TEXAS A&M ENGINEERING EXPERIMENT STATION
NUCLEAR SCIENCE CENTER

TEXAS A&M UNIVERSITY

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October 18, 2016 2016-0054
Docket Number 50-59 / License No. R-23

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

SUBJECT: Final Status Survey Plan 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,
Research and Test Reactors Branch
Office of Nuclear Reactor Regulation

The purpose of this letter is to submit the Final Status Survey (FSS) plan for the unrestricted release of the Zachry Engineering Center. Texas A&M University (TAMU) is submitting the FSS plan now, in order to expedite the U. S. Nuclear Regulatory Commission (NRC) review of the upcoming license amendment request (LAR) that will request the NRC approval of the unrestricted release of the Zachry Engineering Center. This LAR is in its final review stages and we anticipate submitting it to the NRC soon. The attached FSS plan will also be included as an enclosure to the 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.

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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

FINAL STATUS SURVEY PLAN FOR THE UNRESTRICTED RELEASE OF

THE ZACHRY ENGINEERING CENTER

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TEXAS A&M ENGINEERING EXPERIMENT STATION
NUCLEAR SCIENCE CENTER

TEXAS A&M UNIVERSITY

Texas A&M University
Texas A&M Engineering Experiment Station

AGN-201M (Serial Number 106)

Facility License R-23, Facility Docket Number 50-59

SURVEY PLAN FOR THE UNRESTRICTED RADIOLOGICAL RELEASE OF THE AGN-201M RESEARCH REACTOR FACILITY ZACHRY ENGINEERING CENTER

TAMU/TEES AGN-08, Rev. 0
September 26, 2016

Revision History:
Revision 0 September 26, 2016

APPROVALS

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9/26/2016

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Sep. 26, 2016

Date

Texas A&M Radiological Safety Committee Chair

09/27/2016

Date

Texas A&M Reactor Safety Board

Oct 12/16

Date

Contents

ACRONYMS AND ABBREVIATIONS

C	carbon
cm	centimeter
cm^2	square centimeter
cpm	counts per minute
Cs	cesium
dpm	disintegrations per minute
Eu	europium
^3H	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, TEXAS

1.0 INTRODUCTION

Texas A&M University (TAMU) is renovating the Zachry Engineering Center. This Center houses 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 University's criterion for demolition without need for radiological restrictions. These areas have been razed in preparation for renovation.

The reactor and associated components will be 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 will be surveyed, removed, and dispositioned in accordance with the University's criteria.

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

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.

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 is 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 cooling or irradiation loops. 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 UO_2 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 x 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.

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 m^2 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, located in the Basement directly beneath the primary reactor facility, is

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 has not been used, and the tank is empty. The single PVC drain line did not contain detectable radioactivity when examined during scoping surveys. It will be removed and all sections surveyed, along with the polyethylene tank. Rooms 135 and 61A are also equipped with sink drains previously connected to a sump in the State-licensed area of the building. No contamination has been 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 draws 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

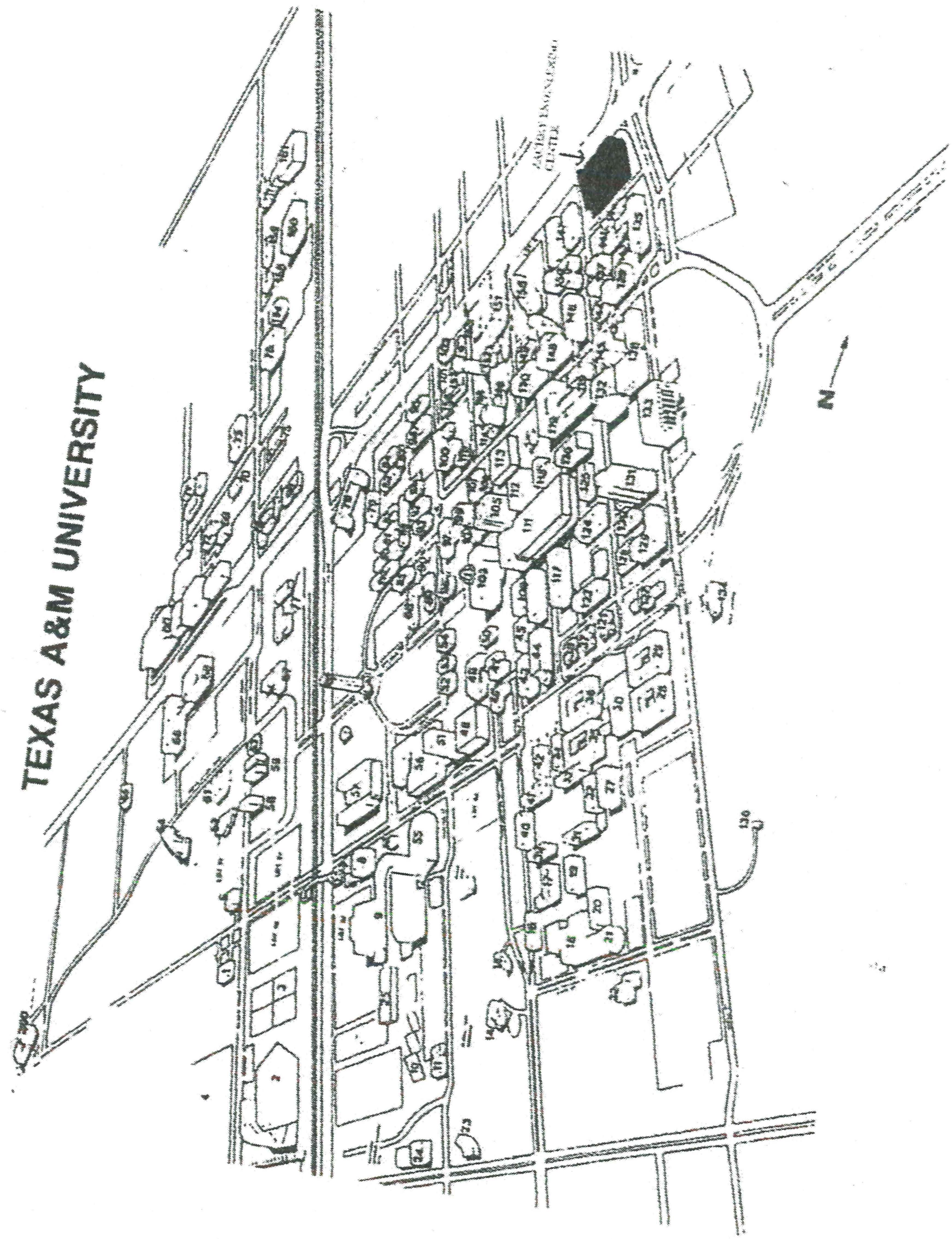


Figure 2 – Cut-away View of AGN-201M Reactor

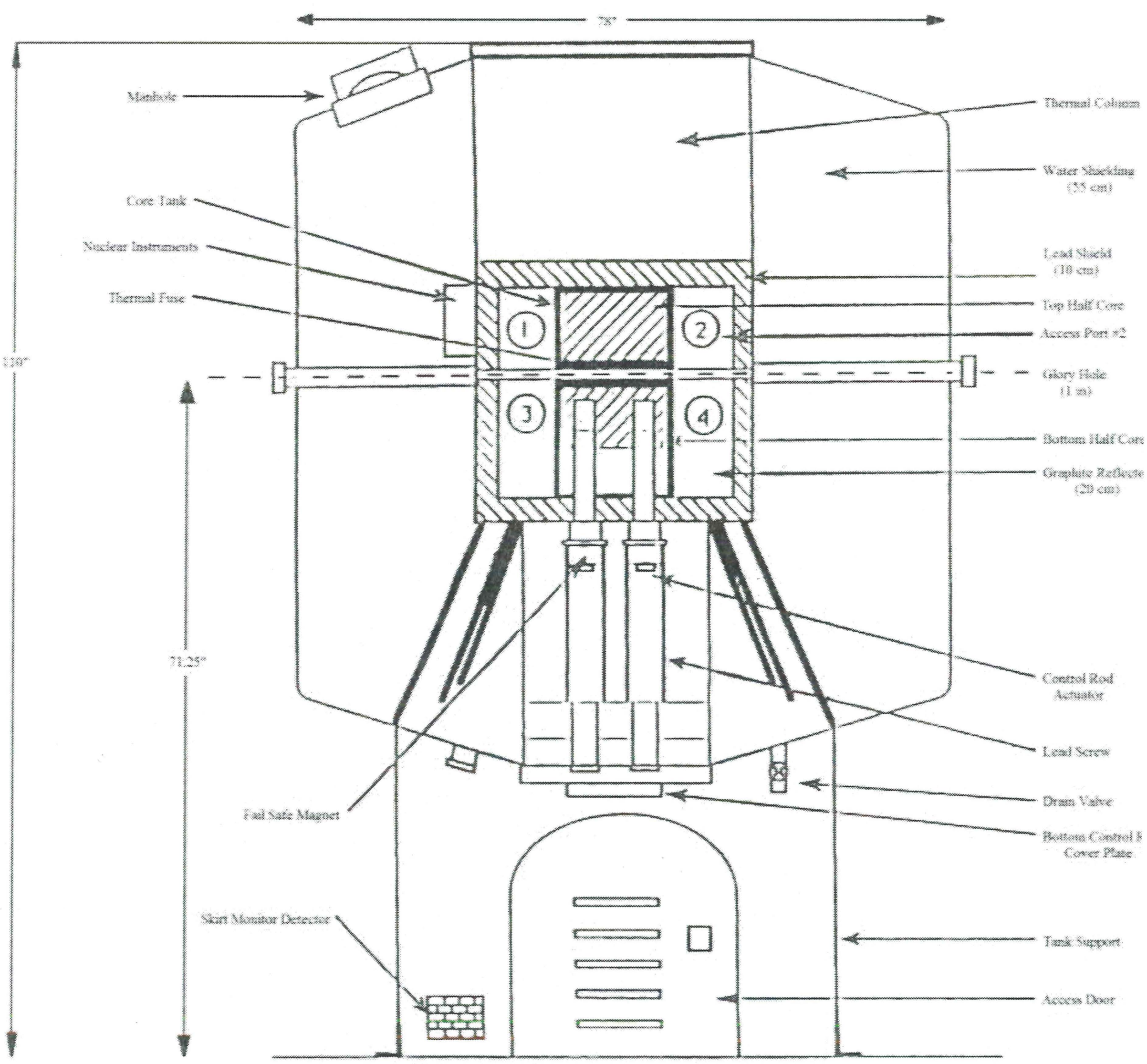


Figure 2a – AGN-201M Reactor without block shielding, as currently located in Zachry Engineering Center

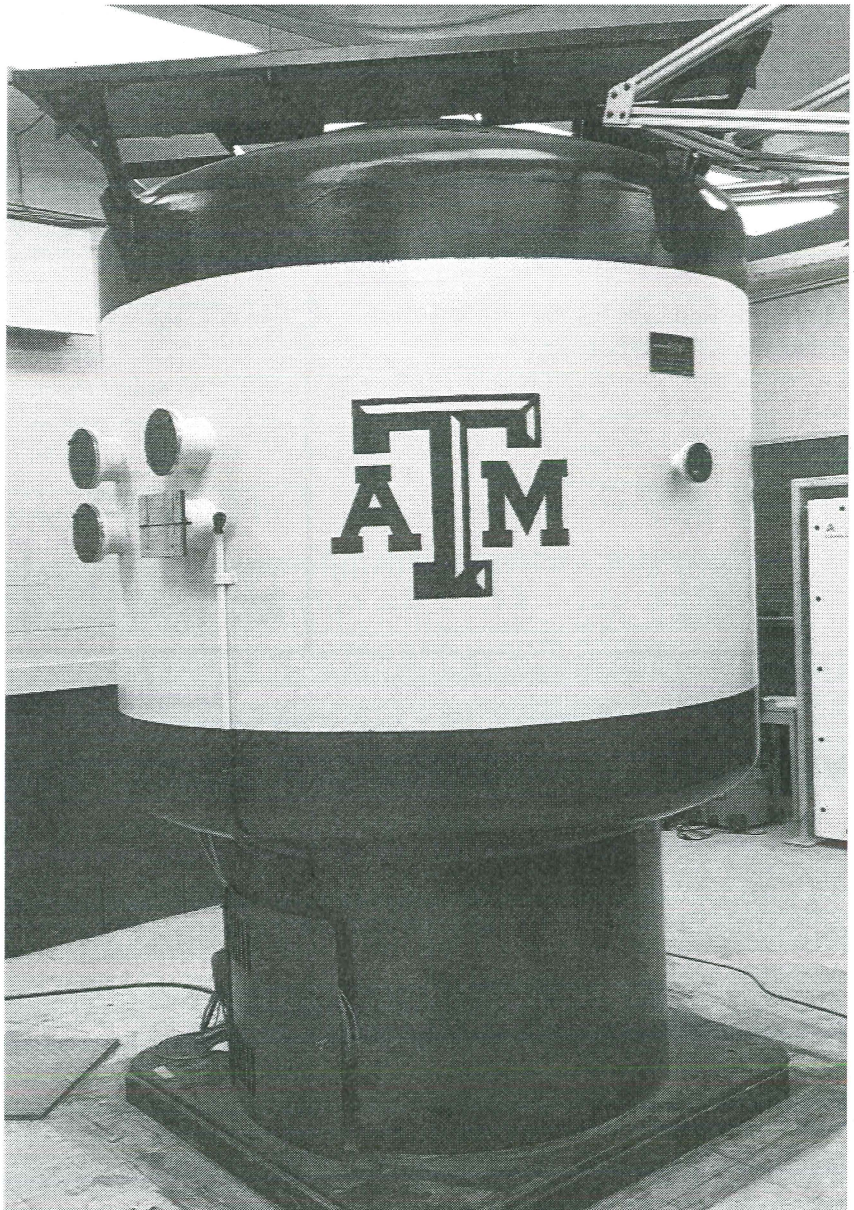


Figure 3 - Reactor Facility Ground Floor; Bolded outline indicates primary Reactor Site Boundary

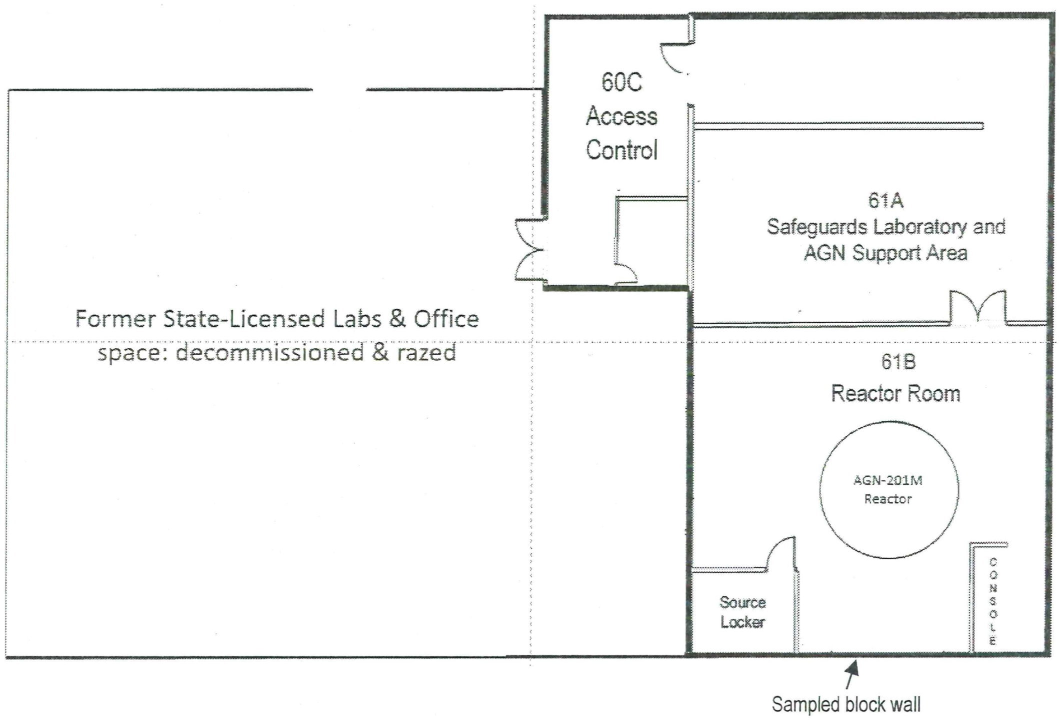


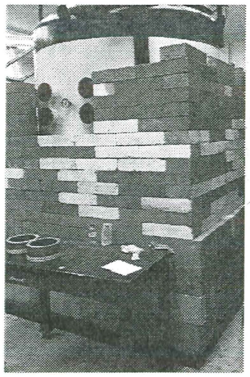
Figure 4 – Reactor Facility First Floor; Bolded outline indicates primary Reactor Site Boundary

+4.0 RADIONUCLIDE CONTAMINANTS AND CRITERIA

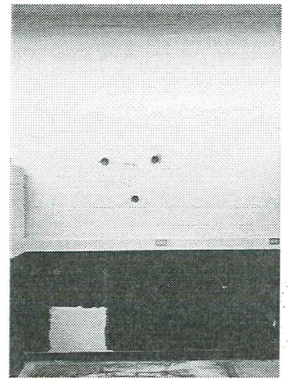
AGN-201M reactor operations in the Zachry Engineering Center began in 1972 and concluded in 2014. During the school years of 1999/2000 through 2009/2010, the reactor was not operated. In other years, annual operating watt- hours ranged from 4.32 to 82.36. Since the 2009/2010 school year, the total operating 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 University Radiological Safety staff have not identified any detectable removable contamination on reactor components or reactor room surfaces. Recent scoping surveys have not detected any fixed or removable contamination on surfaces in rooms 61A and 61B. Direct measurements have identified gamma-radiation levels up to approximately 5 microrem/hr in close proximity to the reactor shield tank; these radiation levels are likely due to fuel, fission products, and activation products within the inaccessible reactor core or core tank. Considering the low power level and limited operating time, low neutron fluence rate (1.5 x 10^8 n/cm^2-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 ^{152}Eu (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. Laboratory analyses (gamma spectrometry and liquid scintillation counting) of core samples from 4 innermost blocks around the reactor and 3 from the block wall in line with the glory hole did not contain detectable activation products. These radionuclides comprise the potential concrete activation source term.

No activity was detected above tabulated 95% C.L. MDCs listed below

		Cement Block wall: IW-1	Cement Block wall 2: IW-2	Cement Block wall 3: IW-3	RX shield block: E1	RX shield block:S1	RX shield block: N1	RX shield block: W1
134Cs	MDC, pCi/g	4.77E-02	4.52E-02	7.01E-02	7.87E-02	8.06E-02	5.54E-02	6.01E-02
60C	MDC, pCi/g	4.30E-02	4.11E-02	6.72E-02	7.69E-02	7.64E-02	7.87E-02	5.80E-02
152Eu	MDC, pCi/g	1.09E-01	1.40E-01	1.29E-01	1.31E-01	1.70E-01	1.64E-01	1.30E-01
154Eu	MDC, pCi/g	1.01E-01	1.30E-01	2.26E-01	2.17E-01	1.86E-01	1.99E-01	1.84E-01
3H	MDC, pCi/g	7.63E+00	7.85E+00	8.08E+00	7.96E+00	8.07E+00	7.88E+00	8.06E+00
14C	MDC, pCi/g	6.26E-01	6.01E-02	6.21E-01	5.98E-01	6.14E-01	6.07E-01	1.00E+00



Shield blocks around the AGN-201M reactor



Core-sampled block wall opposite glory hole

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 $^{239}PuBe$ special-form neutron source containing up to 16 grams of ^{239}Pu for use in reactor operation. This source has been leak tested (no contamination was detected) and removed from the AGN-201M reactor in preparation for transportation to storage

Following removal of the reactor and associated components from the facility, additional screening measurements and samples of concrete from under the reactor will be obtained to confirm the status of radiological contamination or activation. This includes removal of asbestos floor tile beneath the shield tank to allow for survey/sampling of the underlying concrete surface for laboratory analyses. Surfaces or media found to contain residual contamination or activation in excess of release criteria will be remediated or dispositioned as radioactive waste.

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 1) 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 the University.

Table 1 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 University's radiation safety program has a policy of "no detectable activity" for unrestricted use and release. "No detectable activity" is interpreted by the University as not exceeding twice the background level (Ref 1). The instruments used for direct measurements of residual activity have nominal material specific background levels of 3 alpha counts per minute (cpm) and 350 beta cpm (Ref 6). Therefore, net count rates exceeding 3 alpha and 350 beta cpm will be indicative of contamination exceeding the University's criterion. Similarly, the smear counter instrument has nominal alpha and beta background levels of up to 3 and 50 cpm and removable contamination exceeding net count rates of those levels will exceed the University's criterion. For the instruments to be used for direct measurements, 3 net alpha cpm is equivalent to about 33 alpha disintegrations per minute per 100 cm^2 (dpm/100 cm^2) while 350 net beta cpm is equivalent to 3500 beta dpm/100 cm^2. For smears, the twice background count rates are equivalent to 9 alpha dpm/100 cm^2 and 210 beta dpm/100 cm^2. These dpm/100 cm^2 equivalents are well within the values presented in Table 1. Meeting the University's Radiological Safety criteria for release will therefore also satisfy the NRC requirements. For comparison, Table 2 also presents NRC acceptable screening values for commonly-encountered radionuclides comprising surface contamination. This clearly demonstrates

the conservatism in the Texas A&M policy for unrestricted release. As noted, no contamination has been detected and no specific radionuclides have been identified during extensive scoping surveys; accordingly, gross alpha and beta-gamma surveys limits will be applied (i.e., not to exceed twice background, alpha and beta-gamma activity evaluated independently).

Table 2 NUREG 1757: Table H-1 Acceptable License Termination Screening Values of Common Radionuclides for Building-Surface Contamination (examples)

Radionuclide	**Table 1 values (dpm/100 cm^2)**	**Acceptable screening levels for unrestricted release (dpm/100 cm^2) - sum of fractions applies to mixtures** ***Values provided for comparison***
Co-60	5,000	7100
Mn-54	5,000	32,000
H-3	5,000	120,000,000
C-14	5,000	3,700,000

Measured background levels within the Zachry Engineering Center are between 6 and 10 microrem/hr. The University's "less than twice background" criterion will be met by confirming dose rates from any residual licensed material are no more than 5 microrem per hour above background at 1 meter from the surface, 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 3.

Table 3 Survey Unit Area by Classification

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

Table 4 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 4 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 University Radiation Safety Program procedures. Following removal and transfer of the reactor and associated components, drains, ducts, diffusers, grates, cable trays, etc., will be accessed to enable surveys. Any surfaces or construction media found to contain residual activity above the release criteria will be remediated or removed. Building surfaces will be 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 is likely (Class 1) or possible (Class 2) will be gridded at 1-meter intervals. Grid origins will be 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 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.

Surveys will be in the order indicated above. The rigor of surveys will follow the graded approach, based on the likelihood of contamination. Table 5 indicates the survey rigor for various contamination classifications.

6.4 Survey Instrumentation

Table 5 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 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, i.e., all net measurements must be less than the 3 alpha and 350

beta cpm nominal count rates. 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

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

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/4 m^2 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.

Table 6 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

Table 6 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.

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 source efficiencies of 0.25 for alpha emitters and 0.5 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 0.5 cm of the surface and passing the detector over the surface at a rate of approximately ½ detector width per second, while monitoring the audible output of the scaler/ratemeter for immediate identification of increases in count rate. When an alpha count is detected, the detector movement will be halted at the location for approximately 10 seconds to detect a possible elevated count rate.

A Ludlum Model 19 gamma scintillation detector will be used for gamma scans. Gamma scans will be performed with the detector in close proximity to the surfaces of interest while monitoring the audible output signal. General area gamma monitoring with the detector approximately 1 m above the floor will also be performed.

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.

6.7 Removable Contamination Measurements

A smear for removable activity will be performed at each static surface activity measurement location. A 100 cm^2 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).

Media samples (if necessary) will be submitted to General Engineering Laboratories for analyses for gamma emitters and low energy beta emitters (^{14}C and ^{3}H).

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. 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 University criteria of twice background. In addition, the limits established for the project (see section 4) will be converted to units of dpm/100 cm^2 for comparison with Table 1 values.

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	10	985	0.50	0.46	0.25	0.50
43-68	126	3	350	0.36	0.36	0.25	0.50
2929	N/A	1	52	0.33	0.24	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 that probability 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
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)

Using these parameters, detection probability calculations for a contamination level of 100 dpm/100 cm^{2}were performed for a scan rate of ½ detector width per second (i.e., dwell times of 2 seconds). Whenever a count is detected, the detector is paused over the surface for 10 seconds to determine if 2 counts are identified during that time period. The probability of detection during that time period is given by:

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

The resulting probabilities for this 2-step process are 94% for the 43-37 detector and essentially 100% for the 43-68 detector.

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:

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

The resulting value is 63 dpm/100 cm^2.

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:

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

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 as follows:

$$\text{Scan Sensitivity (dpm/100 cm}^2\text{)} = \frac{MDCR}{(0.5)^{1/2} * (\text{cpm/dpm})(\text{probe area}/100)}$$

The resulting values are 250 dpm/100 cm^2 for the 43-37 detector and 881 dpm/100 cm^2 for the 43-68 detector.

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:

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

The resulting value is 278 dpm/100 cm^2.

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 52 beta cpm; 4π detection efficiencies are 0.33 alpha and 0.24 beta. Using the same equation (without probe area correction) as above for direct measurements yields removable activity MDCs of 15 alpha dpm/100 cm^2 and 105 beta dpm/100 cm^2.

References

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

2. Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions, NUREG/CR-1507, U.S. 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)