day to ensure they are operating properly. The daily check also reassures the validity of the previous day's measurements. Instrument control logs/charts will be maintained. The daily checks will include a background measurement and a source check.

Instrument records, including dates of use, efficiencies, calibration due dates and source traceability, will be maintained in accordance with established procedures.

In Table 6.4 of NUREG-1507 (Reference 8.6), the minimum detectable cobalt-60 concentration in soil for scanning measurements (ScanMDC) (walk-over survey with a 2x2 NaI detector) is approximately 3.4 pCi/g. For cesium-137, the ScanMDC is approximated at 6.4 pCi/g. Therefore, based on the gamma photon energies of these isotopes and the other gamma-emitting isotopes of concern, the walk-over surveys should be capable of identifying the presence of residual contamination at levels below the DCGLs.

For additional verification, the cobalt-60 DCGL (3.8 pCi/g) was modeled using MicroShield with an infinite slab source geometry, and is consistent with the analysis reported in Section 6.8.2 of Reference 8.6. This analysis, presented in Attachment 9.1, shows that the expected exposure rate for soil containing 3.8 pCi.g cobalt-60 would be 5.1 uR/hr. Using the conversion factor of 430 cpm per uR/hr from Reference 8.6, the required minimum detectable count rate (MDCR) to measure 3.8 pCi/g of cobalt-60 would be approximately 2,193 cpm.

Using the MDCR equation in Table 3-4 and estimating a background count rate of 8,000 cpm (133.3 cps), the expected MDCR is 1,352 net cpm (841 cpm less than the required MDCR) or approximately 9,350 gross cpm (depending on the true background rate on site). Using the ScanMDC (pCi/g) equation in Table 3-4, this MDCR correlates to an expected ScanMDC of about 3.1 pCi/g for cobalt-60 contamination in the soil.

MARSSIM protocols also recommend the derivation of a DCGL for elevated measurements comparison ($DCGL_{EMC}$). For this derivation, an outside Area Factor is needed. To estimate the Area Factor, the area of each square grid must be approximated based on the size of the survey unit and the number of sampling points. The Class 1 survey unit of the NRL footprint is estimated at 608 m^2 (100' by 65'). Based on 28 sample points, the distance between the sample points should be 4.75 meters with each square grid approximately 22.5 m^2.

Using Table 5-6 from MARSSIM (Reference 8.2), the Area Factor can be approximated for cobalt-60, the primary contaminant of concern, for the grid area of 22.5 m^2. Interpolating between 10 m^2 and 30 m^2, the Area Factor is 1.9. The $DCGL_{EMC}$ (or required ScanMDC) is equal to the DCGL times the Area Factor, or 7.22 pCi/g for cobalt-60. Since the expected ScanMDCs (from Table 6.4 of Reference 8.6 and the MicroShield analysis) are less than the $DCGL_{EMC}$ (required ScanMDC), the survey approach and the number of samples is statistically acceptable.

Table 3-3
FSS Instruments

Application	Primary Instrument
• Walkover surveys and fixed-point measurements	Ludlum 2221 or Ludlum 2350 scaler/rate meters with Ludlum 44-10 2x2 NaI detectors (optional GPS data/position logging system).
• Soil analysis (500 ml samples)	Off-site gamma spectroscopy and analysis for H-3, C-14, Fe-55, and Ni-63
• Exposure rates	Ludlum Model 19 microR meter.
• Surface scans and direct measurements of footing and piles	Ludlum 2360 scaler/rate meters with a 43-93 100 cm^2 α/β phoswich probe
• Analysis of removable contamination smears	Liquid scintillation counter (LSC)

Table 3-4
MDC Calculation Equations

Factor	Equation	Variables
MDCR (cpm)	$MDCR = \frac{d'\sqrt{b_i} \times (60/i)}{\sqrt{0.5}}$ $b_i = b(i/60)$	b = Background count rate (cpm) i = observation interval = 1 s d' = detectability value = 1.38 0.5 = surveyor efficiency
ScanMDC (pCi/g)	$ScanMDC = MDCR\left(\frac{\mu R/hr}{430cpm}\right)\left(\frac{3.8pCi/g}{3.9\mu R/hr}\right)$	Conversion factors from Table 6.4 of Reference 8.6 and MicroShield analysis (Attachment 9.1).
ScanMDC (dpm/100cm^2)	$ScanMDC = \frac{d' * 60/i \sqrt{R_B\, i/60}}{\sqrt{0.5} * \varepsilon_s * \varepsilon_i * \frac{probe.area}{100}}$	R_B = background rate (cpm) i = observation interval = 1 s d' = detectability value = 1.38 0.5 = surveyor efficiency ε_s = surface efficiency ε_i = detector intrinsic (2-pi) efficiency
Direct MDC (dpm/100cm^2)	$MDC = \frac{3 + 3.29\sqrt{R_B t_s\left(1 + t_s/t_B\right)}}{t_s * \varepsilon_s * \varepsilon_i * \frac{probe.area}{100}}$	R_B = background rate (cpm) t_s = sample count time (min) t_B = bkgnd. count time (min) ε_s = surface efficiency ε_i = detector intrinsic (2-pi) efficiency

3.3.2 Surveys of Remaining Building Materials

The FSS may also include surveys of building materials, such as footings and piles, that may remain in the NRL footprint after building demolition. These materials will be covered by backfill during site restoration. The instrument proposed for use to survey these materials is provided in Table 3-3. If necessary, the decommissioning contractor may substitute comparable instruments.

The surveys of the remaining structural materials will consist of surface scans and direct measurements. In accordance with the Draft NUREG-1575, Supplement 1, "Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME)" (Reference 8.7), the remaining building materials can be classified in a manner similar to the survey unit classifications in MARSSIM. Because of the low potential that these materials are contaminated above the release criteria in Table 3-2, these materials will be classified as Class 2 materials.

Surface scans will cover a minimum of 10% of the accessible surfaces. Direct total surface activity measurements will be made at random and biased locations. These surveys will use the instruments identified in Table 3-3 or equivalent. Direct and scan measurements will only identify the gamma-emitting isotopes of concern.

Removable contamination smears will also be collected at each direct measurement location. These smears will be analyzed for tritium and the other hard to detect radionuclides of concern using an LSC.

3.4 Daily Instrument and Background Measurements

Daily instrument checks will be made according to written procedures. These measurements will be made in non-impacted areas using radioactive check sources. These measurements will be recorded for the purpose of ensuring that instruments are operating properly. An instrument control log will be used for each instrument to keep track of background counts and response checks

Daily background measurements will also be made according to written procedures. These measurements will be made in non-impacted areas. Single background measurements used to estimate the mean background will be made for a minimum of 10 minutes for scaling instruments (scalers).

For release surveys conducted on building materials, a background measurement should be made on a similar type of material in a non-impacted area.

3.5 Reference Area Measurements

The radionuclides of concern at the NRL site fall into two distinct categories:

1) The DCGL is low and the radioisotope is generally not detectable in background samples (cobalt and europium), and
2) The DCGL is high compared to the expected background concentration (remaining isotopes)

Therefore, to simplify matters, the site release statistical tests will assume that none of the radioisotopes of concern are present in background. Based on this assumption, MARSSIM recommends the Sign Test for statistical comparisons. No reference area measurements are required for the Sign Test to release the site based on the soil sample results.

4. DATA QUALITY OBJECTIVES

The Data Quality Objective (DQO) process provides systematic procedures for defining the criteria that the FSS survey design should satisfy. The following DQOs are quantitative and qualitative statements derived from the output of the DQO process.

- The null hypothesis (H_0) is defined as: The residual activity in the survey unit exceeds the release criteria.
- The upper bound of the gray region is originally defined as the DCGL and the lower bound of the gray region is defined as one-half of the DCGL but can be adjusted for an acceptable relative shift.
- The Type I and Type II decision error probabilities for determining the number of samples per survey unit for comparison tests are both 5%.
- For off-site soil sample analysis, minimum detectable concentrations (MDC) should be less than 25% of the DCGLs.
- The ScanMDC using a NaI detector for walk-over surveys will not be greater than 100% of the $DCGL_{EMC}$ for gamma-emitting isotopes.
- Survey measurements will be documented and controlled as described in written procedures.
- Locations for soil sampling in Class 1 and Class 2 survey units will be established using a systematic, random start pattern.
- Data quality will be assessed through a combination of on-site analysis of duplicate samples, replicate on-site analyses, and replicate off-site analyses.

DQOs may be adjusted during the course of the project. Allowable modification to the DQOs, such as changing the bounds on the gray region, adjusting the Type II error rate, or slightly altering the limits of the MDCs with respect to the DCGLs, will not alter the intent of the FSS or affect the overall quality of survey data. More significant changes, such as allowing MDCs to exceed the DCGLs, will be reviewed with the NRC.

5. SURVEYS AND SAMPLING

5.1 Survey and Sampling in Class 1 Excavation Area

The excavated area should consist of a single Class 1 survey unit based on MARSSIM recommendations. MARSSIM recommends that exterior Class 1 survey units be limited to 2,000 square meters (m^2), or about 0.5 acres.

The decommissioning contractor will perform a 100% walk-over survey using NaI detectors and collect soil samples at predetermined locations using a random-start grid pattern as shown in Figure 5-1. Prior to collecting soil samples, direct measurements will be made to record gamma count rate and exposure rate. The sampling point coordinates are provided in Table 5-1. There are 28 sample points that fit in the survey unit based on location of the random start point.

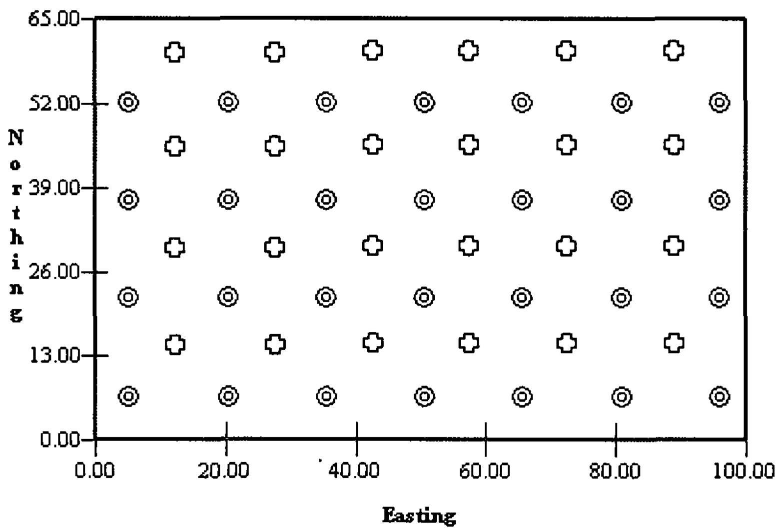
The sample locations are based on the assumption that the excavated footprint will be 100 feet long (east-west) and 65 feet wide (north-south). If the actual excavation differs from these dimensions, new sample locations will need to be determined.

Because may of the isotopes of concern are not detectable with the walk-over survey, additional soil samples will be collected in the Class 1 survey unit for analysis of hard-to-detect isotopes of concern. Since a square grid pattern was used for the initial MARSSIM sampling, the additional samples will be collected from the center of each square grid. In Figure 5-1, there are 24 additional samples for hard-to-detect isotopes.

Soil samples will be collected and prepared in accordance with a site-specific sampling plan. Sample containers and sample volumes will be consistent with off-site laboratory needs. Samples will be packaged according to the off-site laboratory's requirements. Written chain-of-custody procedures will be followed while handling and transferring samples.

Samples will be identified using a unique sample numbering scheme. Duplicate samples should include "(DUP)" in the identification number.

Figure 5-1
NRL Footprint Class 1 Survey Unit



◎ - MARSSIM sample point
✪ - Additional sample point for hard-to-detect isotopes

Table 5-1
MARSSIM Sample Point Coordinates

Sample Number	Feet East	Feet North	Sample Number	Feet East	Feet North
1	5.1	6.9	15	5.1	38.1
2	20.7	6.9	16	20.7	38.1
3	36.3	6.9	17	36.3	38.1
4	51.9	6.9	18	51.9	38.1
5	67.5	6.9	19	67.5	38.1
6	83.1	6.9	20	83.1	38.1
7	98.7	6.9	21	98.7	38.1
8	5.1	22.5	22	5.1	53.7
9	20.7	22.5	23	20.7	53.7
10	36.3	22.5	24	36.3	53.7
11	51.9	22.5	25	51.9	53.7
12	67.5	22.5	26	67.5	53.7
13	83.1	22.5	27	83.1	53.7
14	98.7	22.5	28	98.7	53.7

5.2 Survey and Sampling in Class 2 Area

The Class 2 survey unit will receive a 25% walkover scan using the NaI detectors, direct measurements will be made, and soil samples will be collected at predetermined locations using a random-start grid pattern. The sample grid is not shown is this FSSP because the boundaries will need to be determined following NRL demolition. Radiation measurements and sample collection, preparation, and analysis will be as described in Section 5.1.

For sample locations which are on paved surfaces, the soil samples should come from the initial soil layer below the pavement and base layer in accordance with a site-specific sampling plan.

No additional sampling and analysis for hard-to-detect radionuclides is necessary in Class 2 areas

6. DATA MANAGEMENT

The decommissioning contractor will have the responsibility of on-site data management. Data will be managed in accordance with a site-specific plan or other written procedures. Data management includes recording field data, transcribing data into electronic files, backing up electronic data, filing original and hard copies of data, and obtaining proper reviews and signatures on data records.

7. QUALITY ASSURANCE

The decommissioning contractor shall operate under a strict quality assurance (QA) program that includes the elements described in Chapter 9 of MARSSIM. QA protocols should cover items such as document control, control of measurement and test equipment, chain-of-custody, and data validation and verification.

The off-site laboratory used by the decommissioning contractor should be on either an approved vendors list maintained by the decommissioning contractor's QA department or on an approved vendors list maintained by the University.

8. REFERENCES

8.1 Energy*Solutions*, LLC. Decommissioning Plan, Nuclear Research Laboratory, University of Illinois at Champaign-Urbana, Document No. 82A9581.

8.2 U.S. Nuclear Regulatory Commission. "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)." NUREG-1575, Revision 1. August 2000.

8.3 U.S. Nuclear Regulatory Commission. "Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination." Federal Register. Volume 63, Number 222. November 18, 1998.

8.4 Code of Federal Regulations. Chapter 10, Part 20 (10 CFR 20), Subpart E, Radiological Criteria for License Termination.

8.5 Scientech, LLC. Site Characterization Report, Nuclear Research Laboratory, University of Illinois at Champaign-Urbana, Document No. 82A9571.

8.6 U.S. Nuclear Regulatory Commission. "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions." NUREG-1507. June 1998.

8.7 U.S. Nuclear Regulatory Commission. "Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME)." Draft NUREG-1575, Supplement 1. December 2006.

9. ATTACHMENTS

9.1 MicroShield Analysis for ScanMDC Determination

Attachment 9.1
MicroShield Analysis for ScanMDC Determination

Case Summary of Cobalt-60 In Soil

Page 1 of 1

MicroShield 7.00
EnergySolutions (06-MSD-7.00-1099)

Date	By	Checked

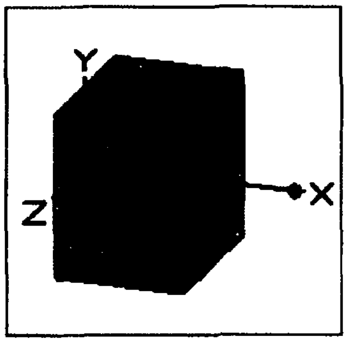
Filename	Run Date	Run Time	Duration
Cobalt in soil.ms6	July 30, 2007	11:20:54 AM	00:00:01

Project Info	
Case Title	Cobalt-60 In Soil
Description	In Support of University of Illinois FSSP
Geometry	16 - Infinite Slab

Source Dimensions	
Thickness	15.0 cm (5.9 in)

Dose Points			
A	X	Y	Z
#1	25.0 cm (9.8 in)	0.0 cm (0.0 in)	0.0 cm (0.0 in)

Shields			
Shield N	Dimension	Material	Density
Source	Infinite	Soil	1.6
Air Gap		Air	0.00122



Source Input: Grouping Method - Actual Photon Energies				
Nuclide	$\mu Ci/cm^3$	Bq/cm^3	$\mu Ci/cm^3$	Bq/cm^3
Co-60	6.0800e-006	2.2496e-001	6.0800e-006	2.2496e-001

Buildup: The material reference is Source
Integration Parameters

Results					
Energy (MeV)	Activity (Photons/sec)	Fluence Rate $MeV/cm^2/sec$ No Buildup	Fluence Rate $MeV/cm^2/sec$ With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	3.670e-05	1.028e-04	1.028e-04	1.985e-07	1.985e-07
1.1732	2.250e-01	1.319e+00	1.319e+00	2.358e-03	2.358e-03
1.3325	2.250e-01	1.579e+00	1.579e+00	2.740e-03	2.740e-03
Totals	**4.500e-01**	**2.899e+00**	**2.899e+00**	**5.098e-03**	**5.098e-03**