University of Arizona Nuclear Reactor Laboratory Decommissioning Plan

Nuclear Regulatory Commission
Facility Operating License R-52

Prepared for:
University of Arizona

THE UNIVERSITY OF ARIZONA

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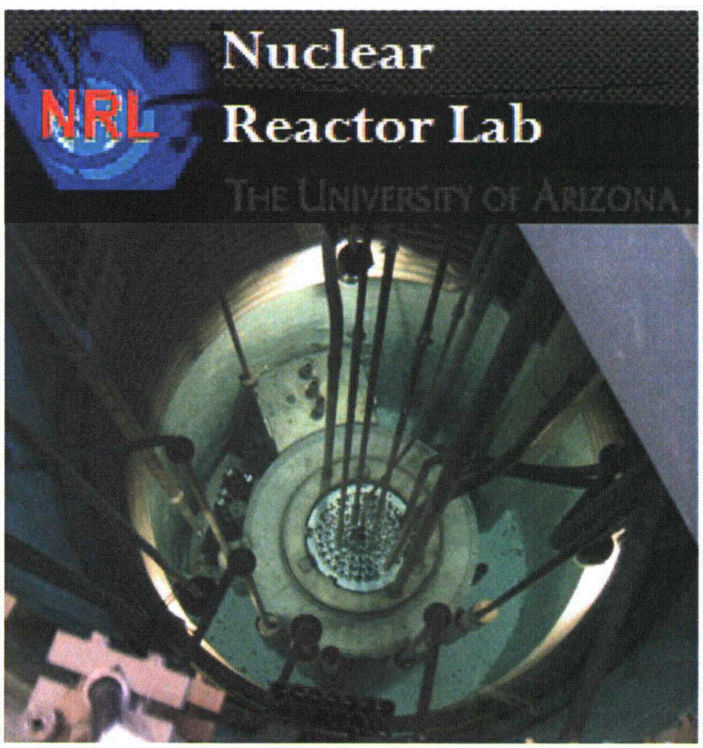

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Acronyms and Abbreviations

ACM	Asbestos Containing Material
ALARA	Asbestos Containing Material
ANL	Argonne National Lab
CAM	Contamination Air Monitor
CEDE	Committed Effective Dose Equivalent
cfm	Cubic feet per minute
CFR	Code of Federal Regulations
D & D	Decontamination and Decommissioning
DC	Decommissioning Contractor
DCGL	Derived Concentration Guideline Levels
DDE	Deep Dose Equivalent
DOE	Department of Energy
DOT	Department of Transportation
DP	Decommissioning Plan
Dpm	Disintegration per minute
DQO	Data Quality Objectives
ENTOMB	Entombment option
EPA	Environmental Protection Agency
ER	Environmental Report
FIR	Fast Irradiation Facility
FSS	Final Status Survey
Gal	Gallon
HASP	Health and Safety Program
HAZWOPER	Hazardous Waste Operations and Emergency Response
HEPA	High Efficiency Particulate Air
ID	Inside Diameter
In	Inch
ISO	International Standards Organization
Kw	Kilowatt
LLRW	Low Level Radioactive Waste
LSC	Liquid Scintillation Counter
MARISSIM	Multi-Agency Radiation Survey and Site Investigation Manual
µCi	Microcuries
mL	Milliliter
mrem	Millirem
NaI	Sodium Iodide
NEPA	National Environmental Policy Act
NMSS	Nuclear Material and Safety and Safeguards
NRC	Nucler Regulatory Commission
NRL	Nuclear Reactor Laboratory
NUREG	Nuclear Regulatory Guide
ODC	Other Direct Costs
OSHA	Occupational Safety and Health Act
QA	Quality Assurance
QAP	Quality Assurance Program
QAPP	Quality Assurance Program Plan
RCO	Radiaction Control Officer

RS	Reactor Supervisor
RWP	Radiation Work Permits
SHSO	Site Health and Safety Officer
SNM	Special Nuclear Material
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent dosimeter
TRIBA	Test, Research, Isotope Production, General Atomics
UA	University of Arizona
UARR	University of Arizona Research Reactor
UPM	University Project Manager
WRS	Wilcoxon Rank System
Yr	Year

UNIVERSITY OF ARIZONA RESEARCH REACTOR DECOMMISSIONING PLAN

1.0 SUMMARY OF DECOMMISSIONING PLAN

1.1 Introduction

The University of Arizona Research Reactor (UARR) is a TRIGA pool-type reactor designed and constructed by General Atomic Division of General Dynamics Corporation. The UARR is located within the University of Arizona Nuclear Reactor Laboratory (NRL) on the 325 acre campus of the University of Arizona (University) in Pima County Arizona in the city of Tucson. The University is about 65 miles north of the Mexican border at Nogales, AZ, 110 miles south east of Phoenix, AZ and 120 miles from the western border of New Mexico.

This Decommissioning Plan (DP) has been 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 safely decontaminate and remove radioactive materials, equipment, systems, components and soil associated with the NRL. This DP also describes the general process that will result in the removal of the UARR components and support systems to allow for unrestricted release of the NRL by the NRC. A final status survey that will be implemented to demonstrate compliance with default derived concentration guideline levels (DCGL) in support of the unrestricted release and license termination is also described.

1.2 Background

The University of Arizona campus is centrally located in the city of Tucson, Arizona, and is roughly bounded by East Speedway Boulevard, North Campbell Avenue, East Sixth Street, and North Park Avenue. The UARR is operated by the Nuclear Reactor Laboratory (NRL) under an operating license from the United States Nuclear Regulatory Commission (NRC). The NRL is located on the Main Campus, in the Engineering Building, on the first floor of the north wing. The physical location of the Engineering Building is shown in Figure 1-1.

The reactor is designated as a Mark I TRIGA reactor, operating at a maximum licensed steady state power of 110 kW (thermal), with a pulsing capability up to peak powers of approximately 650 MW. TRIGA stands for Test, Research, Isotope production, General Atomics.

The University expects to cease operation of the facility at or before the termination of the operating license on May 22, 2010. At the same time, the University will request the U.S. Department of Energy (DOE) to schedule shipment of the fuel as soon as possible after the end of operations.

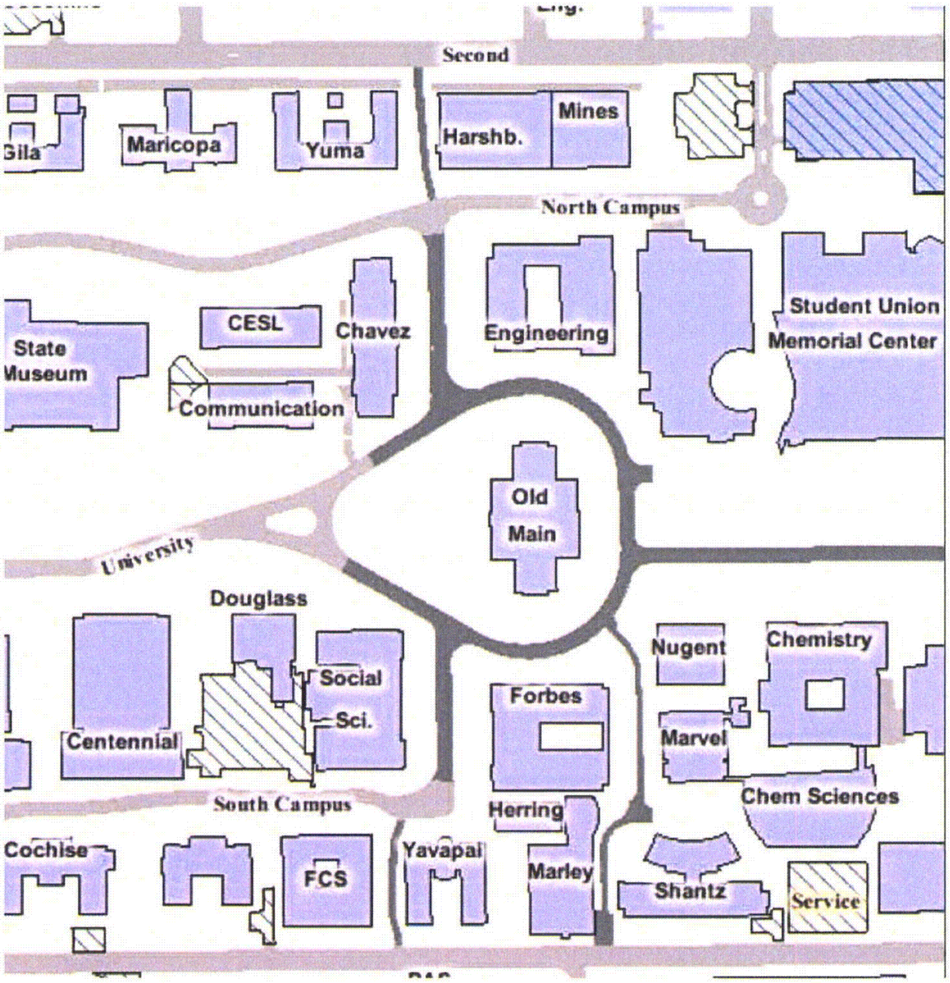


Figure 1-1. Location of Engineering Building on the UA Campus

1.2.1 Facility Description

Four adjacent rooms in the Engineering Building are permanently established as the Nuclear Reactor Laboratory and are designated a controlled access area. These are: 1) Room 122, the Control Room; 2) Room 124, the Reactor Room; 3) Room 216, and 4) Room 124A, the Equipment Storage and Experiment Setup Room. Room 216 is the room directly above the reactor room, which was originally designed to receive a beam of neutrons from the reactor. A 9 inch diameter hole penetrates the floor of this room directly above the center of the reactor core, and a 30 inch by 36 inch hatch to the roof above is directly over the hole. Little use was made of this beam capability, so during the refurbishment of the reactor in 1972, no provision was made in the new bridge for a hole to accommodate the beam tube. At this time the hole in the floor is capped and locked, and the room is used for storage of reactor supplies and departmental records.

Figure 1-2 shows a section of the first floor of the Engineering Building and the location of rooms 122, 124, and 124A in the NRL. All three rooms and room 216 are part of the NRL Security Area. Room 216 is directly above Room 124.

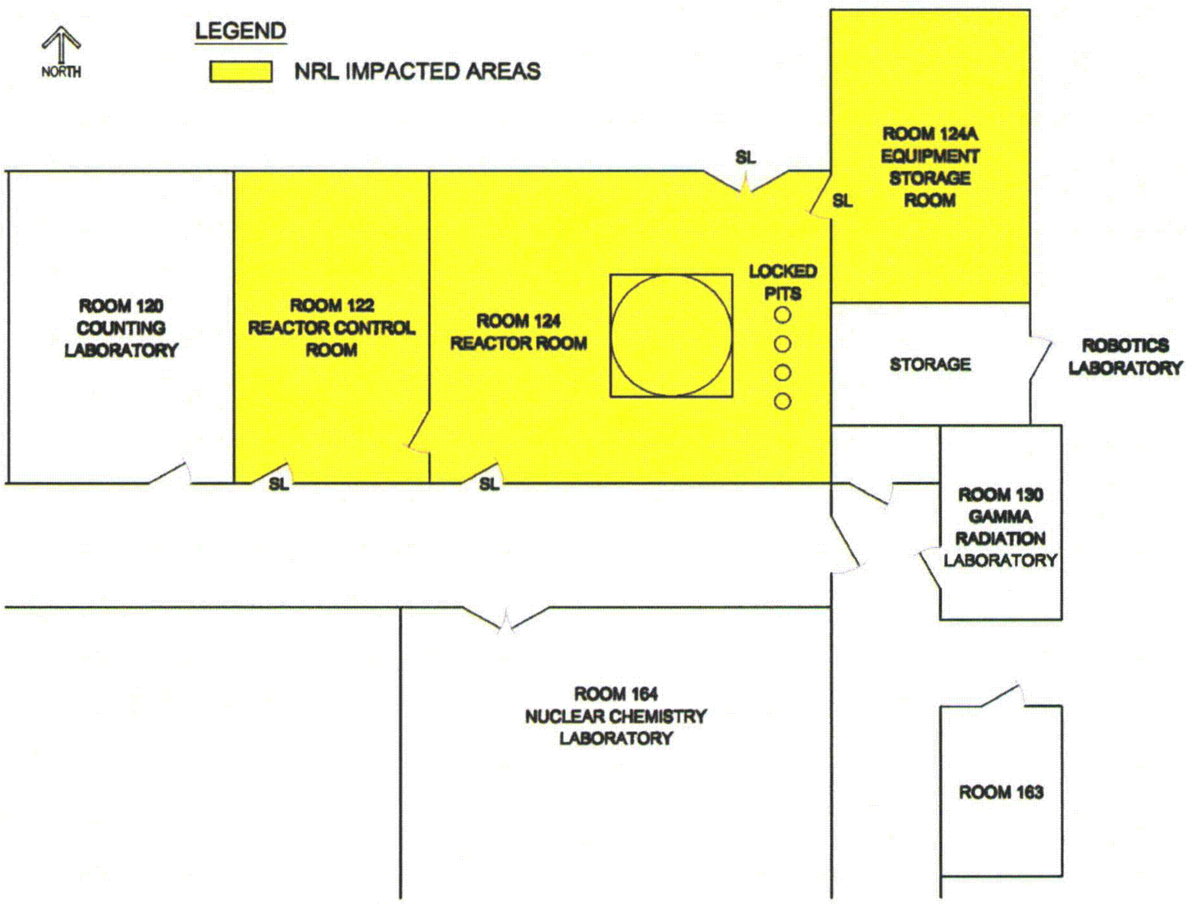


Figure 1-2. Section of the Engineering Building North Wing Showing the Reactor Laboratory

The reactor core is located in a pool-type tank, which is 21 feet deep and 6.5 feet in diameter, located below grade in Room 124, and shown in Figure 1-3. The pit contains a ¼ in. steel tank resting on a 1-ft-thick concrete slab. Approximately 8 in. of poured concrete surrounds the outside of the tank, except for a window 4 ft wide by 1 ft 10 in, high which was left in the concrete to allow for the insertion of a thermal column at a later date, and a 3 inch diameter circular opening which was intended to accept a van de Graaff generator beam tube. The steel tank served as the inner form for pouring the concrete and the outer form was a corrugated steel cylinder, which was left in place after pouring. A depression 1 foot square by 2 inch deep has been provided in the concrete at the bottom of the pit that will facilitate complete draining of the tank by a portable pump and hose should this become necessary. The inside of the steel tank is covered on the sides by a layer of Gunite approximately 2 in. thick and on the bottom by a layer approximately 4 in. thick. The entire inner surface of the Gunite is coated with Amercoat (an epoxy-base paint).

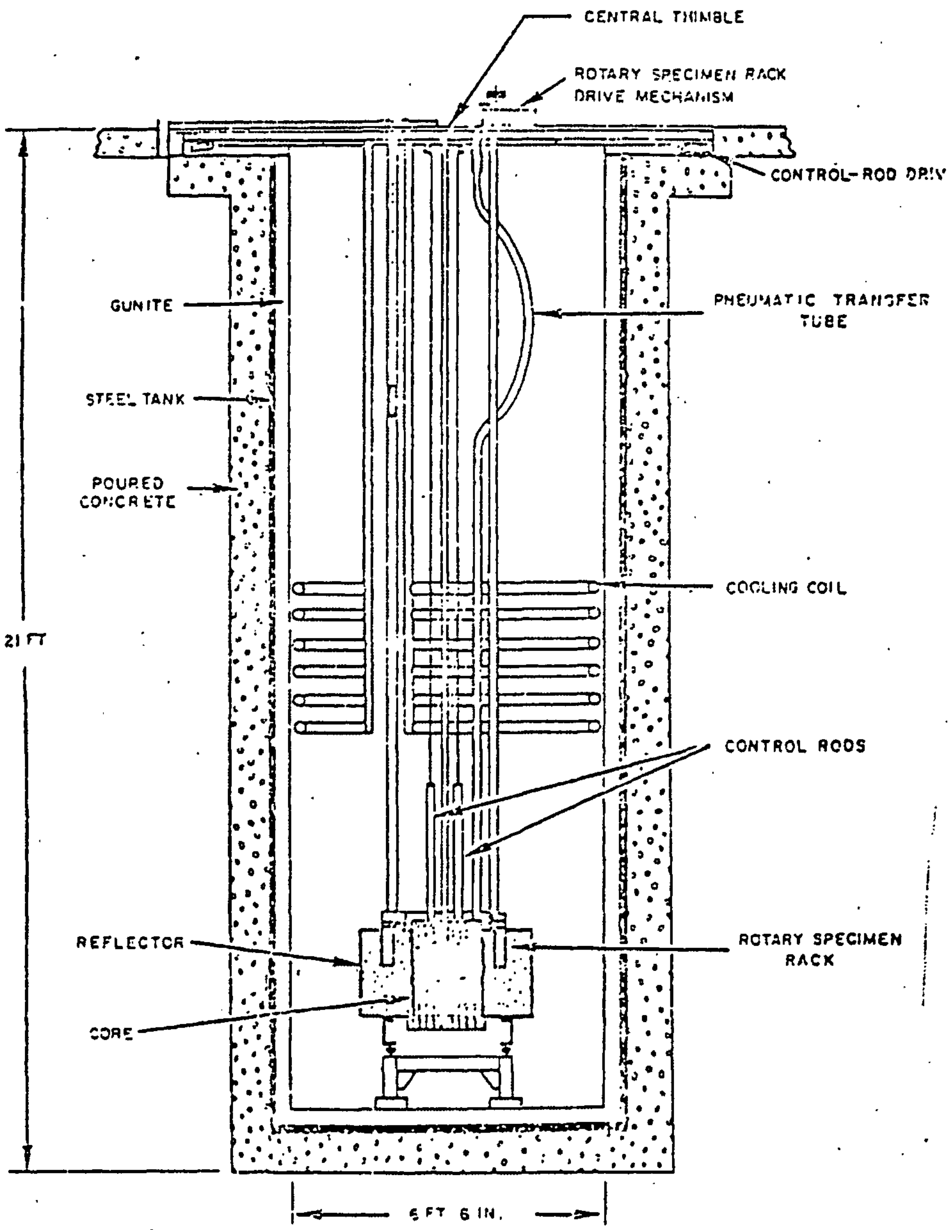


Figure 1-3. Elevation of the Reactor and Tank

1.2.2 History

The UARR reactor was constructed in 1958 and went into operation in December of that year. The licensed power was 10kW thermal, with operations of 30kW possible for short times. The original core loading consisted of 61 aluminum-clad fuel elements. Subsequently, two additional aluminum-clad fuel elements were obtained and the facility was licensed to allow operations at 100kW. The reactor was extensively updated in 1972.

In May of 1972, a new TRIGA control console, control rod drives, and bridge were installed. Based on a revised Safety Analysis, a license amendment was approved in June 1972 allowing for receipt and possession of additional fuel for a complete change over from aluminum-clad to stainless-steel-clad fuel. In December of that year, 87 partially used stainless-steel-clad TRIGA fuel elements were obtained, permitting operation in the pulsed mode. The original TRIGA console, bridge, control rods, control rod drives and all the aluminum-clad- fuel elements were given to the University of Utah for use in their TRIGA Reactor Facility.

In February of 1973, initial criticality with the stainless-steel-clad fuel was obtained with 71 fuel elements containing approximately 2.4 kilograms of U-235 and graphite reflector elements in non-fuel positions in the F-ring. In June of 1973, a neutron radiography tube was installed in the reactor pool. In December of 1975, a motor-driven reactivity oscillator was first placed in the reactor core, after which a fuel element instrumented with thermocouples was installed in the core in January 1976. In August of 1978, an aluminum-clad graphite thermalizer block was installed in the reactor pool.

In October of 1978, a license amendment increased the maximum reactivity insertion in the pulse mode from $2.10 to $2.50. In January of 1981, a new top grid plate was installed to allow vertical flux mapping and void coefficient measurements in the core.

1.2.3 Reactor Description

The reactor core forms a right circular cylinder and consists of a lattice of cylindrical fuel elements in water. The core is located inside a cylindrical graphite reflector 1 ft thick and 22 in. deep, as shown in Figure 1-4. The graphite reflector is completely clad in aluminum. A well to accommodate the rotary specimen rack is provided in the reflector such that the rotary specimen rack and the reflector are each individual water-tight assemblies. The reactor fuel consists of standard stainless-steel clad TRIGA fuel elements, instrumented elements, and fuel followers for the control rods. The fuel elements are standard TRIGA elements 1.47 in. diameter and 28.37 in. long. The U-235 enrichment of the fuel is less than 20 percent. Irradiated fuel is located either in the reactor core or in safe geometry storage racks in the reactor pool. The 5,000 gal of reactor pool water provides shielding of the gamma radiation produced during operation and from the irradiated fuel.

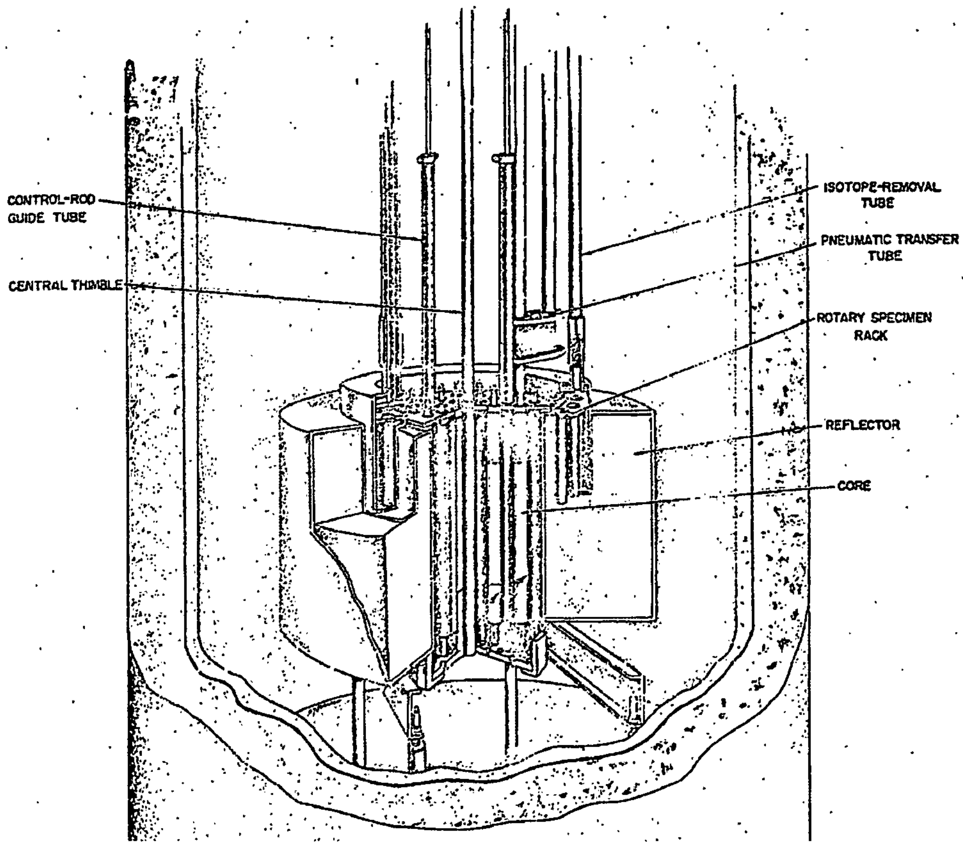


Figure 1-4. Reactor Core and Reflector Configuration

The fuel elements are supported and spaced by top and bottom aluminum grid plates. The bottom grid plate is ¾ in. thick, with holes to receive the lower end-fixtures of the fuel elements and holes to allow convective flow of coolant upward through the reactor core. The top grid plate is also ¾ in. thick, with 1.5-in. ID holes, for the fuel elements and the control rods. The holes serve only to determine the lateral position of the fuel elements and to permit withdrawal of the fuel elements from the core. Space for the passage of cooling water through the top grid plate is provided in part by three spaces machined in the top end-fixture of each fuel element. Cooling water may also flow through holes in the top grid plate between fuel elements, and in the narrow gap between the top grid plate and the reflector.

The three control rods are located near the center of the core and include a transient rod in position C-10, a shim rod in position D-10, and a regulating rod in position C-4. The holes in the bottom grid plate at these locations are 1.505 in. in diameter to accommodate fuel follower-type control rods. The drive assemblies for the control rods are fastened to mounting plates located on the bridge center channel. The shim and regulating rods have electrically driven rack-and-pinion drives. The transient control rod is actuated by an electro-pneumatic system, which is controlled from the reactor console. The transient rod drive system consists of a single-acting pneumatic cylinder with associated electrical and mechanical components, an air accumulator, and a three-way solenoid air valve.

Fuel Storage Racks. Thirty position fuel element storage racks on the floor of the reactor pool and 13 position holsters on the sides of the pool provide storage of fuel during fuel inspections and approach to critical experiments.

1.2.4 Experimental Facilities

Pneumatic Transfer System. The pneumatic transfer system (rabbit system) is used for neutron irradiation of single samples. It consists of two tubes connecting a sender receiver station in the F ring of the reactor core with a terminal station on the reactor bridge. The sample is packaged in an irradiation capsule, which moves in one tube. A blower provides the pressure difference for moving the capsule. The blower and filter are located under a floor plate in the reactor room.

Rotary Specimen Rack. The rotary specimen rack (lazy susan) is used for neutron irradiation of many samples simultaneously. The rack consists of an aluminum ring that can be rotated around the core, in which forty evenly spaced aluminum cups serve as holders for irradiation capsules. The ring can either be driven by an electric motor or rotated manually from the top of the reactor pit. Any cup can be aligned with the isotope removal tube (lazy susan access tube) for inserting and removing specimens. The rotary specimen rack is completely enclosed in a welded aluminum box and is located at approximately the level of the top grid plate. The specimen cups extend from the ring down to about 4 in. below the top of the active lattice.

Fast Irradiation Facility. The "FIR" is a fast neutron irradiation facility, which is inserted in a standard fuel element position. It is a tube whose terminus is lined with boron, cadmium, and gold, to absorb thermal neutrons while permitting fast neutrons to pass into the sample. Samples are lowered into the tube in an aluminum can on a nylon cord. Similar tubes without the lining of thermal neutron absorbing materials are used for in-core irradiation of single samples and to contain neutron detectors for experiments measuring reactor parameters.

Neutron Radiography Tube. The neutron radiography tube (beam tube) measure 12" diameter and permits the streaming of a near parallel, low flux beam of thermal neutrons from the reflector area of the reactor.

This beam is used to expose neutron radiographic pictures. When not in use, a scattering block and two shield plugs prohibit neutron or gamma radiation from reaching the surface of the reactor pool.

Graphite Thermalizer Block. The graphite thermalizer block is enclosed with a watertight cover of 1/8 inch 6061 aluminum and is permanently positioned west of the reactor core. The inner and outer curved surface of the block match the radii of the graphite core reflector and reactor pool, respectively, such that the block occupies the available space between the core reflector and the pool wall. The block is the same height as the core reflector and has three aluminum irradiation thimbles set into its top surface such that the bottom of the thimbles are approximately 5 inches below the vertical centerline of the reactor core. To access either of the thimbles, an access tube with O-ring seals is slipped into the thimble from above the pool surface. The tube is then pumped dry, allowing sample access to the thermalizer block.

Central Thimble. The central thimble is available to permit irradiation of experiments at the center of the core in the region of maximum neutron flux. It can also be used to provide a highly collimated beam of neutrons and gamma rays when it is emptied of water. The thimble is an aluminum tube with an inside diameter of 1.33 inches. It extends from the top of the tank through the two grid plates and terminates in a plug at a point approximately 7.5 inches below the lower grid plate. The tube is normally filled with water, but the water can be replaced with air or a gas such as helium so it can serve as a beam tube. The tube is not installed in the core.

1.2.5 Auxiliary Systems

Cooling System. The reactor fuel is cooled by natural circulation of the pool water. Heat is removed from the pool by a 7.5-ton vapor compression refrigeration system utilizing Freon-22 as the refrigerant. The compressor and air-cooled condenser are located on a concrete slab outside the building. The Freon is circulated through an aluminum evaporator (cooling coils), located in the pool. The evaporator is made from two concentric coils of aluminum tubing in an array approximately 5-1/2 ft ID, 9 in. thick and 4 ft deep. This shape provides clearance for the removal of reactor core components if necessary. The neutron and gamma fluxes at the location of the evaporator are too low to either produce appreciable activation or radiation-induced decomposition of the Freon.

Purification System. The purification system consists of a pump, fiber cartridge filter, mixed-bed type demineralizer, flowmeter, and surface skimmer connected by piping and valving. The components are located outside the building.

HVAC. The Reactor Laboratory is provided with a window-mounted exhaust fan with a minimum flow rate of 1000 cfm. When the continuous air monitor generates an airborne contamination alarm, the normal exhaust fan will stop and an emergency exhaust fan will start. In this mode, air is drawn through a high-efficiency particulate air filter and is exhausted via a vent on the building roof located a minimum of 50 feet above ground level.

Utilities. Potable water is provided by the university water system, and is backed up by the City of Tucson Water Utility. Demineralized water for makeup of the reactor pool is either purchased or distilled in a separate facility. Electric power for the facility is provided by the university electric power distribution system, which is supplied by the Tucson Electric Power Company. Transformers located outside the building step down the campus distribution from 4160 V to 120/220 V.

1.3 Reactor Decommissioning Overview

The University plans to remove the UARR from service, dismantle the reactor and its ancillary support systems, remove all radioactive materials from the NRL, and reduce the radioactivity to levels that will permit release of the licensed area for unrestricted use and allow termination of License R-52.

Many of the reactor components and systems are either activated or contaminated and will need to be segregated from non-radiological components and surfaces so that they can be disposed of as low level radioactive waste (LLRW). Building materials such as parts of the floor need to be removed and disposed of according to their radiological status.

Following nuclear fuel and SNM removal, the following major decommissioning tasks are necessary for site release. The sequence in which these tasks occur may vary:

- Remove loose equipment
- Remove hazardous materials (lead, cadmium) and asbestos
- Perform supplementary characterization
- Install temporary systems and prepare the facility for decommissioning operations

- Remove the control rod drives, rotary rack drive, and bridge
- Remove fuel storage rack, holsters, and cooling coils
- Remove the reactor structure, reflector, and irradiation components
- Remove and disposition pool water
- Segregate and package materials according radioactivity levels and classification
- Remove auxiliary systems (rabbit system, water purification, ventilation)
- Remove gunite, activated portions of the tank liner, concrete, and affected soils
- Decontaminate or remove the dry storage pits
- Decontaminate building surfaces
- Perform the final status survey (FSS)
- Ship waste
- Submit required reports that demonstrates to the NRC that the facility meets the release requirements
- Request license R-52 termination
- Restore the facility for future use by the University.

The UARR is expected to shut down operations on May 22, 2010. After fuel removal, the on-site decommissioning tasks are expected to start within one year and are anticipated to last about 4-6 months.

The Final Status Survey will be developed by the Decommissioning Contractor using the criteria provided in NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM) (NRC 2000).

1.4 Estimated Cost

The decommissioning cost estimate is summarized in Table 1-1. Tasks include the decommissioning contractors (DC) costs as well as subcontractor and other direct costs (ODC). Several portions of the cost estimate are subject to cost increases that cannot be fully quantified at this time. To account for this variability a contingency cost of 25% has been included in Table 1-1 to ensure that sufficient funds are available to cover costs that may result from incomplete characterization, unanticipated conditions or uncertainties in the project scope. Typically, these include external influences such as waste disposal rates or increased waste volumes from undiscovered or uncharacterized areas. In addition, the extended time duration between the development of the DP and the inception of work activities can influence the costs associated with changes in the economy and regulatory requirements.

Table 1-1. Decommissioning Cost Estimate

Major Project Activities	Cost
Preparation and approval of plans and procedures	**$ 113,987**
Site mobilization and training	**$ 47,604**
Facility preparation (hoist, waste handling logistics, etc.)	**$ 109,757**
Perform supplementary characterization	**$ 9,428**
Reactor bridge and control rod drive removal	**$ 44,578**
Remove cooling coils, fuel storage holsters, and fuel storage rack	**$ 53,493**
Reactor component removal including control rods, ion chambers, fission chambers, grid plates, reflector, rabbit piping, rotary specimen rack, irradiation tubes, thermalizer block, core support platform	**$ 136,643**
Pool water removal, cleanup, and disposition	**$ 51,852**
Remove gunite, liner, concrete tank, outer steel and soil	**$ 263,358**
Remove auxiliary systems	**$ 46,033**
Storage pits decontamination or removal	**$ 47,488**
Facility decontamination	**$ 18,850**
Source(s) packaging, transportation, and disposal	**$ 63,599**
Radiological waste packaging, transportation and disposal	**$ 312,624**
Final status surveys and report	**$ 54,727**
Facility restoration (backfill, pour slab, and miscellaneous)	**$ 29,395**
Demobilization	**$ 40,085**
University oversight	**$ 122,000**
Estimated Decommissioning Cost	**$ 1,592,549**
Contingency @ 25%	$ 398,137
Total Decommissioning Cost with Contingency	**$ 1,990,686**

1.5 Availability of Funds

In accordance with 10 CFR 50.75 (e)(1)(iv), the University of Arizona is a state institution and as such will provide financial assurance with a statement of intent containing a cost estimate for decommissioning, indicating that funds will be obtained when necessary.

1.6 Program Quality Assurance

The University will select a qualified DC to assist in the decommissioning of the UARR.

The DC will be responsible for developing a Quality Assurance Program (QAP), along with the associated costs, that is appropriate for both the decommissioning activities and final status survey of the NRL. The QAP will incorporate standard industry and regulatory requirements applicable to decommissioning project planning and management; decontamination, dismantling, and demolition; and radiological sampling, analysis and surveys. The QAP will be reviewed and approved by the Reactor Committee. The QAP will describe quality assurance activities associated with the following:

- Project Management
- Training
- Document and Procedural Control
- Data Management
- Recordkeeping
- Sample Collection
- Sample Analysis
- Radiological Surveys
- Waste Packaging
- Waste Shipping

The decommissioning project will incorporate the existing UARR Quality Assurance Program for the transportation of radioactive materials.

The DC will provide a Quality Assurance (QA) Manager responsible for management and day-to-day execution of the QAP. The QA Manager will be functionally responsible to the DC Project Manager to ensure that decommissioning activities including those of sub-contractors or off-site support meet the requirements of the QAP and other quality assurance documents.

2.0 DECOMMISSIONING ACTIVITIES

2.1 Decommissioning Alternatives

There are four alternatives available to UARR: 1) the no action alternative (SAFSTOR); 2) the entombment option (ENTOMB); 3) complete decontamination and structure demolition (DECON-A); and 4) complete decontamination and release of the structure (DECON-B). The DECON options are recommended by the NRC for non-power reactors.

The facility records and current facility characterization reveal minimal facility contamination from past UARR operations. The facility can be decontaminated with little impact to the structure. The UARR is located in the center of a growing university and the building space could be reused; therefore, decontamination and release of the facility (DECON-B) is the most feasible option. The reactor will be disassembled; all radioactive material greater than the DCGL will be removed from site, and the site will be released and restored for unrestricted use.

2.2 Facility Radiological Status

2.2.1 Facility Operating History

The UARR operated for 243,055 kW-hr between December 1958 and December 2008. Since many components were replaced in 1972, the operating history was 50,388 kW-hr from 1958 through June 1972. The reactor was operated for 192,667 kW-hr from July 1972 through December 2008.

Routine surveys are performed monthly. Any excessive contamination is reported immediately and areas are cleaned up. To date, there have not been any non-routine events such as accidents, spills, unplanned releases or the inadvertent spread of contamination that would lead to the conclusion that unidentified contamination will be encountered. Service irradiations have historically represented the most likely source of radiation and contamination from routine operation of the reactor. Irradiated samples normally remain in the NRL or are transferred to other authorized recipients.

Sealed sources are leak checked every six months and all results were within limits.

An examination of the facility history showed that very few non-routine occurrences took place in the facility. No accidents or spills were noted that resulted in facility contamination. The staff typically identified problems and followed-up with proper and complete corrective measures to ensure that the spread of radioactivity was minimized and contained or to ensure that the release of radioactive materials to the environment or surroundings was kept as low as possible. This has been generally confirmed by the lack of or low-level radioactive contamination found on surfaces throughout the facility and in the pool.

One noteworthy incident occurred in 1974, where an air monitor alarm was traced to a damaged fuel element. The fuel element released gas bubbles when the reactor operating at power. Inspection revealed a dented area near the bottom weld. The fuel element was placed in storage where it remained intact. No elevated contamination levels were noted in the pool water or building surfaces.

A small leak occurred in the cooling coil, in 1997, as observed by a freon film on the water surface. Underwater repair of the coil resulted in water intrusion into the coil. The freon dehumidifier was drained to restore the system to normal operation. Although the drained water revealed no radiological contamination, the cooling coil and in line components are assumed to be internally contaminated.

2.2.2 Current Radiological Status of NRL

In February 2009, a characterization of the NRL facility was performed to assess facility contamination levels. Field surveys were performed in the Reactor Room, the Control Room, the Equipment Storage and Experiment Setup Room, the Second Floor Storage Room, as well as ancillary systems. Measurements for fixed and removable alpha and beta/gamma contamination, and tritium, were conducted throughout the facility. All accessible building surfaces contained no radiological contamination above site release criteria established in Section 2.2.3. The storage pits and reactor tank contents were not characterized due to ongoing reactor operations. A report detailing the methodologies and findings of this characterization is attached in Reference 1.

The reactor components and portions of the tank are neutron activated. An activation analysis has been performed on the core internals and reactor tank to determine the activated product radionuclide concentrations of reactor core components. Results from this analysis indicate that all radioactive waste associated with the NRL is considered Class A LLRW and that no Class B&C LLRW is expected in the NRL. This activation analysis is available in Reference 2.

Prior to commencement of D&D activities, a limited characterization and review of facility documents pertaining to the period from the end of the characterization until removal of fuel will be conducted to validate that radiological conditions at the facility have not changed from those that were encountered during the characterization. Additionally, characterization of the reactor tank, internals and underlying site soils should be conducted to normalize results presented in the activation analysis and to verify that site soils are free of radiological contamination. The activation analysis does indicate that minimal soil activation is anticipated around the reactor tank.

Water samples are collected quarterly from the UARR pool. Gamma spectroscopy data indicates no activity significantly above background. The water does contain tritium at approximately $3x10^{-6}$ μCi/mL. The maximum allowable effluent concentrations for the release of tritium to the environment and sanitary sewerage, according to 10CFR20.1001 – 20.2401 are:

1) $1x10^{-2}$ μCi/mL of wastewater when diluted by the average monthly quantity released into the sewer system,
2) $1x10^{-3}$ μCi/mL when released to the environment as treated effluent water, and
3) No more than 5 curies of tritium may be disposed into the sanitary sewer system in one year. The tritium inventory in the pool is 17,600 L x $3x10^{-3}$ μCi/L = 53μCi.

The facility currently possesses several small beta and gamma sources each of which is less than 20 μCi. Additionally, some Co-60 sources and a Cf-252 source are present; however, these are managed under a separate radioactive material license and are not part of this project. A 4.71 Ci AmBe neutron source, doubly encapsulated in stainless steel, is currently positioned outside the graphite reflector in the southeast quadrant of the core.

The special nuclear material (SNM) inventory, not including the nuclear fuel, includes six fission chambers, 1,396 subcritical assembly slugs containing natural uranium, and 1,600 g of depleted uranium pellets. Fission chambers are typically lined with enriched uranium to enhance the ionization current. One fission chamber is located near the core and the other five are stored in Room 124A.

Table 2-1. Estimated Volumes and Masses of Radiologically Impacted Components and Systems

Component/ System	Material	Volume (ft^3)[(a)]	Mass (lb)
Reactor components	Steel, aluminum, graphite	126	6,255
Reactor tank gunite [(b)]	Concrete	60	6,019
Reactor tank [(c)]	Concrete	281	28,078
Reactor Tank liner [(c)]	Steel	42	4,155
Storage pits	Concrete	9	859
Ventilation and Water Filtration Systems	Steel	40	920
Soil	Soil	116	11,601
Total		**753**	**59,488**

a. Estimated packaged volume.
b. Assumes all gunite is removed.
c. Assumes a maximum of 8 ft height in the activation region is removed for reactor tank concrete and tank liner.

2.2.3 Release Criteria

The decommissioning alternative that has been proposed in this decommissioning plan does not require the dismantling of the Engineering Building or any structures. The results of the site and facility radiological characterization indicate that the building structure may be directly releasable without the need for extensive decontamination. This section provides the specific radiological criteria that will be applicable for unrestricted release of the building and termination of NRC license R-52.

The decommissioning alternative proposed in this DP includes the removal of all activated and contaminated materials, equipment and components. The remaining equipment and surfaces will be released to the NRC required 25 millirem annual (mrem/yr) Total Effective Dose Equivalent (TEDE) following guidance contained in MARSSIM[1]. The release criterion will be determined to have been met by demonstrating surface or volumetric activities, as appropriate, meet NRC screening values as presented in Tables 2-2 and 2-3 or to an alternative site-specific release criteria as discussed below.

<u>**Alternative Release Criteria**</u>

Alternative site-specific release criterion may be developed using a dose modeling software code such as the RESRAD family of codes. Argonne National Laboratory (ANL) developed the RESRAD family of computer codes under the sponsorship of the U.S. Department of Energy (DOE). The codes have been

[1] State of Arizona is an NRC Agreement State and as such, the Arizona Radiation Regulatory Agency has established Radiological Criteria for License Termination of 15 mrem/yr TEDE for licensees terminating under State of Arizona jurisdiction (R12-1-452)

used widely by DOE and its contractors, the U.S. Nuclear Regulatory Commission (NRC), U.S. Environmental Protection Agency (EPA), U.S. Army Corps of Engineers, industrial firms, universities, and foreign government agencies and institutions. The codes are pathway analysis models designed to evaluate potential radiological doses to an average member of the specific critical group based on a defined occupancy or site reuse scenario. If alternative site-specific release criteria would be developed, the appropriately conservative end use scenario would be approved by NRC.

Release Criteria for Surfaces

Surfaces and materials destined for reuse, recycling, or disposal as clean waste will be shown to be free of detectable surface contamination in accordance with the guidelines provided by the NRC in IE Circular 81-07 (NRC 1981). Monitoring for residual radioactivity will use instrumentation and techniques (background radiation levels, scan speed, counting times) necessary to detect activity no greater than 5,000 dpm/100cm^2 total and 1,000 dpm/100cm^2 removable beta/gamma contamination. All instruments shall be calibrated with radiation sources having an energy spectrum and instrument response consistent with the radionuclides being investigated. If alpha contamination is suspected, appropriate residual radioactivity measurements capable of detecting alpha activity no greater than 100 dpm/100cm^2 fixed and 20 dpm/100cm^2 removable will be used.

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

For surface tritium contamination, only 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 minimum detection no greater than 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 shall be measured in an LSC.

Building surfaces and materials destined to remain in place will be shown to meet an annual TEDE of 25 mrem/yr by demonstrating the surface activity meets NRC screening values as presented in Table 2-2 or to alternative site-specific surface release criteria developed using an appropriately conservative end use scenario.

Table 2-2. NRC License Termination Screening Levels for Surfaces

Radionuclide	Acceptable Screening Levels for Unrestricted Release (dpm/100cm^2)
Tritium (H-3)	1.2E+08
Carbon-14	3.7E+06
Manganese-54	3.2E+04
Iron-55	4.5E+06
Cobalt-60	7.1E+03
Nickel-63	1.8E+06
Technitium-99	1.3E+06
Cesium-137	2.8E+04

Release Criteria for Soils

To date, there have been no indications that the subsurface soils have been impacted by activities at UARR. Should future data indicate that subsurface soils contain radionuclides of interest associated with activities at UARR, they will be removed and the area sampled. Soils remaining in place will be shown to meet an annual TEDE of 25 mrem/yr by released by demonstrating the volumetric activity meets NRC screening values as presented in Table 2-3 or to alternative site-specific volumetric release criteria developed using an appropriately conservative end use scenario. Residual concrete and steel from the reactor tank will also be sampled to ensure that residual radioactivity meets the following NRC screening values for soils.

Table 2-3. NRC License Termination Screening Levels for Soils

Radionuclide	Default DCGL (pCi/g)
Cobalt 60	3.8E+00
Tritium	1.1E+02
Carbon 14	1.2E+01
Iron 55	1.0E+04
Nickel 63	2.1E+03
Cesium 137	1.1E+01
Europium 152	8.7E+00
Europium 154	8.0E+00

2.3 Decommissioning Tasks

2.3.1 Preparation Tasks and Activities for Decommissioning

Following reactor shut down and fuel removal, several activities will be conducted to prepare the UARR for decommissioning.

Remove Miscellaneous Equipment and Materials – Loose materials and equipment that is located in the NRL will be surveyed for release or packaged for disposal. Radioactively contaminated materials will be segregated from clean materials using the criteria described in Section 2.2.3. Clean materials that are not intended to support the D&D activities will be released for reuse or disposal according to survey procedures and the release criteria in Section 2.2.5 and the waste management program discussed in Section 3.2. These materials include furniture, tooling, experimental materials, and other items.

Isolate Inactive Systems- All inactive systems not required by either technical specifications, safety or for support of decommissioning activities will be isolated, de-energized, and drained. Unnecessary systems may be removed from the facility to avoid cross contamination during the removal of activated or contaminated items.

Remove Hazardous Materials and ACM- The DC will be responsible for the disposal of all hazardous material utilizing existing University programs. A licensed asbestos contractor will remove ACM utilizing industry practices. The DC will provide the asbestos workers with appropriate radiation safety training commensurate with the potential for exposure to radioactive materials. Characterization activities have not identified radioactively contaminated ACM. However, the DC will provide health physics support to the asbestos workers to ensure that, if ACM contaminated with radioactive material is identified, it will be controlled and segregated from ACM that is not contaminated with radioactive material.

Temporary System Installation- Temporary systems needed to support decommissioning activities such as temporary power, portable lighting, temporary ventilation systems and portable air monitoring systems will be installed. The DC will evaluate the existing facility jib crane for use on planned activities. Temporary lifting capability will be setup, if deemed necessary. A personnel retrieval or rescue system will be established, if deemed necessary during the DC planning effort. If facility modifications are required to facilitate waste handling, they will be performed following approval of the Reactor Committee.

Install Temporary Wheel Chair Access Ramp – Wheel chair access for the Engineering Building starts at the southeast corner of the Engineering building, proceeds north, and then passes the NRL through the northern most East-West corridor to get to the building elevators. Decommissioning activities may require the restriction of public access to this corridor. Therefore, a temporary ramp should be installed over the steps that are located in the southernmost East-West corridor to maintain proper wheelchair access to the building. The temporary ramp will be installed in accordance with applicable state and federal regulations. The design of the temporary ramp may require UA Facilities Management approval.

2.3.2 Tasks and Activities for Reactor Demolition

Best available standard industry techniques will be employed for D&D of the reactor and its components. These techniques may employ, but are not limited to, the use of long handled and standard tools, hydraulic cutters, torches, plasma arc torches, wire saws, needle guns, jackhammers, hand-scabblers, high pressure and ultra high-pressure sprayers and cutters. Any cost-effective tool or technique that is developed and available for use that achieves the goal of D&D while maintaining the principals of ALARA and with consideration to end state of the waste will be employed. Approved procedures and work packages will be developed by the DC designating the specific tool or technique to be employed. Any method that will minimize the spread of contamination such as portable high-efficiency particulate air (HEPA) ventilation systems or an encapsulation medium will be used. The utilization of temporary shielding will be used to maintain personnel exposures as low as reasonably achievable (ALARA).

The following D&D activities are presented as an overview and may not be followed in the sequence presented. ALARA, safety, and scheduling requirements may dictate that a different sequence be employed.

Remove the Reactor Bridge- The reactor bridge and bridge mounted components will be removed to allow access to the reactor core and pool components. The control rod drives and lazy susan drive will initially be removed. The bridge center channel assembly, deck plates, and supports will then be removed and set aside. The bridge and other components should be easily decontaminated and surveyed for release. These activities would be performed with the reactor pool full of water to provide shielding while working.

Remove Cooling Coil, Fuel Storage Rack, Fuel Holsters, and Tank Interference- The cooling, which has had the freon collected and purged, shall be rigged, cut, and removed from the tank. It can be set on the reactor room floor for size reduction and packaging as radioactive waste. Other items to be removed from the tank include the fuel storage rack, fuel holster, and any other items that require no use to support future tasks.

Remove Fission Chamber, Ion Chambers, Startup Source, Beam Tube, Experiment Tubes and Assemblies – The aluminum beam tube will be rigged, lifted from the pool, and size reduced for disposal. The pneumatic transfer system (rabbit) will be disconnected and removed. The fission chamber and ion chambers shall be removed using long-handled tools and dispositioned appropriately. The AmBe startup source will be removed using a long-handled tool and will be placed in a shielded drum, specifically approved for transportation and disposal of transuranic material.

Remove Reactor Components- The reactor components will be removed in a sequence specified by the DC. These components include the thermalizer, lazy susan rotary rack, reflector, grid plates, and reactor support platform. Removal of the reactor components should be performed while water is still in the reactor pool, if possible, to provide shielding and to maintain exposures ALARA. Long-handle tools and remotely operated equipment may be used to disassemble the components. The components will be lifted using the jib crane or other device, and will be placed in LLRW containers, with appropriate shielding to keep dose rates ALARA and within DOT and disposal facility acceptance limits.

Drain the Reactor Tank- The water in the tank will be drained, filtered, and discharged to the sanitary sewer system, as approved by the University. Radioactive sediments on the bottom of the tank can be removed by agitating the sediments to remove them during pumping, for subsequent filtering. Alternatively, the sediments may be left still and removed manually after the tank is drained. The tank water will be filtered to remove the suspended material, purified further, stored and sampled according to current procedures to allow discharge of sampled water to the sanitary sewer system. All water from any ancillary systems, such as the water purification system, should also be processed in the same manner. Filter media and water not meeting the discharge limits will be managed as LLRW.

Remove Gunite Layer and Activated Portions of the Reactor Tank- The inner gunite layer will be removed in its entirety, since it was in contact with the tank water. The gunite may be coated with an encapsulant to minimize airborne contamination levels during dismantlement. Following removal of the gunite, the activated areas of the lower tank will be surveyed to identify hotspots and to verify limits of the activation zone. Cores will be advanced through the concrete and both steel liners at locations in the mid plane core and axial centerline underneath the concrete to determine levels of activation in adjacent and underlying soils. The inner steel liner, concrete wall, and outer steel casing will be removed only in the activation region, where neutron activation products would be present. It is anticipated that the maximum amount of concrete /steel removal will include the entire floor area and a height of up to 8 ft above floor level. This also includes portions of the concrete and steel lined window adjacent to the tank. All removed tank materials will be packaged for disposal as LLRW.

Remove Activated Soils – Based upon the activation analysis, it is anticipated that minor soil remediation will be required with the highest activation expected in underlying soils at the axial centerline.

The removal of the lower portion of the Reactor Tank and activated soils will require a tank and soil removal plan to ensure that exposed soils in the sidewalls and the remaining concrete /steel tank will be shored to permit safe conduct of work activities. This plan shall be reviewed and approved by a Professional Structural Engineer supplied by the DC. The plan will specify methods to support soil in exposed sidewalls the upper portions of the reactor tank left in place.The DC is responsible for the costs associated with the development, and implementation of the tank and soil removal plan.

Characterize Remaining Tank Liner and Concrete. The remaining portions of the tank liner and concrete will be surveyed and sampled to ensure they are within the volumetric and surface release limits specified in Section 2.2.3. The tank will be decontaminated, as necessary. If the entire tank requires removal, it will be performed in accordance with a detailed plan that ensures a safe working environment, and allows for final status surveys and verification. The approach will either involve performing surveys and backfilling in lifts, or use of a designed shoring system.

2.3.3 Support Systems Removal

The remaining contaminated systems associated with the reactor are the Water Purification System and portions of the Reactor Cooling system.

The Purification System will be disassembled and packaged after the reactor tank has been drained and all water has been processed. Following electrical isolation and draining, the filters and resin will be removed, the pump, motor, piping, and vessels will be removed, and all materials will be packaged. The piping travels through a room that is not part of the license, to the demineralizer and pumps located outside. These areas will be surveyed following the removal activities to ensure no residual radioactivity remains. The components of the system will be managed in accordance with the criteria in Section 2.2.3. Any component that cannot be released will be packaged for disposal as LLRW.

The reactor cooling system cooling coils are potentially internally contaminated, due to a past leak. The refrigeration unit will de-energized, verified that the refrigerant was removed and collected, and disassembled. The cooling coil will be managed as LLRW and the refrigeration unit will be released in accordance with the criteria in Section 2.2.3.

There are 6 dry storage tubes located in the floor of the reactor area. These tubes will be emptied and their contents released per Section 2.2.3 or packaged as LLRW for shipment. The tubes will then be surveyed and decontaminated, as necessary.

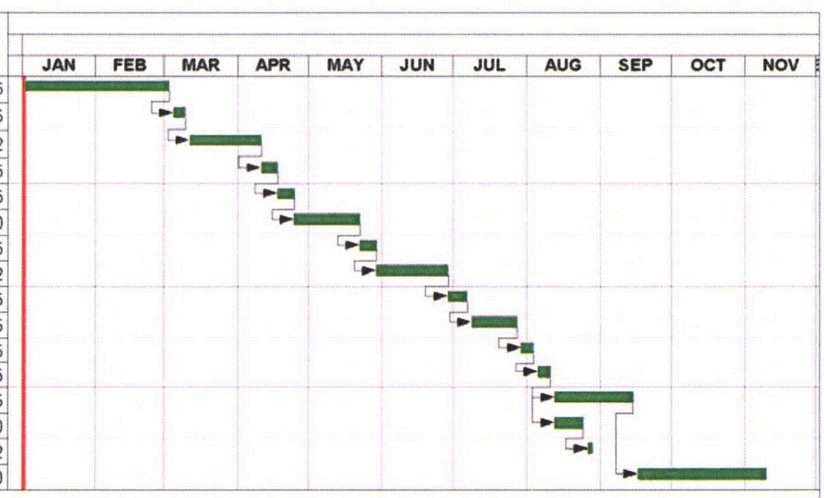
2.3.4 Schedule

The project time duration, from contract award to a Decommissioning Contractor through release of the site for unrestricted use, is approximately 11 months. The proposed decommissioning schedule is presented in Figure 2.1. Changes to the schedule may be made at the University's discretion as a result of resource allocation, availability of a radioactive waste burial site, interference with ongoing University activities, ALARA considerations, further characterization requirements, and/or temporary on-site radioactive waste storage operations. The project schedule duration is consistent with other recently completed university research reactor decommissioning projects.

The schedule includes all activities from project planning through NRC review of the Final Status Survey Report and license termination. The schedule is based on a 5-day work week with the number of work days listed for each activity.

Figure 2-1. Proposed Project Schedule

Act ID	Description	Orig Dur
10	Project Planning	45
20	Mobilization and Training	5
30	Facility Preparation	22
40	Remove Control Rod Drives and Bridge	5
50	Remove Fuel Storage Rack, Cooling Coils,	5
60	Remove Reactor Components	20
70	Drain Pool, Process Water	5
80	Remove Gunite, Liner, Concrete, Soil	22
90	Remove Auxiliary Systems	5
100	Decontaminate Storage Pits	15
110	Decontaminate Surfaces	5
120	Perform Final Status Surveys & Sampling	5
130	Final Status Survey Report	25
140	Site Restoration	10
150	Demobilization	2
160	NRC Review and License Termination	40



2.4 Decommissioning Organization and Responsibilities

The D&D Project and the UARR is under the supervision of the Nuclear Reactor Lab Director who is responsible for assuring that all D&D activities are conducted in a safe manner and within the requirements of the UARR NRC License, this Decommissioning Plan, the UARR Radiation Protection Program, and the provisions of the Reactor Committee. The University will appoint a University Project Manager (UPM) to oversee the decommissioning process. The following duties, as a minimum but not limited to these, will be assigned either to the UPM or the Reactor Supervisor (RS).

- Selecting a decommissioning contractor in accordance with UA procurement guidelines (UPM)
- Overseeing the decommissioning contractors performance relative to the terms of their contract (UPM)
- Overseeing the decommissioning contractors performance relative to the Decommissioning Plan (UPM)
- Overseeing the decommissioning contractors performance relative to all subsequent plans and procedures (UPM)
- Overseeing all activities of contractor personnel on the licensed site (RS)
- Ensuring that all decommissioning activities are performed in compliance with applicable regulations and license conditions (RS)
- Review of all plans and procedures required for decommissioning (RS)
- Reviewing and submitting to the Reactor Committee all needed changes and subsequent plans and procedures that do not change the original intent or result in an unreviewed safety question (RS)
- Communicating with all appropriate regulatory agencies (RS)
- Communicating with the Nuclear Regulatory Agency, The University Administration, and the decommissioning contractor and sub-contractors (RS)

The University Radiation Control Office (RCO) is responsible for monitoring and overseeing radiological safety at the UARR and NRL. The RCO has the responsibility and authority to stop any plan or activity that has the potential to result in an unacceptable radiological hazard.

The function, responsibilities, and makeup of the Reactor Committee are defined in the Technical Specifications (University of Arizona Research Reactor License R-52). Among those responsibilities but not limited to them are:

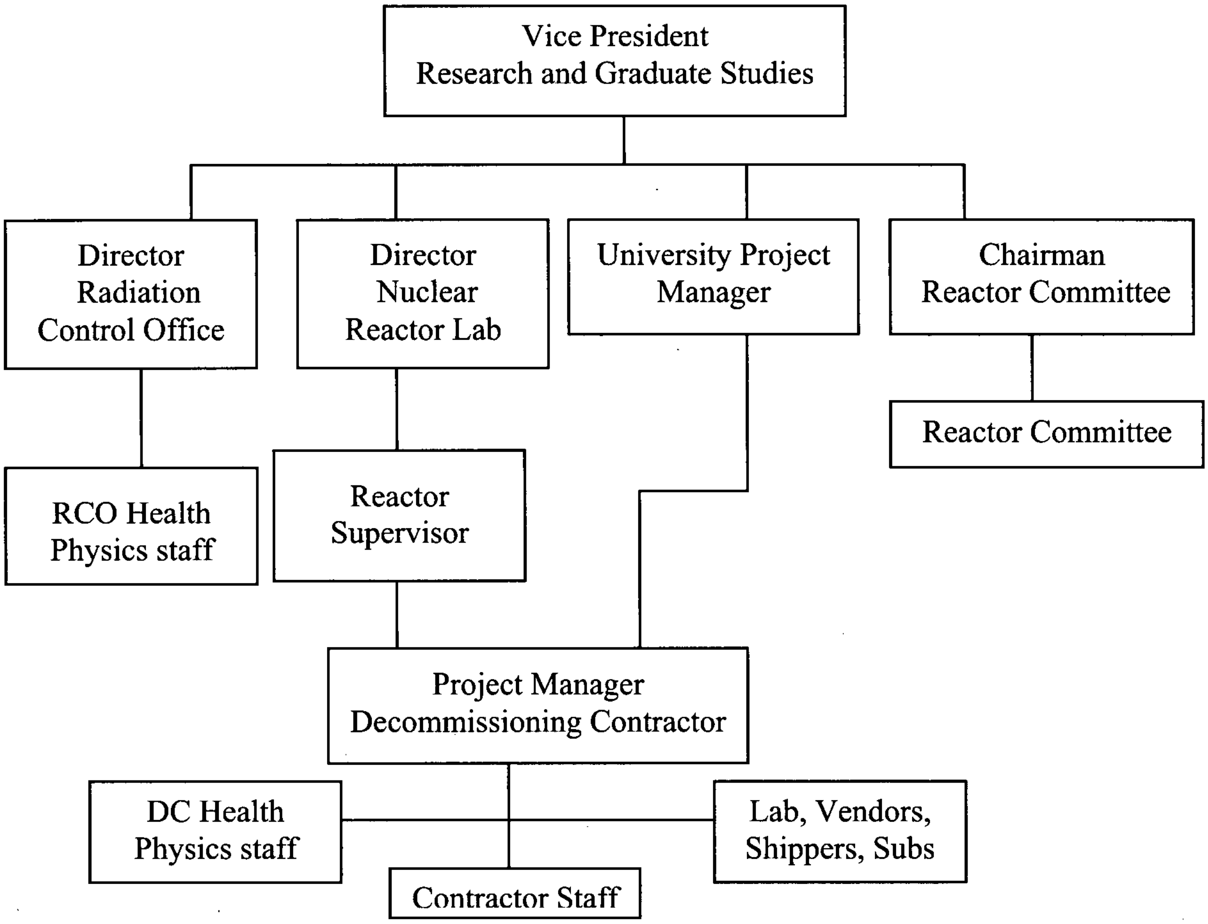
- Approval of all plans and procedures required for decommissioning.
- Review and approval of all proposed changes to the facility, procedures and Technical Specifications and Decommissioning Plan.
- Determination of whether a proposed change, test or experiment would constitute an unreviewed safety question or a change in the Technical Specifications as required by 10CFR 50.59, and review and approval of required safety analysis.

The DC is responsible for the development, implementation, and costs of all safety and regulatory compliance programs that are applicable to the UA NRL decommissioning. The responsibilities include, but are not limited to, the following:

- The safe and regulatory compliant implementation of the UA Decommissioning Plan.
- Development, implementation, and associated costs of a Radiation Safety program compliant with 10CFR20.
- Development, implementation, and associated costs of an OSHA compliant Health and Safety Program.
- Development, implementation, and associated costs of work plans and procedures necessary for the safe and compliant decommissioning of the UA NRL.
- Acquiring applicable permits for radiological waste disposal and transportation, along with applicable costs.

The following organizational chart in Figure 2-2 shows the lines of reporting at the UA NRL.

Figure 2-2. Organizational Chart



2.5 <u>Training Program</u>

2.5.1 General Site Training

A general training program will be designed and implemented by the DC and approved by the RCO 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 but is not limited to:

- 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 DC'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 DC'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 DC Project Manager or Site Safety Officer. If additional radiation safety training is required, it will be provided by the site RSO as necessary.

2.5.2 Radiation Worker Training

The UARR D&D operations will be performed by the DC and its subcontractors. As such, the DC 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 DC Site RSO will be responsible for on-site radiation safety training of workers and verifying pervious training and qualification. The DC's radiation safety training program will be administered by a Site Radiation Safety Officer who will approve all training materials and qualification of workers. The University RCO may 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 documented equivalent radiation worker training from another site or employer maybe waived from taking the training but must take the written and practical factors examinations.

2.6 Contractor Assistance

The University will select a qualified contractor to perform all or parts of the UARR 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 DCs and subcontractors. Prior history and performance of the prospective contractor on non-power reactor or similar decommissioning projects will be used to help the University select a qualified contractor to perform the facility D&D.

The selected DC 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.

2.7 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:

Code of Federal Regulations

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 DC 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 University RCO will administer the D&D project Radiation Protection Program. The DC will supplement this Program with detailed plans and procedures related to facility D&D. The University RCO, the DC Site RSO and DC 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 DC will prepare a Radiation Protection and ALARA Plan that will incorporate provisions for minimizing occupational and public radiation exposures. The Plan will describe specific administrative and engineering controls that will be put in place during specific D&D project activities. Examples of administrative and engineering controls include limiting access to certain areas, mock-up training, use of remote-handling devices, temporary shielding, containment structures, portable HEPA filtered ventilation, and specialized protective equipment and respiratory protection.

3.1.2 Health Physics Program

The project Health Physics Program will be implemented under the authority of the University RCO with the assistance of the DC 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.

- Monitor personnel internal and external exposure in accordance with 10 CFR 20 requirements. The University RCO will provide and manage TLD's; however, full compliance and costs (other than TLDs), are the responsibility of the DC.

3.1.3 Dose Estimates

The total estimated occupational exposure to complete the UARR Decommissioning Project is 2.39 person-rem. The dose estimate for decommissioning of the UARR was prepared using the individual work activity durations and work crew sizes, based upon the results of the characterization results to date and was based upon recent experience in performing similar activities at the University of Washington, combined with ENERCON's ongoing experience at other sites. Using these individual work activity durations, and work crew sizes and characterizations results, a dose estimate was generated for each activity. The estimated doses were then compared to the actual doses experienced during similar activities performed as part of other decommissioning efforts. The doses were then adjusted based upon expected changes in the characterization results due to the anticipated shutdown of the reactor in 2010. The doses from each activity were categorized and are provided by those categories in Table 3-1.

This estimate is provided for planning purposes only. Detailed exposure estimates and exposure controls will be developed in accordance with the requirements of the ALARA program during detailed planning of the decommissioning activities.

The primary doses expected to be received by D&D project workers will be from external exposure to activated metals and concrete, with little dose expected from internal exposure. External exposure will be monitored using whole-body and extremity TLDs, 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.

The dose estimate to members of the public as a result of decommissioning activities is estimated to be negligible. This is because the area immediately surrounding the facility is under the control of the Reactor Supervisor and because the area where decommissioning activities are taking place are fully contained within the facility (with the exception of loading and unloading of shipments of equipment and radioactive materials).

Table 3-1. Project Dose Estimate

Task	Task Name	Time (hours)	Number of People	Average Dose Rate (mR/hr)	Collective Dose (person – rem)
1	Project Planning	80	2	0.05	0.008
2	Site Mobilization and Training	40	6	0.05	0.012
3	Facility Preparation	160	6	0.1	0.096
4	Remove Control Rod Drives and Bridge	40	6	0.5	0.12
5	Remove Cooling Coil, Fuel Storage Rack, and Interference	80	6	0.6	0.288
6A	Remove Reactor Ancillary Components	200	6	0.5	0.6
6B	Remove Reactor Core Components	1	6	100	0.6
7	Drain Pool and Process Water	40	4	0.5	0.08
8	Remove Gunite, Liner, Tank, Soil	160	6	0.5	0.48
9	Remove Ancillary Systems	40	4	0.1	0.016
10	Decontaminate Storage Pits	80	4	0.2	0.064
11	Decontaminate Surfaces	40	4	0.1	0.016
12	Perform FSS	40	3	0.05	0.006
				Total	2.39

3.2 Waste Management

The DC will implement a Waste Management Plan at the UARR D&D project. The Waste Management Plan will be submitted to the Reactor Laboratory Director 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. The DC is responsible for acquiring applicable waste permits and for all costs associated with waste management, including, but not limited to, radiological surveys, characterization and sample analysis, packaging, transportation, and disposal fees. At the sole discretion of the University, waste disposal contracts may be secured prior to decommissioning activities or selection of the DC contractor. Minimal generation of hazardous wastes including asbestos is expected during the course of this project. The DC is responsible for all costs associated with permitting and disposal of any hazardous wastes generated during the decommissioning of the UA NRL.

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 DC 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 existing hazardous waste disposal system.

Clean construction and demolition waste will be released by the DC according to release criteria specified in Section 2.2.3. Construction waste will be disposed of at the disposal facility being utilized by the University at the time the material is generated.

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 or recycle, and Class A LLRW. Additionally, mixed wastes and non-radiological hazardous waste will be segregated from LLRW. Based on the site characterization and reactor activation analysis, Class B and C LLRW are not expected at the UARR.

3.2.1 Fuel Removal

The fuel will be removed from the UARR reactor after the shut down date of May 22, 2010 and transferred to the U.S. Department of Energy for reuse.

3.2.2 Radioactive Waste Processing

The UARR 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 DC 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 rubblization beyond that required for demolition is not expected to occur on site. Complying with written procedures, standard work practices, and operating with the limits of the NRC license 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.

3.2.3 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. 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 UARR.

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

DC 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 DC will prepare a site specific Health and Safety Plan (HASP) to document safety requirements and accident response procedures.

All DC 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 DC subcontractors will be determined by the DC Site Health and Safety Officer (SHSO) based on the specific task the subcontractor is performing.

The HASP will be reviewed and approved by the University's Risk Management Department. The HASP will direct site activities necessary for ensuring that the UARR D&D project meets occupational safety and health requirements for protection of project personnel. The functional responsibility of the HASP will be to ensure compliance with the Occupational Safety and Health Act (OSHA) of1973. Arizona adopts federal OSHA standards by reference and enforces OSHA standards contained within 29CFR, parts 1910 (General Industry), and 1926 (Construction). The DC implements the HASP on-site.

As a minimum, the HASP 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 Radiological Accident Analyses

Potential radiological accidents during decommissioning the UARR were evaluated by determining UARR components and areas that contain the highest radioactive material inventory. The proposed decommissioning activities and methods in which radioactive material could be released to the work area or environment were considered. Since all special nuclear material will have been removed prior to decommissioning, the majority of the accidents discussed in the current license are not applicable. The accident identification process was supplemented by reviewing experiences at other non-power reactor decommissioning projects. The following radiological accidents were considered to present the highest potential consequences:

- Fire
- Spill tritium loaded water into the environment
- Release airborne contamination to the environment
- 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 most likely result in additional unplanned external exposures. There will be no fissile materials located on site that could result in a criticality incident.

3.4.1 Fire

The consequences of a fire during decommissioning of the UARR were considered and are not significantly different than the consequences of a fire during reactor operations. Most materials are metals, concrete, or similar non-combustible materials. Although some torch cutting operations may be performed during decommissioning, the likelihood is low that a fire would start or that a fire could become intense enough to release radioactive material.

Dry radioactive waste is normally collected and packaged, to limit the volume of dry radioactive waste available for consumption by fire to a few pounds and lower the potential for a fire to consume additional waste collections. Any fire in dry radioactive waste would be limited to a few microcuries of radioactivity.

3.4.2 Spill Tritium Loaded Water

The spilling of tritium-loaded water could occur during pool water pumping or removal operations. Hoses could leak or break, resulting in an uncontrolled release. To mitigate the extent of such releases, processes involving contaminated liquids will only be operated with personnel present. Personnel will watch for leaks and spills and respond by shutting down the activity. This will not allow for additional water to leak from the system. A spill kit will be readily available to respond to any incidents.

The total tritium inventory in the pool is less than 100 μCi. All other radionuclides in the pool water are at negligible concentrations. If the entire pool inventory were released to the environment and conservatively assumed to evaporate in one day, releasing 100 μCi, the maximum exposure to an individual would be 6.4 mrem.

3.4.3 Release of Airborne Contamination

An uncontrolled release of airborne radioactivity could occur during cutting and demolition activities involving contaminated or activated materials, such as removal and segmentation of reactor components, or removal of tank steel and concrete. Such activities may take place inside temporary containment structures equipped with local HEPA filter ventilation systems.

Temporary containment systems with local HEPA filter systems will likely vent to the NRL rooms or tie into existing building ventilation. A failure in the HEPA filter system could result in the uncontrolled release of airborne radioactive materials. A 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.

While the actual concentrations of airborne radioactive materials are unknown at this time, 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.

3.4.4 Transportation Accidents

Various forms and quantities of radioactive waste will be shipped from the UARR during the D&D project. The dose consequence from transportation accidents could be higher than the contamination accident scenarios described above 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 UARR reactor and support components will be removed prior to site release. Consequently, the Final Status Survey (FSS) will include only the exposed soils and concrete in the reactor pit and the balance of building surfaces in rooms covered by license R-52. The DC is responsible for the implementation, along with the associated costs, of the FSS.

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 2.2.3. 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 an FSS Plan by trained DC technicians experienced in performing FSS. The technicians will follow written procedures regarding surveys and sampling, sample collection and handling, chain-of-custody, and recordkeeping. The FSS Plan will define sampling locations, required analysis, and survey types. Any additional release criteria set forth by the University or the State of Arizona will be contained within the FSS Plan which will direct surveys or sampling efforts required to demonstrate compliance with such criteria.

The FSS may include surface gamma surveys using sodium-iodide (NaI) gamma scintillation detectors. Surface and subsurface soil samples will be collected using either a random-start grid pattern or randomly generated locations as appropriate commensurate to the classification of the survey area. Soil samples will be analyzed for contaminants of concern using standard analytical methods including liquid scintillation counting for hard-to-detect beta-emitting radionuclides (i.e., Carbon-14 and tritium) 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 UARR site satisfy the decommissioning criteria provided in Section 2.2.3. The Data Quality Objectives (DQO's) in the MARSSIM survey approach will provide a 95% confidence level for the false negative (Type I error) in demonstrating that the site meets the criteria. Typically, the false positive (Type II error) will also be defined as a 95% confidence level, but may be modified to apply to a specific situation. Therefore, the Type I decision error will be 5-percent. The decision error rates are used in determining the required number of samples necessary in each survey unit as well as the required minimum number of data points used for the final nonparametric statistical test performed to evaluate contaminant concentrations in the survey units against release criteria. DQOs, will be fully described in the FSS Plan and will include limits on the sensitivities of survey and analytical methods.

The QAPP will incorporate standard regulatory and industry measures applicable to the FSS. The QAPP will be reviewed and approved by the Reactor Committee.

4.3 Identification and Classification of Survey Units

4.3.1 Method for Classification

Survey units are classified based on contamination potential according to the methods described in MARSSIM. In general, there are two overall classifications, non-impacted and impacted. Non-Impacted areas have no reasonable potential for residual contamination because there was no known impact from facility operations.

Impacted areas may contain residual radioactivity from facility operations. Based on the levels of residual radioactivity present, impacted areas are further divided into Class 1, Class 2 or Class 3 designations. Class 1 areas have the greatest potential for residual activity while Class 3 areas have the least potential for impacted areas. Each classification will typically be bounded by areas classified one step lower to provide a buffer zone around the higher class. Exceptions occur when an area is surrounded by a significant physical barrier, such as a wall, that would make transport of residual activity unlikely from one area to the adjacent area. In such cases, each area will be classified solely on its own merit using the most reliable information available. The class definitions provided below are from Section 4.4 of the MARSSIM.

Class 1

> "Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys). Examples of Class 1 areas include: 1) site areas previously subjected to

remedial actions, 2) locations where leaks or spills are known to have occurred, 3) former burial or disposal sites, 4) waste storage sites, and 5) areas with contaminants in discrete solid pieces of material high specific activity. Note that areas containing contamination in excess of the $DCGL_w$ prior to remediation should be classified as Class 1 areas."

Class 2

"These areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_w$. To justify changing an area's classification from Class 1 to Class 2, the existing data (from the HSA, scoping surveys, or characterization surveys) should provide a high degree of confidence that no individual measurement would exceed the $DCGL_w$. Other justifications for this change in an area's classification may be appropriate based on the outcome of the DQO process. Examples of areas that might be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form (e.g., process facilities), 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper walls and ceilings of some buildings or rooms subjected to airborne radioactivity, 5) areas where low concentrations of radioactive materials were handled, and 6) areas on the perimeter of former contamination control areas."

Class 3

"Any impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_w$, based on site operating history and previous radiological surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification."

The size of a survey unit is directly affected by its classification. Section 4.6 of MARSSIM provides suggested sizes for survey units. However, as stated in MARSSIM, the suggested survey unit sizes were based on a finding of reasonable sample density and consistency with commonly used dose modeling codes. It is expected that the majority of the UARR and NRL will be designated as a Class 1 area. MARSSIM limits the size of a Class 1 survey unit to 2,000 square meters for an open land area and 100 square meters for a building surface. Therefore, the building footprint of the facility may include only one

Class 1 survey Unit while the land area surrounding the Class 1 survey unit will likely be designated as a single Class 2 or Class 3 survey unit to a maximum survey unit size of 10,000 square meters.

For standing buildings, MARSSIM recommends 100-m^2 of floor surface in a Class 1 area as a survey unit size based on the dose model assumption that a 100-m^2 office space would be occupied. The source term attributed to the total area in this case is essentially the 100-m^2 floor surface which equates to 180-m^2 if the lower walls (up to 2 meter height) are included. If there is a potential for residual contamination on the upper surfaces, i.e. the ceiling and walls above 2-m, the area may be broken into multiple survey units or the number of required data locations may be adjusted to account for the increased area.

Table 4-1 summarizes MARSSIM recommendations for survey unit sizes based on their type and classification.

Table 4-1: Recommended Survey Unit Sizes

	Minimum[2] / Maximum[3]	
Classification	**Buildings**	**Open Land**
Class 1	10-m^2 / 100-m^2[4]	100-m^2 / 2,000-m^2
Class 2	100-m^2 / 1,000-m^2	2,000-m^2 / 10,000-m^2
Class 3	1,000-m^2 / No limit	10,000-m^2 / No limit

4.3.2 NRL MARSSIM Classifications

The areas of the NRL have been classified per the methods described in Section 4.3.1, from the operating history of the NRL, and from the radiological characterization performed in February 2009. The MARSSIM classifications are described below and summarized in Table 4-2.

Class 1 Areas – The reactor pit is the only Class 1 area in the NRL. Areas in the bottom of the reactor pit will be greater than the DCGLS.

Class 2 Areas – The Class 2 areas of the NRL are limited to the floors and walls up to 2 meters of the Reactor Room and the Equipment Storage Room, Room 124 and Room 124A, respectively. Radioactive

[2] Recommended minimum size

[3] From MARSSIM Section 4.6

[4] Floor area of 100-m^2 plus the lower walls (2-m high) for a total area of 180-m^2

material and sources were handled in these areas, however, if residual contamination is found, it should be a small fraction of the DCGL.

Class 3 Areas – The Class 3 areas of the NRL are the walls above 2-meters in Rooms 124 and 124A; all surfaces in Room 122, the Reactor Control Room; and all surfaces in Room 216, the Second Floor Storage Room. The walls above 2-meters in Rooms 124 and 124A have very little potential to have any level of contamination, but there are no barriers to prevent possible contamination from being spread to these areas. There is also very little potential for any contamination in Rooms 122 and 216, but these rooms are a part of the NRL licensed area and are to be included in the FSS.

Table 4-2: NRL MARSSIM Classifications

Survey Area		MARSSIM Classification
Reactor Pit	All surfaces	1
Room 124	Floor and Walls <2m	2
	Walls >2m and Ceiling	3
Room 124A	Floor and Walls <2m	2
	Walls >2m and Ceiling	3
Room 122	All surfaces	3
Room 216	All surfaces	3

4.4 Data Collection

Survey methods are applied differently depending on the data requirements of a survey area. For example, removable activity measurements provide little, if any, benefit when attempting to assess the radiological conditions in an excavation. Conversely, assessing a building surface via volumetric sampling would provide the necessary data, but at great costs of time and money. This section will discuss the steps necessary to strike a reasonable balance between data needs and ease of survey performance based on the data needs of the survey area.

4.4.1 Buildings, Equipment, and Components

Buildings, equipment, and components that are destined to remain after license termination require the following surveys to demonstrate they meet the appropriate release criteria.

Scans

Buildings, equipment, and components require two-stage scan measurements as part of the FSS process at appropriate coverage rates and speeds. Gross beta and/or gross alpha measurements are utilized as appropriate to the potential contamination. The measurements typically are performed at a distance of 1-cm or less from the surface. Adjustments to scan speed and distance may be made in accordance with approved procedures.

Direct Measurements

Direct measurements are required for buildings, equipment, and components as part of the FSS process. The required quantity of direct measurements is a calculated value. The calculation is described in MARSSIM. Direct measurement data for buildings, equipment and components is collected with an appropriate detector. As much as practical, the detector is of an appropriate size to maintain the surface to detector distance of no greater than the calibrated distance +0.5-cm.

4.4.2 Soils

Soil areas require the following data to demonstrate they meet the appropriate release criteria.

Scans

Soil areas will require gamma scan measurements as part of the FSS process at appropriate coverage rates and speeds.

Volumetric Samples

Volumetric samples are required to demonstrate a soil area meets the appropriate release criteria. In lieu of volumetric samples, soil areas may receive direct measurements using in-situ gamma spectroscopy, as equipment and trained personnel are available. Volumetric sampling differs slightly depending on the situation for which the sample is desired. The required quantity of volumetric samples for an open land survey unit is a calculated value that is discussed in MARSSIM.

4.5 Data evaluation

Data evaluation is performed on FSS results for individual survey units to determine the weather the survey unit meets release criterion. Appropriate tests will be used for the statistical evaluation of survey data. Tests such as the Sign test and Wilcoxon Rank Sum (WRS) test will be implemented using unity rules, surrogate methodologies, or combinations of unity rules and surrogate methodologies, as described in the MARSSIM and NUREG-1505 chapters 11 and 12.

If the contaminant is not in the background or constitutes a small fraction of the DCGL, the Sign test will be used. If background is a significant fraction of the DCGL the Wilcoxon Rank Sum (WRS) test will be used. It is anticipated that the sign test will be the only statistical test applied to the collected data because of the small fraction of the DCGL that background radionuclides will contribute.

4.6 Final Status Survey Report

The Final Status Survey Report will be prepared to document and present the findings of the FSS, including all FSS data and data analysis. The report will summarize the decommissioning activities conducted at the NRL in support of license termination. This report will be provided to the NRC to support the request for termination of License R-52.

5.0 TECHNICAL SPECIFICATIONS

The University of Arizona NRL facility currently operates under technical specifications that are included as Tab E of NRC License R-52. These technical specifications are in place to insure the safe operation of the NRL facility. After the reactor ceases operation, most of the technical specifications will no longer be applicable. Additionally, other technical specifications that apply to non-operating conditions will be amended at the time of reactor shutdown and fuel removal. 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 through a license amendment.

6.0 PHYSICAL SECURITY PLAN

The regulations in Section 73.67(c)(1) of Part 73 require facilities to maintain a physical security plan when they possess special nuclear materials of moderate strategic significance or 10 kg or more of special nuclear material of low strategic significance. The nuclear fuel is to be shipped off-site prior to the start of decommissioning activities. Once the license has been amended for no possession of nuclear fuel, a physical security plan is no longer required.

It is recognized that the regulations in Sub Part I, Storage and Control of Licensed Material" of Part 20 are applicable to the remaining byproduct and special nuclear materials possessed by the UAR. All UARR licensed materials that are in storage will be secured from unauthorized access or removal; and licensed materials that are not in storage will be under the control and constant surveillance of authorized UARR personnel as required by 10 CFR 20.

7.0 EMERGENCY PLAN

The University has an Emergency Plan for responding to emergencies at the UARR facility. This Emergency Plan includes the response of the University police department, the local fire department and local emergency medical services. The plan covers events involving the potential or actual release of radioactivity and provides measures for facility evacuation, reentry and recovery. The plan also covers medical support for afflicted personnel.

The NRL D&D project will adopt the Emergency Plan as written. Minor changes to this Plan that do not change the effectiveness of the Plan do not require NRC approval, but will require approval by the Reactor Committee.

8.0 ENVIRONMENTAL REPORT

The Environmental Report (ER) was prepared and 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). The ER is intended 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

Following NRC review and approval of the Decommissioning Plan, the Plan will be incorporated as an amendment to License R-52. Minor changes to the Decommissioning Plan that do not change the original intent of the Plan and which do not involve an unreviewed safety question may be approved by the Reactor Committee.

If a significant change to the Decommissioning Plan is required, the Reactor Committee will apply the criteria identified in 10 CFR 50.59 (March 2001) as it applies to non-power reactors in decommissioning. If the Reactor Committee determines 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 DP will be documented and records of changes will be maintained until license termination. The FSSR will include a description of all changes to the DP.

10.0 REFERENCES

1. Enercon Services, Inc. May 2009. "University of Arizona Nuclear Reactor Laboratory Radiological Characterization Report"

2. WMG, Inc. May 2009. "University of Arizona Activation Analysis and Component Characterization" 08-125D-RE-122.

3. General Energy and Environmental Systems. August 1971. "TRIGA Mark I Reactor Mechanical Operation and Maintenance Manual" Gulf-EL-1042.

4. Gulf Energy and Environmental Systems. September 1971. "TRIGA Mark I Reactor Mechanical Operation and Maintenance Manual - Appendix" E-117-139.

5. International Organization for Standardization (ISO). 1988. "Evaluation of Surface Contamination-Part 2: Tritium Surface Contamination." ISO 7503-2. August 1.

6. U.S. Nuclear Regulatory Commission (NRC) (formerly the U.S. Atomic Energy Commission). 1974."Termination of Operating Licenses for Nuclear Reactors." Regulatory Guide 1.86. June.

7. NRC. 1981. "Control of Radioactively Contaminated Materials." NRC Office of Inspection and Enforcement (IE). IE Circular 81-07. May 14.

8. NRC. 1996. "Guidance for Preparing and Reviewing Applications for Licensing of Non-Power Reactors."NUREG-1537, Part 1. February.

9. NRC. 1998. "Supplemental Information of Implementation of the Final on Radiological Criteria for License Termination. Federal Register, Vol. 63, No. 222, page 64132. November 18.

10. NRC. 2002. "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)." NUREG-1575, Revision 1. August.

11. NRC. 2003. "Radiological Assessments for Clearance of Materials From Nuclear Facilities." NUREG-1640, Volume 1. June.

12. NRC. 2003b. "Environmental Review Guidance for Licensing Actions Associated with NMSS Programs." Office of Nuclear Material and Safety and Safeguards. NUREG-1748

13. NRC. 2009. "Standards for the protection against radiation" Title 10, Part 20, Code of Federal Regulations.