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November 10, 2016
Docket Number 50-59 / License No. R-23

2016-0057

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555
Ref: 10 CFR 50.90

SUBJECT: Final Status Survey Plan Revision 1 for the Unrestricted Release of the Zachry Engineering Center

Attn: Mr. Alexander Adams Jr., Chief,
Research and Test Reactors Branch
Office of Nuclear Reactor Regulation

Mr. Patrick M. Boyle, Project Manager,
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The purpose of this letter is to submit Revision 1 of the Final Status Survey (FSS) plan for the unrestricted release of the Zachry Engineering Center. Texas A&M University (TAMU) submitted the original FSS plan on October 18, 2016 (*NRC ADAMS Accession No. ML16299A537*). Revision 1 of the FSS is being submitting to provide additional radiological information obtained during recent (October 2016) removal of the AGN-201M reactor and associated operational work in the Zachry Engineering Center, which supports the rationale for the proposed FSS plan methodology. TAMU will next be submitting a license amendment request (LAR) that will request the U.S. Nuclear Regulatory Commission (NRC) approval of the unrestricted release of the Zachry Engineering Center. The LAR is in its final review stages and TAMU anticipates submitting the LAR to the NRC soon. This FSS plan will also be an enclosure to the upcoming LAR.

Should you have any questions regarding the information provided in this submittal, please contact me or Mr. Jerry Newhouse at (979) 845-7551 or via email at mcdeavitt@tamu.edu or newhouse@tamu.edu.



Oath of Affirmation

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Sincerely,

Sean M. McDeavitt, PhD
Director, TEES Nuclear Science Center

Submitted with Level 2 Delegate Authorization from Dr. Yassin Hassan in letter dated February 8, 2016 (ADAMS Accession No. ML16043A048)

Enclosure: FSS Survey Plan for the Unrestricted Release of the Zachry Engineering Center

CC: next page



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ENCLOSURE
TEXAS A&M UNIVERSITY
FACILITY LICENSE R-23, DOCKET NO. 50-59
AMENDED FACILITY OPERATING LICENSE
AGN-201M REACTOR
REVISION 1 FINAL STATUS SURVEY PLAN FOR THE UNRESTRICTED
RELEASE OF THE ZACHRY ENGINEERING CENTER

**SURVEY PLAN FOR THE
UNRESTRICTED RADIOLOGICAL RELEASE OF THE
AGN-201M RESEARCH REACTOR FACILITY
ZACHRY ENGINEERING CENTER
TEXAS A&M UNIVERSITY
COLLEGE STATION, TEXAS**

**Revision 1
November 10, 2016**

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ACRONYMS, ABBREVIATIONS, AND UNITS

C	carbon
cm	centimeter
cm ²	square centimeter
cpm	counts per minute
Cs	cesium
dpm	disintegrations per minute
Eu	europium
³ H	tritium
hr	hour
keV	kiloelectron volt
m	meter
m ²	square meter
MARSSIM	Multi-Agency Radiation Survey and Site Investigation
Manual MDC	minimum detectable concentration
MDCR	minimum detectable count rate
MeV	million electron volts
NRC	Nuclear Regulatory Commission
pCi	picocurie
pCi/g	picocurie per gram
PuBe	plutonium-beryllium (neutron source)
RSSI	Radiation Site Survey and Investigation
TAMU	Texas A&M University
TDSHS	Texas Department of State Health Services
U	uranium
μCi	microcurie

UNRESTRICTED RADIOLOGICAL RELEASE SURVEY PLAN
AGN-201M RESEARCH REACTOR FACILITY
ZACHRY ENGINEERING CENTER
TEXAS A&M UNIVERSITY, COLLEGE STATION, TX

1.0 INTRODUCTION

Texas A&M University (TAMU) is renovating the Zachry Engineering Center. This Center housed the AGN-201M reactor, licensed by the Nuclear Regulatory Commission (Facility License R-23). It also contained offices and laboratories in which radiological materials were used in support of reactor operations and other activities, as authorized under Texas Department of State Health Services (TDSHS) license L00448. Furnishings, materials and equipment were surveyed and removed from the State-licensed areas of the facility. Building surfaces in those non-reactor areas were surveyed and demonstrated to satisfy the TAMU criterion for demolition without need for radiological restrictions. These areas have been razed in preparation for renovation.

The reactor and associated components have been packaged and placed in secure offsite storage, awaiting reinstallation in a new facility (note that the Part 50 license is not being terminated). Remaining materials and equipment in the reactor facility have been surveyed, removed, and dispositioned in accordance with the TAMU criteria. Radiological surveys performed over the operating history of the AGN-201M reactor have identified no contamination over TAMU release limits in the reactor areas of Zachry Engineering Center. In addition, recent surveys conducted during reactor disassembly and in support of the removal, packaging and transport of the reactor fuel and SNM have not identified contamination over TAMU release limits on the reactor external surfaces or on reactor internal component surfaces (e.g., in the core tank, on control rod drive thimbles that pass through the unclad fuel disks, and on the lower core plate).

ReNuke Services, Inc., of Oak Ridge, TN, has been contracted by TAMU to remove and relocate the reactor, develop a survey plan, and conduct unrestricted release surveys of the building. The final results will be submitted as a supplement to a license amendment request for the unrestricted release of the Zachry Engineering Center.

Office furnishings, miscellaneous materials and non-reactor equipment have been surveyed in accordance with the TAMU Radiological Safety Program and removed from the facility. No contaminated items were identified. Screening surveys of the reactor facility surfaces have been performed, with no contamination detected and no need for decontamination identified. Based upon reactor power history and neutron surveys during power operation, activation of the building structure is considered very unlikely. Concrete samples from shield blocks around the reactor support skirt and from walls in the reactor room have been analyzed by an offsite laboratory for the presence of neutron activation products, and support this assessment; no activation products were detected. Based upon these surveys, the AGN-201M design

characteristics, and the facility historical uses, the areas have been classified as to contamination potential. Radiological surveys of the impacted areas will be conducted to demonstrate that the facility conditions satisfy requirements for unrestricted future use and thus enable building renovations to proceed without radiological safety constraints.

2.0 PURPOSE AND SCOPE

The purpose of the release surveys is to demonstrate that areas of the Texas A&M University Zachry Engineering Center, which houses the AGN-201M reactor facility, satisfy criteria of the Nuclear Regulatory Commission, Texas Department of State Health Services, and Texas A&M University Radiological Safety, Environmental Health and Safety for unrestricted release. By satisfying these criteria, the remaining structure can be demolished or reused without radiological restrictions.

Texas A&M University will comply with the requirements of 10 CFR 20.1402, radiological criteria for the unrestricted release of the Zachry Engineering Center. In accordance with this rule, the site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent (TEDE) to an average member of the public does not exceed 25 mrem (0.25 mSv) per year. No contamination has been detected on any surfaces or components during extensive surveys conducted in support of defueling and during earlier scoping surveys, and TAMU commits to using the default screening values for surface contamination as presented in Appendix H to NUREG -1757, Volume 2, Revision 1 as upper limits for the project. Site characteristics support the use of these values, as only superficial surface contamination is expected. There are no buried pipes or potentially contaminated structures, and no unusual radionuclides are anticipated. The screening values have been determined by the NRC to be ALARA; no further pathways evaluations are required (Appendix N to NUREG – 1757, Volume 2). TAMU’s self-imposed release criteria are more limiting (contamination is not to exceed twice background, using appropriate instrumentation), as explained below.

3.0 SITE DESCRIPTION

Figure 1 is a site map of the Texas A&M campus, indicating the location of the Zachry Engineering Center, on Bizzell Street near University Drive. This Center was home to Engineering Student Services and Academic Programs Office, as well as the Department of Nuclear Engineering and the Department of Electrical and Computer Engineering. The building is a large concrete structure and consists of a basement level, a ground level, and three additional floors. As depicted in Figure 3, the AGN-201M reactor was located in Room 61B on the ground floor, in the southwest portion of the building. It is a fully self-contained unit, with no external coolant or irradiation systems. The reactor core is a right cylinder, approximately 26 cm diameter by 24 cm high consisting of nine fuel discs and fueled control rods containing nominally 665 grams of U-235 at an enrichment of just less than 20%. The fuel is a mixture of UO₂ microspheres in a polyethylene matrix. The core and the control and safety rods are surrounded by a leak tight, 95 cm diameter by 148 cm high core-tank.

A 10 cm thick lead shield surrounds the core-tank and 20 cm thick graphite reflectors. A 198 cm diameter by 213 cm height water shield tank surrounds the reactor core assembly (Figure 2). The maximum authorized steady state operating power level is 5 watts, thermal. The reactor has not operated for several years. The design of the AGN-201M reactor precludes the possibility of groundwater or soil contamination, as there are no external coolant pumps, heat exchangers, or coolant makeup/cleanup systems, and no external irradiation loops. In addition, the basic design precludes the need for radioactive waste processing systems (e.g., no waste compaction, liquid waste treatment, or contaminated off gas treatment systems). Accordingly, the FSS does not address soil or groundwater sampling.

Room 60C was primarily used for office space and access control. Room 61A was used in support of reactor operations (e.g., safeguards laboratory work, experiment preparation). Room 61B contains the reactor control console and a small inner room where radioactive sources were stored. Access to the top of the reactor is through Room 135 on the 1st floor level, directly above the reactor room. Rooms 60C, 61A, 61B, and 135 (which also previously contained an ion-implant particle accelerator) constitute the primary site security boundaries for the reactor. These rooms occupy approximately 170 m² on each level. They have 1-m reinforced concrete walls; the accelerator room ceiling is also 1-m thick concrete and a steel plate liner. Figures 3 and 4 show the layouts of the reactor facility; bolded outlines indicate Primary Reactor Site boundaries.

A polyethylene tank was located in the Basement directly beneath the primary reactor facility and was connected to a floor drain in Room 61B to allow collection of water in the unlikely event of leakage from the reactor shield tank (radiologically uncontaminated chromated water). This capability was not used, and the tank remained empty until its survey and removal. The single PVC drain line did not contain detectable radioactivity when examined during scoping surveys. It was removed and all sections surveyed, along with the polyethylene tank. Rooms 135 and 61A were also equipped with sink drains previously connected to a sump in the State-licensed area of the building. No contamination was detected in the glass drain line or the in-line trap in the reactor areas, and no contamination was identified during sampling of the sump or the State-licensed laboratory drains. These drains were terminated and the sump released as part of the laboratory decommissioning.

The facility shares electric power and air supply with the remainder of Zachry Engineering Center building. During normal power operation, ventilation for the reactor area was provided by a ventilation fan in Room 135, which pulled air through a grated opening in the Room 61B ceiling. Portions of the ventilation system were surveyed in early 2016 during laboratory facility surveys, and found to meet the applicable release criteria.

Figure 1 – Map of Texas A&M Campus, indicating location of Zachry Engineering Center

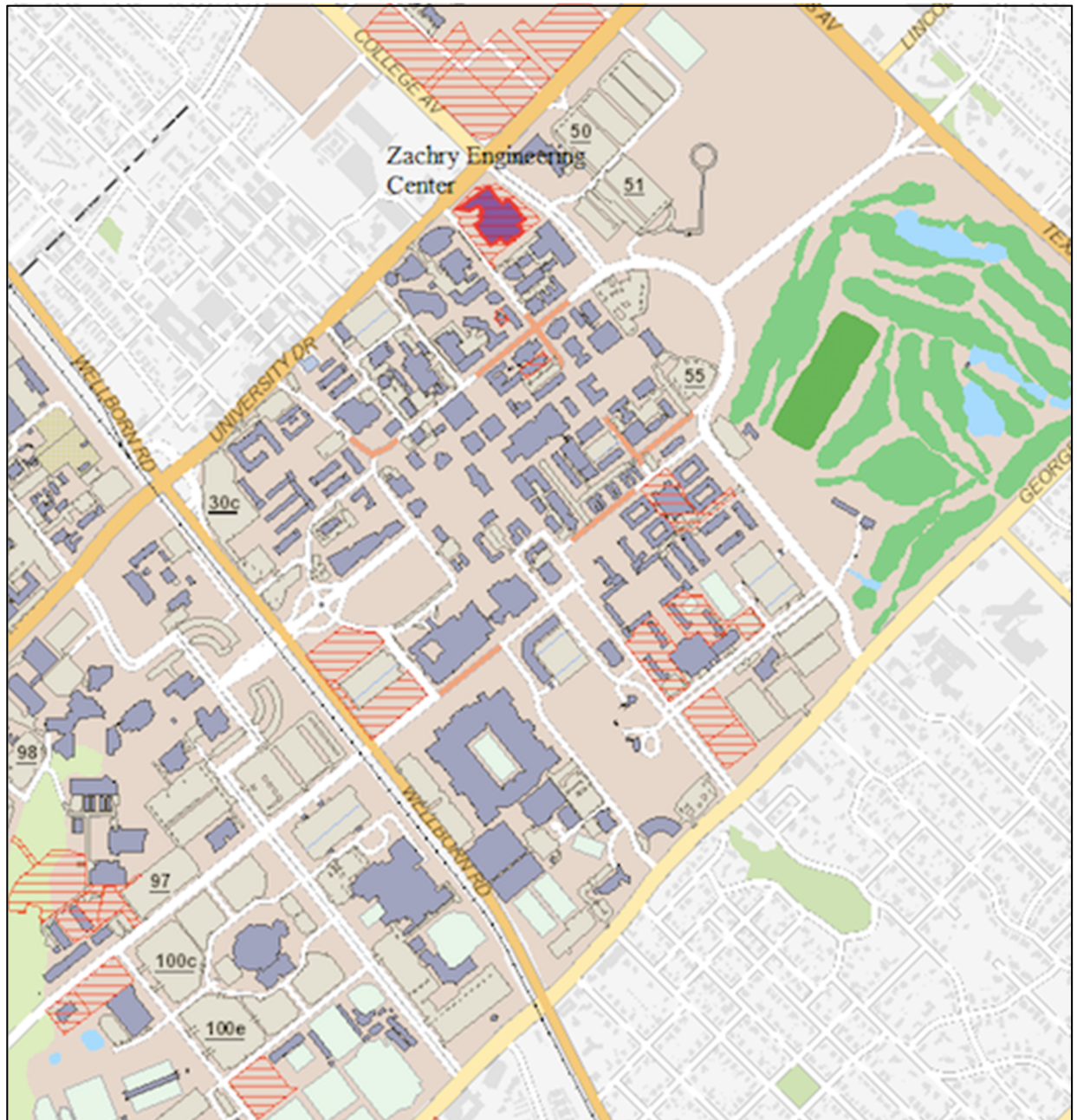


Figure 2 – Cut-away View of AGN-201M Reactor

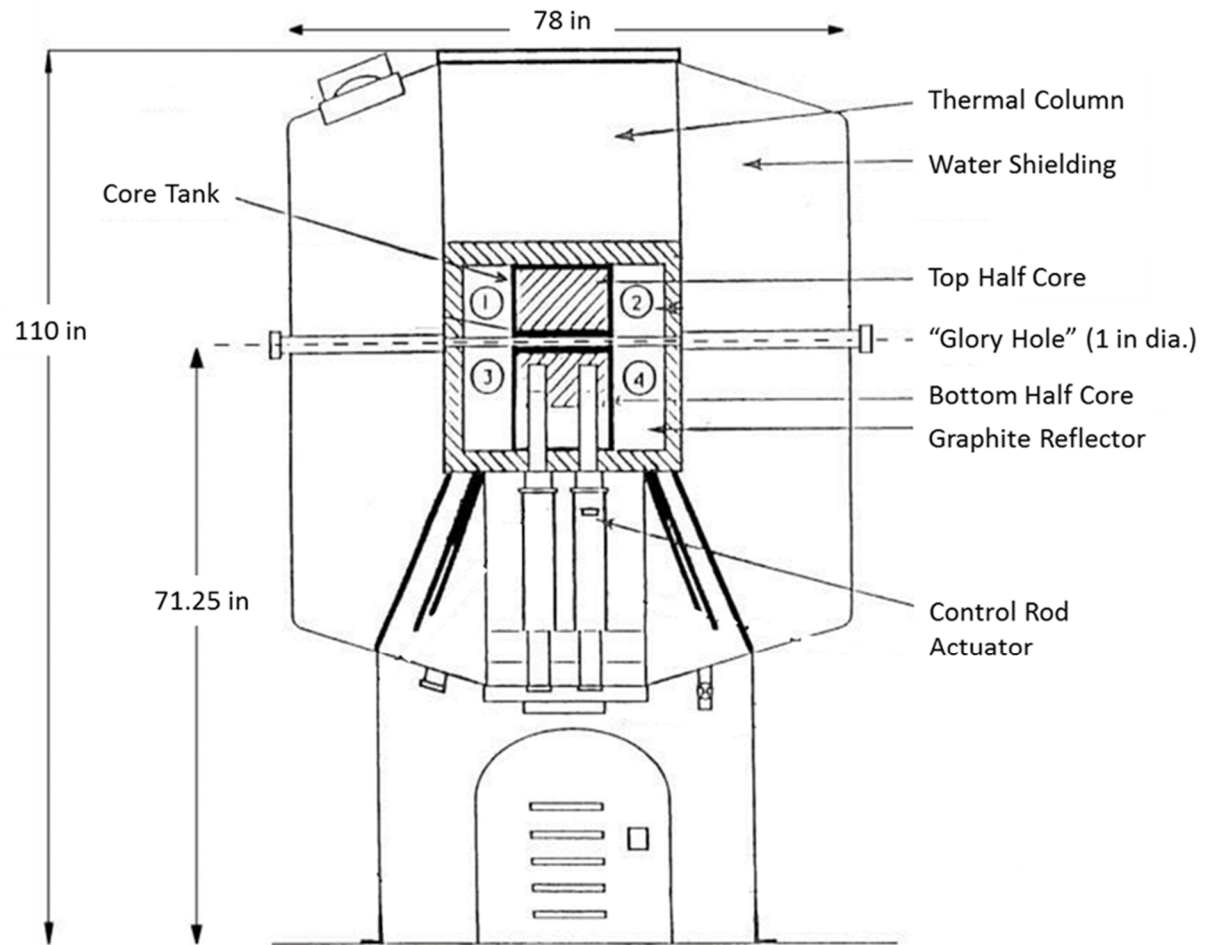


Figure 2a – AGN-201M Reactor without block shielding, as currently located in Zachry Engineering Center



Figure 3 - Reactor Facility Ground Floor; Bolded outline indicates Security Boundary

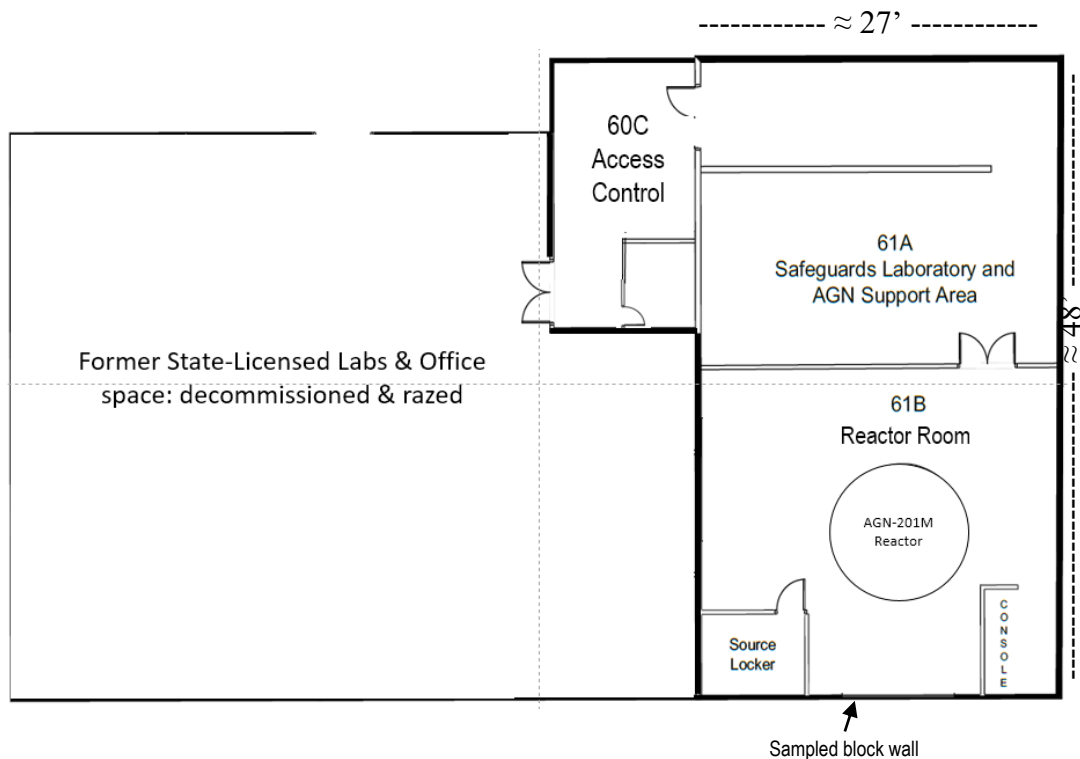
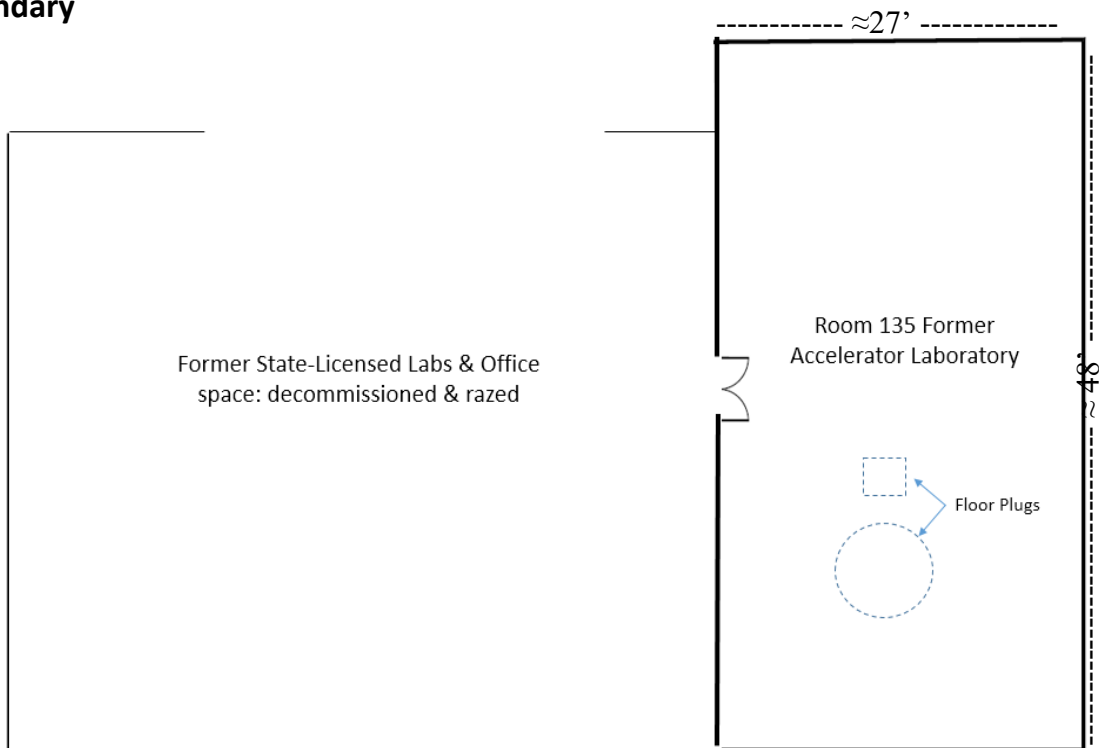


Figure 4 – Reactor Facility First Floor; Bolded outline indicates primary Reactor Site Boundary




4.0 RADIONUCLIDE CONTAMINANTS AND CRITERIA

AGN-201M reactor operations in the Zachry Engineering Center began in 1972 and concluded in 2014. During the school years of 1999/2000 through 2009/2010, the reactor was not operated. In other years, annual operating watt- hours ranged from 4.32 to 82.36. Since the 2009/2010 school year, the total operating time has been approximately 138 watt-hours. There has been no reactor operation since 2014. Records and anecdotal information from the previous Senior Reactor Operator have not revealed any reactor incidents or occurrences which may have resulted in contamination of surfaces external to the reactor shield tank. Results of surveys performed by the TAMU Radiological Safety staff did not identify any detectable removable contamination on reactor components or reactor room surfaces. Recent scoping surveys did not detect any fixed or removable contamination on surfaces in rooms 61A and 61B. Considering the low power level and limited operating time, low neutron fluence rate (1.5×10^8 n/cm²-sec, average at the 5 watts maximum licensed power), inherent shielding provided by the reactor components and containment tank, and the decay time since last operation, the likelihood of detectable activity in facility structural media is considered to be negligible.


Conservative, bounding calculations estimate ¹⁵²Eu (likely the predominant activation product in concrete) specific activity in the range of 10^{-3} pCi/g in concrete shield blocks that were located around the reactor support skirt. Sampling and analyses was conducted to validate the calculation-based conclusion that no activation products are present at detectable levels. Candidate radionuclides for concrete activation include ¹⁵²Eu, ¹⁵⁴Eu, ⁶⁰Co, ¹³⁴Cs, ³H, and ¹⁴C. Laboratory analyses (gamma spectrometry and liquid scintillation counting) of core samples are summarized below. Samples included cores from 4 innermost shield blocks previously around the reactor skirt (shielding subject to the highest neutron dose), 1 core from the North room wall, collected directly across from the glory hole, 3 cores from the South block wall in line with the glory hole, 2 cores from the nominal 3 ½" concrete reactor pad, 2 cores from the concrete floor directly under the reactor pad, and 1 core from the wall in the access hallway, an area with no significant neutron exposure. None of the samples were found to contain detectable activation products, with minimum detectable concentrations for the radionuclides of interest less than 10% of the NUREG 1757 soil screening values. Europium MDC's were also < 10% of the EPA values for residential soils. These screening values are directly applicable as a portion of the South wall was opened for removal of the reactor shield tank. Exposure rate surveys 1 meter over the reactor pad were not different than the nominal 5 microR/h ambient exposure rates throughout the facility. Sample results are presented in Table 1.

Table 1. Volumetric sample data


	134Cs	60C	152Eu	154Eu	3H	14C
<i>No activation products detected</i>	MDC, pCi/g	MDC, pCi/g	MDC, pCi/g	MDC, pCi/g	MDC, pCi/g	MDC, pCi/g
Wall: hallway (no neutron irradiation)	5.82E-02	7.23E-02	1.12E-01	1.73E-01	5.43E+00	1.41E+00
Wall: N side, opposite glory hole	8.97E-02	9.91E-02	1.90E-01	2.97E-01	5.44E+00	1.37E+00
S. Wall 1: IW-1 (wall removed)	4.77E-02	4.30E-02	1.09E-01	1.01E-01	7.63E+00	6.26E-01
S. Wall 2: IW-2 (wall removed)	4.52E-02	4.11E-02	1.40E-01	1.30E-01	7.85E+00	6.01E-02
S. Wall 3: IW-3 (wall removed)	7.01E-02	6.72E-02	1.29E-01	2.26E-01	8.08E+00	6.21E-01
Reactor shield block: E1	7.87E-02	7.69E-02	1.31E-01	2.17E-01	7.96E+00	5.98E-01
Reactor shield block: S1	8.06E-02	7.64E-02	1.70E-01	1.86E-01	8.07E+00	6.14E-01
Reactor shield block: N1	5.54E-02	7.87E-02	1.64E-01	1.99E-01	7.88E+00	6.07E-01
Reactor shield block: W1	6.01E-02	5.80E-02	1.30E-01	1.84E-01	8.06E+00	1.00E+00
Reactor pad concrete 1	5.89E-02	5.19E-02	1.26E-01	1.55E-01	6.85E+00	6.34E-01
Reactor pad concrete 2	5.59E-02	5.20E-02	1.14E-01	1.41E-01	6.30E+00	6.35E-01
Floor under reactor pad, 1	9.17E-02	6.72E-02	1.77E-01	2.07E-01	6.27E+00	6.51E-01
Floor under reactor pad, 2	6.87E-02	5.17E-02	1.69E-01	1.98E-01	6.58E+00	6.30E-01



South Wall (IW-1 to IW-3)



Shield Block Wall (4 Samples)



Reactor Pad Concrete (4 samples)

Coverings have not been applied over any known location of contamination. The location in the facility considered most likely to have been impacted by reactor operations is the concrete floor directly beneath the reactor shield tank. Potential activation radionuclides include the same radionuclides in the above table. The core assembly contains enriched uranium fuel and (likely) very small quantities of longer-half-life fission products including ^{137}Cs , ^{90}Sr , ^{144}Ce , and ^{95}Zr and activation products such as ^{60}Co in components; however, there is no history of contamination by these radionuclides on surfaces external to the reactor.

The NRC reactor license includes a $^{239}\text{PuBe}$ special-form neutron source containing up to 16 grams of ^{239}Pu for use in reactor operation. This source was leak tested (no contamination was detected), removed from the AGN-201M reactor, and transported to offsite storage

Section 17.1.4 of NUREG-1537 establishes the following criteria to release non-power reactor facilities for unrestricted use

1. a) no more than 5 microrem per hour above background at 1 meter from the surface measured for indoor gamma radiation fields from concrete, components, and structures, or

b) no more than 10 millirem per year for gamma emitters above background absorbed dose to any person, considering reasonable occupancy and proximity (NRC letters dated March 17, 1981 and April 21, 1982).
2. residual surface contamination consistent with Regulatory Guide 1.86.

Regulatory Guide 1.86 was withdrawn by NRC, effective August 12, 2016, although similar numerical guidance remains in Regulatory Guides 8.21, "Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants", and 8.30, "Health Physics Surveys in Uranium Recovery Facilities". The table of surface contamination values has been retained (see Table 2) for the project as these values are also in Texas Regulation 25 TAC §289.202(ggg)(6), Acceptable surface contamination levels (Ref 2), and are applicable to State-licensed activities at TAMU.

Table 2. Acceptable Surface Contamination Levels based on Detectability

Nuclide ^a	Total	Removable
U-nat, U-235, U-238, and associated decay products	5000 dpm/100 cm ²	1000 dpm/100 cm ²
Transuranics, Ra-226, Ra-228, Th-230, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm ²	20 dpm/100 cm ²
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm ²	200 dpm/100 cm ²
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above	5000 dpm/100 cm ²	1000 dpm/100 cm ²

^a. Where surface contamination by both alpha- and beta-gamma emitting radionuclides exist, the limits established for alpha- and beta-gamma-emitting radionuclides apply independently.

The TAMU radiation safety program has a policy of "no detectable activity" for unrestricted use and release. "No detectable activity" is interpreted by TAMU as not exceeding twice the background level (Ref 1).

Due to other Zachry Engineering Center renovation work starting prior to the reactor relocation project, suitable concrete surfaces outside the reactor complex were not available for reference background measurements. Background measurements were performed in a Class 3 area (Room 60C) on a section of poured concrete floor unlikely to have been impacted by reactor operations. This area, as with the other floor areas, was covered with floor tile since the start of reactor operations. The tile has since been removed from all areas. The poured concrete background levels for the 126 cm² gas-flow proportional detectors are effectively the observed ambient background levels of 3 alpha cpm and 250 beta cpm. As expected, the larger (nominally 580 cm²) gas-flow proportional floor monitor exhibits higher ambient background levels of approximately 9 alpha cpm and

300 beta cpm. Initial survey work supports the conservative use of ambient background levels for scanning surveys of the floors and the remainder of the facility surfaces (primarily poured concrete walls, steel and glass). The ambient background values will also be conservatively applied to surveys of two higher density block shield walls in Room 61B. Laboratory analysis demonstrates a higher concentration of the ^{232}Th -series radionuclides relative to other block and concrete used in facility construction, and the impact on surveys is an increased background count rate of 100 cpm on these blocks. This material-specific background will not, however, be applied. The ambient background will also be used for scanning surveys of these walls. To summarize, no materials-specific background count rates will be subtracted from scanning surveys of facility surfaces. Note that 2-step surveys will be used for static measurements, as described in section 6.3, Integrated Survey Strategy.

The nominal ambient background values are used to calculate the sensitivity of the scanning surveys to ensure they are adequate relative to the surface contamination Screening Values presented NUREG 1757, Appendix H. Because no residual radioactivity has been identified during extensive surveys of the reactor, associated components, and the facility, no specific radionuclides of interest have been identified. The NUREG 1757, Appendix H surface contamination screening value for ^{60}Co (7,100 dpm/100 cm²) has been conservatively chosen for evaluation of potential building contamination. Table 3 presents a summary of the minimum detectable surface contamination, the NRC screening value, and the surface contamination levels corresponding to twice the background levels (the TAMU constraint).

During scanning surveys, net count rates exceeding 450 cpm beta when using the 126 cm² detectors, or 800 cpm beta for the 580 cm² detector, will be indicative of contamination exceeding the screening value for ^{60}Co .

The assumption that the removable activity fraction does not exceed 10% will be evaluated if contaminated areas are identified. Table 3 clearly demonstrates that meeting the TAMU Radiological Safety criteria for release will also satisfy the NRC requirements. As noted, no contamination has been detected and no specific radionuclides have been identified during extensive scoping surveys; ^{60}Co has been conservatively selected as a possible radionuclide of concern. Gross alpha and beta-gamma surveys limits will be applied (i.e., not to exceed twice background, alpha and beta-gamma activity evaluated independently). Also note that net count rates exceeding 3 cpm alpha or 250 cpm beta when using the 126 cm² detectors, or 9 cpm alpha or 300 cpm beta for the 580 cm² detector, will be indicative of contamination exceeding the TAMU criteria (twice background) and will require further investigation.

Table 3. Summary Data

Detector/Application	MDA (dpm/100 cm ²)	NRC Co-60 Default Screening Value (dpm/100 cm ²)	TAMU twice background equivalent (net dpm/100 cm ²)
126 cm ² gas proportional/scan	Alpha - N/A * 1500 beta	Alpha - N/A* 7100	N/A 3000
126 cm ² gas proportional/static	63 alpha 470 beta	Alpha - N/A* 7100	26 alpha (< MDC) 2200 beta
580 cm ² gas proportional/scan	Alpha - N/A* 2280 beta (for a 100 cm ² spot)	Alpha - N/A* 7100	A > 2 x background reading will be investigated with a 126 cm ² detector
Laboratory counter/removable activity	19 alpha 90 beta	Alpha - N/A* 710	4 alpha 500 beta
*No potential alpha-emitting contaminants have been identified			

Measured background exposure rates within the Zachry Engineering Center are 5 to 8 microR/hr. The TAMU “less than twice background” criterion will be met by confirming that dose rates from background plus residual licensed material are no more than 10 microR/h, measured at 1 meter from building surfaces. This is consistent with the previously noted guidance from NUREG 1537.

5.0 IMPACTED AREAS AND SURVEY UNITS

The Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (Ref 3) defines impacted areas as those with a possibility of residual radioactivity in excess of background levels. Radiological surveys of impacted areas are required to demonstrate that established criteria have been satisfied. Non-impacted areas are those with no reasonable expectation of residual contamination; no surveys of non-impacted areas are required. Impacted areas are classified as to contamination potential as follows:

- 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 radiological surveys) expected to be in excess of established unrestricted release criteria.
- Class 2: Areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed established criteria.
- Class 3: Areas that are potentially impacted but are not expected to contain any residual radioactivity, or are expected to contain levels of residual activity at a small fraction of the established criteria, based on site operating history and previous radiological surveys.

The rigor of a release survey is based on these contamination potential classifications. Structure survey units are established with size limitations presented in Table 4.

Table 4. Survey Unit Area by Classification

Classification	Maximum Area(m²)
Class 1	100 (floor surface)
Class 2	100 to 1000
Class 3	No limit

Table 5 contains a preliminary list of AGN-201M reactor facility impacted areas and survey units. This list is based on use history and previous monitoring records. Screening surveys, conducted during removal of furnishings, materials, and equipment from the facility support the recommended classifications.

Table 5. Impacted Areas and Survey Units

Class	Level	Room(s)	Surfaces	Number of Survey Units
Class 1	Ground	61A	Floor and lower walls	1
	Ground	61B	Floor and lower walls	1
	First	135	Floor and lower walls	2
Class 2	Ground	61A and 61B	Upper walls and ceiling	1
Class 2	First	135	Upper walls and ceiling	1
Class 3	Ground	60C	All	1

6.0 SURVEY APPROACH

6.1 General

This survey plan was prepared in accordance with guidelines and recommendations, presented in the Multi-Agency Radiation Survey and Site Investigation Manual. The process described in this reference emphasizes and incorporates the use of Data Quality Objectives and Data Quality Assessment, along with a quality assurance/quality control program. A quality assurance program for survey activities will be implemented. The graded approach is followed to assure that survey efforts are maximized in those areas having the greatest potential for residual contamination or the highest potential for adverse impacts of residual contamination.

Trained and qualified radiological technicians will conduct field measurements, following standard procedures and using calibrated instruments, sensitive to the potential contaminants. Professional health physics personnel will assess and evaluate the survey data and prepare a report of the findings.

6.2 Site Preparation

Furnishings, materials and equipment have been removed from the facility in accordance with TAMU Radiation Safety Program procedures. Following removal and transfer of the reactor and associated components, drains, ducts, diffusers, grates, cable trays, etc., were accessed and surveyed. Nominal 100 cm² dual phosphor detectors (Ludlum Instruments Model 43-93) have been used with dual channel scaler/ratemeters (Ludlum Instruments Model 2360) for health physics surveys conducted in support of defueling and cleanup work. Note that background and efficiencies for these scintillation detectors are identical to the gas flow proportional detectors selected for final status surveys. Use of the model 43-93 scintillation detectors was augmented with thin-window “pancake” Geiger-Mueller detectors with scaler/ratemeters (Ludlum Instruments Model 43-9 detectors with Model 3 scaler/ratemeters) to access smaller diameter penetrations (e.g., used for electric cables, water supply lines, natural gas lines, etc). To date, no building surfaces have been found to contain detectable residual activity, with MDC’s well below the NRC and TAMU release criteria as presented in Table 2 and no remediation has been required. Building surfaces have been appropriately gridded to provide a means for referencing survey locations. Measurements will be identified by grid coordinate or, if not practical, by referencing to building features or by photograph. Surfaces where contamination was deemed likely (Class 1) or possible (Class 2) have been gridded at 1-meter intervals, as is practical for the conditions. Grid origins are in the southwest corner of the room. If, during the survey, contamination above limits is identified in Class 2 or Class 3 areas, the rigor of the survey unit will be increased to that of Class 1 areas.

6.3 Integrated Survey Strategy

Radiological surveys will consist of:

- surface scans for elevated levels of gross alpha, beta, and gamma radiation levels,
- static measurements of gross alpha and gross beta activity,
- smears for removable gross alpha and gross beta activity, and
- sampling for laboratory analysis of specific radionuclide contaminants, if net activity is found

Based upon facility history and surveys performed in support of reactor disassembly and removal of the shield tank, the rigor of surveys will follow a graded approach based on the likelihood of contamination. Table 6 indicates the survey rigor for various contamination classifications.

Table 6. Survey Rigor for Each Survey Unit

Contamination Class	Alpha, Beta, and Gamma Scan	Static Alpha and Beta	Removable Alpha and Beta
1	100% - all structure surfaces	Systematic static measurement at a minimum of 18 locations and at additional locations of highest potential contamination, based on professional judgment and scan results	At each static measurement location
2	50% - floor and lower walls; 10% upper walls and ceiling surfaces	Systematic static measurement at a minimum of 18 locations and at additional locations of highest potential contamination, based on professional judgment and scan results	At each static measurement location
3	10 % - floor and 1 m ² around each static measurement location on lower walls	One floor measurement and 1 lower wall measurement per 10 m ² of floor area in each room, and measurements and at additional locations of potential contamination, based on professional judgment and scan results (minimum of 18 data points per survey unit).	At each static measurement location

Because the acceptable release criterion of twice background is low, Scenario B, as recommended by NUREG-1505 (Ref 4), is the basis for the survey design. The Null Hypothesis for that Scenario is:

“The survey unit meets the release criterion.”

The objective of the release survey is to accept this Null Hypotheses, by demonstrating at a Type I (α) decision error level of 0.05 and a Type II (β) decision error level of 0.025 that residual contamination is less than twice background. There are multiple building surface types (concrete, metal, wood, glass, etc.) in most survey units and, background levels will likely vary, by instrument, material, time of day, and location within the facility. To facilitate adjusting measurements for appropriate localized background contributions, a paired measurement approach will be used. To perform paired measurements, a measurement is first performed by placing a piece of nominal ¼-inch plastic or metal shield material, between the surface and the Ludlum 43-68 detector face. The static measurement is repeated without the intervening shield material, and the difference between the second and first measurement indicates the net contamination level.

Each measurement will be individually evaluated and no individual measurement may indicate detectable activity in excess of the project limits (e.g., all net measurements must be less than the 3 alpha and 250 beta cpm nominal count rates as measured with the 43-68 detectors the more restrictive TAMU limits). No statistical test will be required to demonstrate compliance with release criteria.

To establish the number of measurements needed to demonstrate that residual contamination criteria have been satisfied, a parameter known as the “relative shift”, which effectively describes the distribution of final sample data, is calculated, as follows:

$$(1) \Delta/\sigma = (DCGL-LBGR)/\sigma$$

where:

Δ/σ = relative shift

DCGL Criteria= cleanup criteria

LBGR = lower bound of the gray region and is defined in the DQOs as 50 percent of the DCGL. Where final sample data are not yet available, MARSSIM guidance (Section 5.5.2.2) assigns a value of one-half of the DCGL for the LBGR.

σ = standard deviation of the sample concentrations in the survey unit. Where final sample data are not yet available, MARSSIM guidance (Section 5.5.2.2) recommends a value of 30 percent of the DCGL.

Using the equation for relative shift and MARSSIM guidance for situations where final sample data are not yet available, the relative shift for design purposes is $(1 - 0.5)/0.3$ for a value of 1.67. Based on the relative shift of 1.67 and Type I and Type II decision errors of 0.05 and 0.025, respectively, the number of required data points from each survey unit to perform the evaluation, as obtained from MARSSIM guidance (Table 5.5) is 18.

For static measurement locations on Class 1 and Class 2 room surfaces, a random start point will be identified on the floor and additional measurement locations will be systematically selected on a triangular spacing from that start point. Spacing distance, L, is determined by

$$L = [(Survey\ Unit\ Area)/0.866 \times number\ of\ data\ points]^{0.5}$$

Internal surfaces of ductwork and piping will be accessed, scanned, and static measurements performed at the entrance and discharge and additional points at a frequency of 1 measurement per 4 m² of internal surface area.

Static measurement locations on Class 3 room surfaces will be at locations of highest contamination potential, as selected by professional judgment.

6.4 FSS Survey Instrumentation

Table 7 is a list of radiological survey instrumentation that will be used to implement the AGN-201M reactor facility surveys. These instruments will be maintained, calibrated, and operated in accordance with written procedures. For application to unrestricted release, instrument response (efficiency) is based on NIST-traceable sources of Tc-99 (beta $E_{MAX} = 292$ keV) and Th-230 (alpha $E = 4.68$ MeV). The energies of these radionuclides are representative of the dominant potential contaminants. Note that the 126 cm² 43-68 gas flow proportional detectors have detector efficiencies and ambient background count rates that are identical to the 100 cm² 43-93 dual phosphor scintillation detectors used for health physics surveys during facility preparations for release; survey data are directly comparable.

Table 7. Instrumentation for Release Surveys

Detector	Display	Application
Ludlum 43-37	Ludlum 2360	Alpha /beta scans
Ludlum 43-68	Ludlum 2360	Alpha/beta scans
Ludlum 43-68	Ludlum 2360	Alpha/beta static measurements
Ludlum 43-10	Ludlum 2929	Removable alpha/beta measurements (scaler)
Ludlum 19	N/A	Gamma scans/direct gamma measurements

For field measurement applications, calibration represents 2π response. Effects of surface conditions on measurements are integrated into the overall instrument response through use of a “source efficiency” factor, in accordance with the guidance in ISO-7503-1 (Ref 7) and NUREG/CR-1507 (Ref 8). Default surface efficiencies of 0.25 for alpha emitters and 0.25 for beta emitters will be used.

Detection sensitivities are estimated using the guidance in MARSSIM and NUREG/CR-1507. Instrumentation and survey techniques are chosen with the objective of achieving detection sensitivities of $\leq 50\%$ of the criteria for structure surfaces, for both scanning and direct measurement. These detection sensitivities assure identification of areas potentially exceeding the established project criteria. Minimum detectable activity levels (refer to Appendix A) for this survey satisfy the Table 1 values.

6.5 Surface Scans

Scans of surfaces will be performed to identify locations of potential residual surface contamination and induced activity. Gas proportional detectors will be used for alpha and beta scans. A Ludlum Model 43-37 gas proportional detector (580 cm^2) will be used with a Ludlum Model 2360 scaler/ratemeter to scan the floor surfaces. Surfaces not accessible with this large detector will be scanned with the smaller Ludlum Model 43-68 gas proportional detectors (126 cm^2) used with Ludlum Model 2360 scaler/ratemeters. Alpha/beta scanning will be performed by maintaining the detector within $\frac{1}{4}$ inch of the surface and passing the detector over the surface at a rate of approximately $\frac{1}{2}$ detector width per second, while monitoring the audible output of the scaler/ratemeter for immediate identification of increases in count rate. When 2 alpha counts are detected within approximately 2 seconds, the detector movement will be halted at the location for approximately 10 seconds to detect a possible elevated count rate. This is consistent with Appendix J guidance in the MARSSIM document (NUREG 1575).

A Ludlum Model 19 gamma scintillation detector will be used for gamma scans. General area gamma monitoring will be performed with the detector approximately 1 m above the floor.

6.6 Static Surface Activity Measurements

Static measurement of alpha and beta surface activity will be performed using Ludlum Model 43-68 gas proportional detectors with Ludlum Model 2360 scaler/ratemeters. Measurements will be conducted by holding the detector in position within ¼ inch of the surface and integrating the count over a 2-minute period. Two measurements, 1 shielded and 1 unshielded, will be performed at each static measurement location, with the net count rate being the difference between the two measurements, for alpha and beta detection.

6.7 Removable Contamination Measurements

A smear for removable activity will be performed at each static surface activity measurement location. A 100 cm² surface area will be wiped with a nominal 2-inch diameter cloth smear, using moderate pressure.

6.8 Samples and Analyses

Smears will be analyzed onsite for gross alpha and gross beta activity using a Ludlum Model 2929 scaler with a Model 43-10-1 dual scintillation detector (or equivalent instrumentation).

6.9 Quality Assurance/Quality Control

Measurements will be performed in accordance with the survey plan by qualified personnel following written instrument operating procedures. Instrument calibration practices meet ANSI standards and daily background and source response checks of instruments will be performed daily. For quality control purposes, replicate static and removable activity measurements were obtained at 2 locations in each survey unit.

7.0 DATA EVALUATION

Surface contamination measurement data will be adjusted for background contributions and converted to units of net counts per minute. Data will be assessed to verify that the type, quantity, and quality are consistent with the survey plan and design assumptions. Individual data values will be compared with the count rate limit derived from NUREG 1757 surface contamination screening values for ⁶⁰Co and the TAMU criteria of twice background. Residual contamination limits established for the project (see Section 4) are presented in Table 3, including a comparison of the TAMU limits and the NRC screening values.

Evaluation of volumetric sample data is discussed in Section 4, Radionuclide Contaminants and Criteria

8.0 ISOLATION AND CONTROL

Following completion of the release survey, the facility will be isolated and access controlled until NRC approval for unrestricted release is received. It is recognized that the NRC may choose to conduct independent surveys to confirm the findings of this survey. These areas will not be available for general access or work until NRC approval for unrestricted release is obtained.

9.0 REPORT

A draft report describing the survey procedures and findings will be prepared. This report will stand alone and provide a complete record, documenting the facility's radiological status satisfies established project criteria and that the facility is therefore ready for unrestricted release. Appendix B is a sample of the report content. The report will include all sample data supporting this determination. Comments on the draft report will be resolved and a final report prepared and submitted to the NRC for review and approval.

10.0 REFERENCES

1. Radiological Safety Program Manual, Radiological Safety Environmental Health and Safety Department, Texas A&M University, July 2004.
2. Texas Regulations for acceptable contamination levels for unrestricted use, 25 TAC§289.202(ggg)(6).
3. Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), NUREG-1575 (Rev. 1), U.S. Nuclear Regulatory Commission, 2000.
4. A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys, NUREG-1505 (Rev 1) U.S. Nuclear Regulatory Commission, 1998.
5. Evaluation of Surface Contamination – Part 1: Beta Emitters and Alpha Emitters, ISO- 7503-1, International Organization for Standardization, 1988.
6. Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions, NUREG/CR-1507, U.S. Nuclear Regulatory Commission, 1997.
7. Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, NUREG-1757, Vol. 2, Rev. 1, U.S. Nuclear Regulatory Commission, 2006.

Appendix A

Measurement/Detection Sensitivities of Survey Techniques

The methods for calculating survey detection sensitivities are presented in MARSSIM (Ref 1) and NUREG-1507(Ref 2). Detector parameters used in these calculation are background count rate, efficiency (instrument response and surface correction factors), and detector area. The following table presents typical values of these parameters for detectors used for surveys of concrete structure surfaces of the AGN reactor facility. Background levels for concrete are the highest for surface media remaining in this facility, and therefore direct measurements on other media will be more sensitive than those presented here for concrete.

Detector/ Instrument	Probe Area (cm ²)	Background (cpm)		Detector efficiency		Surface correction	
		alpha	beta	alpha	beta	alpha	beta
43-37	580	9	300	0.50	0.46	0.25	0.25
43-68	126	3	250	0.36	0.36	0.25	0.25
2929	N/A	1	25	0.25	0.20	N/A	N/A

Alpha Scans

Surface scans for alpha activity are conducted using Ludlum Model 43-37 and Model 43-68 gas proportional detectors, coupled with Ludlum Model 2360 scaler/ratemeters. MARSSIM recommends the use of Poisson summation statistics to estimate the probability of detecting a small number of counts that may indicate the possible presence of alpha contamination during a relatively short observation period. The equation for estimating the probability of detecting 1 or more counts is:

$$P(n \geq 1) = 1 - e^{-[(GE + B)t]/60}$$

where:

$P(n \geq 1)$ = Probability of getting 1 or more counts during the time interval

\underline{G} = Source activity (disintegrations per minute, dpm)

B = Background count rate (counts per minute, cpm)

E = Detector efficiency (counts/disintegration)

t = Dwell time over source (sec)

The probability of detecting 2 or more counts is given by:

$$P(n \geq 2) = 1 - (e^{-[(GE + B)t]/60} - ((GE+B)(t))/60) \cdot e^{-[(GE + B)t]/60}$$

Using these parameters, detection probability calculations for a contamination level of 100

dpm/100 cm² were performed for a scan rate of ½ detector width per second (i.e., dwell times of 2 seconds). The probabilities of detecting a single alpha count during a 2-second dwell time are approximately 33% for the 43-68 detector and 52% for the 43-37 detector. Because of the higher background count rate associated with the 43-37 floor monitor detector, MARSSIM (Appendix I) recommends using 2 counts as a screening value when scanning for alpha contamination. The probability of detecting 2 counts with the larger detector increases to approximately 82%. Whenever a count is detected, the detector is paused over the surface for 10 seconds to determine whether there is actually elevated alpha activity present, in which case, a static measurement is then performed. A 10 second pause results in a 90% or greater probability of identifying the presence of alpha activity exceeding 100 dpm/100 cm². Although the calculated scan detection probabilities may appear relatively low, it should be noted that historic records and characterization surveys have not identified any potential for alpha contamination in this facility.

Alpha Activity Static Measurements

Static measurements of alpha surface activity are performed using 43-68 detectors, with the same background and response characteristics as indicated above for alpha scanning. A static measurement is performed by placing the detector on the surface and allowed to integrate the count for a period of 2 minutes. The minimum detectable alpha contamination level (MDC) is calculated as follows:

$$\text{MDC} = [3 + 4.65 (\text{BKGD})^{1/2}] / (\text{efficiency factors})(\text{detector area}/100)(\text{count time})$$

The resulting value is approximately 63 dpm/100 cm².

Beta Scans

Surface scans for beta activity are conducted using Ludlum Model 43-37 and Model 43-68 gas proportional detectors, coupled with Ludlum Model 2360 scaler/ratemeters. The detector is passed over the surface at a rate of 1/2 detector width/sec, while maintaining the distance from the detector to the surface at approximately 0.5 cm. The audible signal from the instrument is monitored by the surveyor. Detectable changes in the count rate are noted, and the immediate area resurveyed at a reduced speed to confirm the change in audible signal and, if applicable, to identify the boundary of the impacted area. The minimum detectable count rate (MDCR) is a function of the background count rate (BKGD) in counts per minute (cpm) and the time (i) in seconds that the detector is within close proximity to the source of radiation. Equation 6-6 of NUREG-1507 provides the following relationship:

$$\text{MDCR} = d' [\text{BKGD} * i / 60]^{1/2} * 60 / i$$

A high probability (95%) of true detection is the objective, and the survey is willing to accept a high probability of false-positive detections (60%) with resulting investigations. The value of d' is selected from Table 6.1 in NUREG-1507 to be 1.38.

To account for less than ideal survey performance, a surveyor efficiency factor (p) of $(0.5)^{1/2}$ was also incorporated into the final calculation of beta scan sensitivity as follows:

$$\frac{\text{MDCR}}{(0.5)^{1/2} * (\text{cpm/dpm})(\text{probe area}/100)} \quad (\text{B2})$$

The resulting values are approximately 1500 dpm/100 cm² for the 43-68 detector and an average of 390 dpm/100 cm² for the 43-37 detector. If only a single 100 cm² area is present, the scan sensitivity for the 43-37 detector would be approximately 2280 dpm.

Beta Activity Static Measurements

Static measurements of beta surface activity are performed using 43-68 detectors. A static measurement is performed by placing the detector on the surface and allowing it to integrate the count for a period of 2 minutes. The minimum detectable beta contamination level (MDC) is calculated as follows:

$$\text{MDC} = [3 + 4.65 (\text{BKGD})^{1/2}] / (\text{efficiency factors})(\text{detector area}/100)(\text{count time})$$

The resulting value is approximately 470 dpm/100 cm².

Removable Alpha and Beta Activity Measurements

Smears for removable activity are counted for 2 minutes in a Ludlum Model 2929 alpha/beta counter. The backgrounds are 1 alpha cpm and 25 beta cpm; 4π detection efficiencies are 0.25 alpha and 0.20 beta. Using the same equation (without probe area correction) as above for direct measurements yields removable activity MDCs of approximately 19 alpha dpm/100 cm² and 90 beta dpm/100 cm².

References

1. Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), NUREG-1575 (Rev. 1), US Nuclear Regulatory Commission, 2000.
2. Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions, NUREG/CR-1507, US Nuclear Regulatory Commission, 1997.

APPENDIX B

Sample Outline of Release Survey Report

- 1.0 Executive Summary
- 2.0 Introduction
- 3.0 Purpose and Scope
- 4.0 Site Description
- 5.0 Radionuclide Contaminants and Criteria
- 6.0 Survey Approach
- 7.0 Survey Results Summary
- 8.0 Conclusion
- 9.0 References

Attachments

Field data (electronic)