UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 7, 2021

Ms. Charlotte Engstrom, Vice President
and General Counsel
General Atomics
P.O. Box 85608
San Diego, CA 92186-9784

SUBJECT: GENERAL ATOMICS TRIGA REACTOR FACILITY – APPROVAL OF LICENSE TERMINATION BASED ON THE FINAL STATUS SURVEY REPORT AND SUPPORTING INFORMATION (CAC 000083; EPID L-2019-DF1-0000)

Dear Ms. Engstrom:

By letter dated December 14, 2020, General Atomics (GA, the licensee) requested the U.S. Nuclear Regulatory Commission (NRC) to terminate GA's licenses for its TRIGA Mark I and Mark F non-power research reactors, License No. R-38 and License No. R-67, respectively, which are located at GA's TRIGA Reactor Facility in San Diego, California. In support of its request, GA submitted its Final Status Survey Report (FSSR) for the facility, in which it documented, among other things, the levels of residual contamination located at the site. This submittal supports termination of the GA operating licenses, which were issued pursuant to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10 of the *Code of Federal Regulations* (10 CFR). The proposed action would terminate the GA Part 50 licenses now that all decommissioning activities at the TRIGA facility are complete in accordance with the GA Decommissioning Plan approved by the NRC on August 12, 1999.

The GA FSSR provides the details of the licensee activities related to characterizing, identifying, and remediating the remaining residual radioactivity at the GA TRIGA Reactor Facility site to a level that allows the site to be released for unrestricted use. The GA FSSR also describes how the licensee confirmed the extent and success of remediation through radiological surveys and subsequent analyses of the survey results using NRC-approved guidance for conducting final status survey (FSS) activities. The NRC staff has completed its review of the GA FSSR to ensure that the site, in aggregate, meets the criteria for unrestricted release contained in Subpart E, "Radiological Criteria for License Termination," of 10 CFR Part 20, "Standards for Protection Against Radiation," which is necessary to terminate the Part 50 licenses.

In connection with the license termination, enclosed are two copies of Amendment No. 33 to Indemnity Agreement No. B-9. Please sign and return one copy to this office.

Also enclosed is the NRC staff's evaluation and independent analysis of the GA FSSR in support of the license termination decision, as well as a discussion of other inspection and verification activities that were completed to support termination of the GA TRIGA operating licenses. The NRC has now terminated License No. R-38 and License No. R-67, and NRC oversight of the GA TRIGA Reactor Facility is now ended. A Notice of License Termination will be sent to the Office of the Federal Register for publication.

The NRC staff has reviewed the residual radioactivity values in the GA FSSR and compared them to the trigger values in the 2002 Memorandum of Understanding (MOU) between the NRC and the U.S. Environmental Protection Agency (EPA) entitled "Consultation and Finality on Decommissioning and Decontamination of Contaminated Sites." Based on this review, the residual radioactivity in soil and groundwater at the site do not exceed the trigger values in the MOU and, as such, consultation with EPA in accordance with the MOU is not required.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at https://www.nrc.gov/reading-rm/adams.html.

If you have any questions concerning this evaluation please contact me or Marlayna Doell, the General Atomics Project Manager, at (301) 415-3178 or via e-mail at Marlayna.Doell@nrc.gov.

Sincerely,

Signed by Roberts, Ashley
on 12/07/21

Ashley B. Roberts, Deputy Director
Division of Decommissioning, Uranium Recovery,
and Waste Programs
Office of Nuclear Material Safety
and Safeguards

Docket Nos.: 50-089 and 50-163
License No.: R-38 and R-67

Enclosures:
1. Staff Evaluation
2. Two copies of Amendment No. 33 to
Indemnity Agreement No. B-9

cc w/enclosure: Distribution via ListServ

SUBJECT: GENERAL ATOMICS TRIGA REACTOR FACILITY – APPROVAL OF LICENSE TERMINATION BASED ON THE FINAL STATUS SURVEY REPORT AND SUPPORTING INFORMATION (CAC 000083; EPID L 2019-DF1-0000).
Date: December 07, 2021.

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EVALUATION BY THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

FINAL STATUS SURVEY REPORT AND LICENSE TERMINATION

GENERAL ATOMICS

GENERAL ATOMICS TRIGA REACTOR, MARK I AND MARK F, FACILITY

DOCKET NOS. 50-089 AND 50-163

1.0 INTRODUCTION

By letter dated December 14, 2020 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML21012A268), General Atomics (GA, the licensee) requested the U.S. Nuclear Regulatory Commission (NRC) to terminate General Atomics' licensees for its Mark I (License R- 38) and Mark F (License R-67) TRIGA® research reactors, both of which are located at GA's TRIGA Reactor Facility, in San Diego, California. In support of its request, GA submitted its Final Status Survey Report (FSSR) for the facility, in which it documented, among other things, the levels of residual contamination located at the site. A collection of decommissioning and license termination information, including the GA decommissioning plan (DP) and associated NRC safety evaluation, as well as Revision 1 of the final status survey (FSS) Plan, is available at ADAMS Package Accession No. ML21246A250. This submittal supports termination of the GA operating licenses, which were issued pursuant to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10 of the *Code of Federal Regulations* (10 CFR). The proposed action would terminate the GA Part 50 licenses now that all decommissioning activities at the TRIGA facility are complete in accordance with the GA DP approved by the NRC on August 12, 1999.

The GA FSSR provides the details of the licensee activities related to characterizing, identifying, and remediating the remaining residual radioactivity at the GA TRIGA Reactor Facility site to a level that allows the site to be released for unrestricted use. The GA FSSR also describes how the licensee confirmed the extent and success of remediation through radiological surveys and subsequent analyses of the survey results using NRC-approved guidance for conducting FSS activities. The NRC staff has completed its review of the GA FSSR to ensure, in particular, that the site, in aggregate, meets the criteria for unrestricted release contained in Subpart E, "Radiological Criteria for License Termination," of 10 CFR Part 20, "Standards for Protection Against Radiation." The NRC staff's safety, technical, and compliance evaluation is described below.

2.0 FACILITY BACKGROUND

The GA TRIGA Reactor Facility in San Diego, California, is located on the Torrey Pines Mesa within the larger GA campus. The TRIGA Mark I was the initial prototype TRIGA reactor, achieved initial criticality on May 3, 1958, and was in continuous operation until late 1997. On October 29, 1997, the TRIGA Mark I license (License No. R-38) was amended to possession only. The TRIGA Mark F achieved initial criticality on July 2, 1960 and was in continuous operation until March 22, 1995. The TRIGA Mark F license (License No. R-67) was amended to possession only in 1995. In 2010, all irradiated fuel elements from the TRIGA reactors located on the Torrey Pines Mesa were shipped to an authorized off-site storage facility at the Idaho National Laboratory. By letter dated April 18, 1997, as supplemented by letters dated November 20, 1998, January 28 and 29, February 3, April 22, May 3 and 12, and June 15, 16, and 22, 1999, GA submitted a request to the NRC to approve the TRIGA Reactor Facility DP. The NRC approved the GA DP by Amendment No. 36 to License No. R-38 and Amendment No. 45 to License No. R-67, dated August 12, 1999.

In February 2020, GA submitted Revision 2 of the "TRIGA Reactor Facility Final Status Survey Plan" (ADAMS Accession No. ML20049A039), which the NRC staff determined was consistent with the decommissioning guidance and methodology contained in NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" (ADAMS Accession No. ML003761445), and NUREG-1757, "Consolidated Decommissioning Guidance" (ADAMS Accession No. ML14093A221). The GA decommissioning activities included decontamination, dismantlement, and demolition of various systems, structures, and components followed by MARSSIM-based FSSs. The FSS was performed to demonstrate that the residual radioactivity remaining at the GA TRIGA Reactor Facility site satisfies the NRC's release criteria in 10 CFR 20.1402, "Radiological criteria for unrestricted use," which are: (1) an annual dose limit of less than 25 millirem per year (mrem/yr) Total Effective Dose Equivalent (TEDE) to an average member of the critical group (i.e., member of the public) and (2) the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). During site remediation, the licensee found a buried pipe with residual contamination that exceeded the previously approved release criteria. The dose from the pipe was evaluated separately as described in Section 3.6 below. The residual radioactivity in all other areas of the site were evaluated based on the previously approved release criteria and survey methodology.

By letter dated December 14, 2020, GA submitted the FSSR for the TRIGA Reactor Facility and requested the termination of the GA TRIGA licenses. The NRC staff reviewed the FSSR, which states that the criteria for license termination set forth in the GA licenses, and as established in the previously submitted DP and FSS Plan, have been satisfied. Supplemental information was provided in emails from the licensee dated February 26 and May 18, 2021, which addressed additional questions and items requiring clarification that were provided to the licensee by the NRC staff during the review of the FSSR.

3.0 EVALUATION

3.1 Applicable Release Criteria

The TRIGA reactors are located on an approximately 415-acre site (“the original site”), some of which was home to other GA activities licensed by the NRC. GA discontinued these other activities and decommissioned the buildings and site associated with them in accordance with NRC-approved decommissioning plans (DPs), including land decontamination. GA obtained the release of these land areas for unrestricted use based upon meeting the radiological release criteria provided in the NRC-approved DPs. Examples include GA's Hot Cell DP and the GA Site DP. Both of these decommissioning plans contain tables with soil and concrete rubble radiological release criteria (in picocuries per gram [pCi/g]) for specific individual radionuclides of concern. With only one exception, these NRC-approved plans and the set of radiological release criteria contained therein have been the basis for the release for unrestricted use of every building (~35) and all land areas comprising GA's original site. The only exception has been GA's TRIGA Reactor Facility (and associated licensed land area), which is located in the middle of the GA campus.

GA's NRC-approved TRIGA Reactor Facility DP (July 1999) does not contain a table of isotope-specific radiological release criteria applicable to activated concrete in the reactor pit configuration or to activated or contaminated soil (e.g., the soil behind the biological shield). Instead, the criteria are expressed as acceptable surface contamination levels (floors and walls), and exposure rates at one meter above the surface, which cannot be directly translated into soil and concrete rubble radiological release criteria. These radiological release criteria were chosen consistent with the guidance contained in NRC Regulatory Guide (RG) 1.86, “Termination of Operating Licenses for Nuclear Reactors,” dated June 1974, and NUREG/CR-5849, “Manual for Conducting Radiological Surveys in Support of License Termination,” draft for comment dated June 1992. These documents represented the most contemporary guidance available for the selection of NRC-approved generic radiological release criteria, as well as an acceptable process for conducting FSS, when the TRIGA Reactor Facility DP was assembled by GA in April 1997 and subsequently revised in January 1999.

Subsequent to the NRC’s approval of the TRIGA Reactor Facility DP in June 1999, the staff consolidated and updated the policies and guidance of the NRC’s decommissioning program in a three-volume NUREG series, NUREG-1757, which was first published in September 2003. NUREG-1757 replaced NUREG-1727, “NMSS Decommissioning Standard Review Plan,” and NUREG/BR-0241, “NMSS Handbook for Decommissioning Fuel Cycle and Materials Licensees,” and became the new standard for reactor and materials licensees seeking to terminate their NRC licenses, thereby rendering RG 1.86 mostly obsolete (note that RG 1.86 has since been withdrawn by the NRC, effective August 12, 2016 [81 *Federal Register* 53507]). In addition, NUREG/CR-5849 was replaced by NUREG-1575, which was first published in December 1997.

As a result, by letter dated December 18, 2015 (ADAMS Accession No. ML15362A506), as supplemented by letter dated August 15, 2016 (ADAMS Accession No. ML16242A319), GA requested to update the isotope-specific radiological release criteria applicable to the decontamination of GA's TRIGA Reactor Facility. This request was approved by the NRC in a letter dated February 1, 2017 (ADAMS Accession No. ML16285A300). The approval updated the DP to: (1) be consistent with newer decommissioning guidance documents available since the GA DP was approved in 1999, and (2) provide clear, isotope-specific radiological release criteria for the portions of the facility undergoing dismantlement and decontamination in order to ensure that any remaining residual radioactivity would be below the level required for unrestricted site release. As part of this review, the NRC staff concluded that this approach represented an acceptable method for GA to meet the unrestricted release criteria requirements established in the NRC-approved DP by following the more contemporary guidance.

Specifically, GA adopted the derived concentration guideline levels (DCGLs) contained in NUREG-1757, Volume 2, “Characterization, Survey, and Determination of Radiological Criteria,” for the screening of soil as the isotope specific radiological release criteria to be used for release for unrestricted use of residual soil and concrete at the GA TRIGA Reactor Facility, with the exceptions of Europium-152 (Eu-152) and Europium-154 (Eu-154), as is described in more detail in Section 3.3. Use of the NUREG-1757 screening DCGLs during the FSS and license termination phase for the remainder of the GA decommissioning project was considered a conservative approach compared to use of the radiological release criteria contained in the original TRIGA Reactor Facility DP because the screening values are significantly lower than those in the 1999 version of the GA DP. In addition, using these criteria to establish the appropriate release guidelines for the activated concrete and soil remaining at the GA site would ensure that the revised approach meets the previously approved radiological release limits.

The NRC staff also approved GA’s proposal to remove only those portions of the TRIGA Reactor Facility concrete structures and soils that exceed the screening levels for the radionuclides of concern, which were Cobalt-60 (Co-60), Strontium-90 (Sr-90), Cesium-137 (Cs-137), Eu-152, and Eu-154 for the GA facility. Further, the staff approved GA’s proposal to demonstrate that the facility meets the radiological release criteria for license termination using accepted sampling and measurement techniques found in MARSSIM. The use of MARSSIM ensured that the “sum of fractions” (SOF), or unity rule, was used to evaluate the remaining residual activity from all isotopes against the screening values since Table H.2, “Screening Values of Common Radionuclides for Soil Surface Contamination Levels,” in Appendix H, “Criteria for Conducting Screening Dose Modeling Evaluations,” of NUREG-1757 is intended for the evaluation of single radionuclides only. The use of the default screening values from NUREG-1757, Table H.2, for the radionuclides of concern identified in the GA TRIGA Reactor Facility FSSR is acceptable to the NRC staff as a means to demonstrate compliance with the radiological requirements for unrestricted site release of less than 25 mrem/yr TEDE and ALARA in 10 CFR 20.1402.

3.2 Area to be Released

The area the licensee intends to release once the GA TRIGA Reactor Facility licenses are terminated consists of approximately 60 acres that GA partitioned into 16 individual survey units and one subunit, consisting of eleven Class 1 survey units (as defined below in § 3.3.3 of this safety evaluation), five Class 2 survey units, and one Class 3 survey unit ranging in size from 5.6 to 886 square meters. GA performed the FSS for each of these survey units in accordance with the GA TRIGA FSSR, MARSSIM, and numerous GA implementing procedures. Using the MARSSIM guidance, GA collected and analyzed a total of approximately 780 systematic and judgmental samples in the 17 survey units.

The licensee stated that the FSSR was prepared to provide a complete and unambiguous record of the as-left radiological status, relative to the specified release criteria, consistent with the guidance provided in NUREG-1757. Sufficient data and information is contained in the GA FSSR to enable an independent evaluation of both the survey activities and the derived results, as well as to provide a summary of the survey results and the overall conclusions, which demonstrate that the site meets the radiological criteria for unrestricted use including the ALARA criterion. The NRC staff reviewed the FSSR in its entirety for consistency with the associated guidance and records; the results are summarized below.

3.3 Design of Final Status Surveys

The basis for the GA TRIGA FSS is the NRC-approved TRIGA Reactor Facility DP, the TRIGA Reactor Facility Final Status Survey Plan, Revision 2, NUREG-1757, and MARSSIM. The purpose of an FSS is to demonstrate that the total residual radioactivity in each survey unit meets the radiological requirements for unrestricted release. The licensee stated that for surfaces and structures, the release criteria listed in GA's TRIGA Reactor Facility DP were used. Further, GA indicated that for land areas and the Mark I and Mark F reactor pits, the screening values listed in NUREG-1757, Volume 2, Table H.2 were used. GA applied the unity rule to the release criteria for survey units with multiple radionuclides of concern.

The default screening values from NUREG-1757 are the concentration of an individual radionuclide that is equivalent to 25 mrem/yr TEDE. The default screening values of the radionuclides of concern for surfaces and structures at the GA TRIGA Reactor Facility are listed in Table 3.1, "Default Surfaces & Structures Screening Values for Nuclides of Concern," of the GA TRIGA FSSR. The default screening values of the radionuclides of concern for soil and concrete at the GA TRIGA Reactor Facility are listed in Table 3.2, "Default Concrete & Soil Screening Values for Nuclides of Concern," of the GA TRIGA FSSR. The default screening values used for the radionuclides of concern in concrete and soil are from Table H.2 of NUREG-1757, Volume 2, with one exception: a value of 7.0 pCi/g for Eu-152 is used to adopt the value from the NRC Memorandum of Understanding (MOU) with the U.S. Environmental Protection Agency (EPA) (ADAMS Accession No. ML022830208). GA noted that this value is lower than the NUREG-1757, Table H.2 default screening value of 8.7 pCi/g. The lower EPA MOU screening value for Eu-152 is more conservative and is therefore considered acceptable by the NRC staff for use in evaluating the residual radioactivity at the GA TRIGA Reactor Facility site, as documented in the safety evaluation report for the DP (add detail).

3.3.1 Background Measurements

The radionuclides of concern for the GA TRIGA Reactor Facility are Co-60, Sr-90, Cs-137, Eu-152, and Eu-154. GA took background radiation measurements for concrete, cinder block, and metal from areas outside the potential influence of the TRIGA Reactor Facility with similar physical, chemical, geological, and radiological characteristics as the survey units being evaluated. GA also used an ambient or general area background measurement. GA subtracted the average background from all gross radiological measurements.

In accordance with MARSSIM Section 4.5, “Select Background Reference Areas,” the NRC staff determined that a background reference area measurement was not needed for land areas in which none of the radionuclides of concern are present in background. The NRC staff evaluated the information provided by the licensee in the GA TRIGA FSSR and verified that the licensee followed the guidance in MARSSIM to measure the background. Accordingly, the NRC staff concludes that the methodology GA used for determining background at the TRIGA Reactor Facility is acceptable.

3.3.2 Data Quality Objectives

The Data Quality Objectives (DQOs) for the TRIGA Reactor Facility FSS for static measurements and scanning minimum detectable concentrations (MDCs) are less than 50 percent of the DCGLs or default screening values from NUREG-1757 and adopted in the TRIGA DP; individual measurements are at a 95 percent confidence level; and the decision error probability rates are 0.05 for both scenarios (α and β). The null hypothesis was that the residual radioactivity in the survey unit exceeds the release criterion, while the alternate null hypothesis was that the residual radioactivity in the survey unit does not exceed the release criterion.

MARSSIM Section 5.5.2, “Survey Design,” states that appropriate radiological survey design begins with the development of DQOs that are applicable to the site-specific considerations for the decommissioning facility. Appendix D, “The Planning Phase of the Data Life Cycle,” of MARSSIM further details the DQO process as it applies to decommissioning surveys. Based on an evaluation of the information provided by the licensee in the GA TRIGA FSSR and supporting documents, the NRC staff determined that the DQO process for the TRIGA Reactor Facility surveys follows the guidance in MARSSIM and is therefore acceptable.

3.3.3 Area Classifications and Survey Units

The basis for survey unit area classifications within the TRIGA Reactor Facility is historical site assessment (HSA) data and radiological site characterization survey results. The HSA and characterization survey results were used to determine which areas of the GA TRIGA Reactor Facility site were non-impacted and which were impacted by previous licensed activities that resulted in the release of radioactivity. Impacted areas are specified as Class 1, 2, or 3 depending on the potential for exceeding a dose above the unrestricted release criteria and/or containing small areas of elevated activity from site operations. The TRIGA Reactor Facility has eleven Class 1, five Class 2, and one Class 3 survey units.

MARSSIM Section 4.4, “Classify Areas by Contamination Potential,” discusses an approach for classification of survey areas based on contamination potential. The staff verified that GA used the guidance on survey unit classification and sizes in MARSSIM Section 4.4 to establish the TRIGA Reactor Facility survey units. Accordingly, the NRC staff concludes that survey unit classification is based on contamination potential, historical use, and characterization data, and is acceptable.

3.3.4 Scanning, Static, and Removeable Activity Measurements

Scanning survey coverage is determined by the classification of the survey unit and is used to identify and mark locations within the survey unit where the measured activity exceeds a predetermined action level. The licensee stated that for the floors in the GA TRIGA Reactor Facility, Class 1 areas received 100 percent scan coverage, Class 2 areas received 100 percent scan coverage, and Class 3 areas received 50 percent scan coverage. For all other structures, Class 1 areas received 100 percent scan coverage, Class 2 areas received 50 percent scan coverage, and Class 3 areas received 10 percent scan coverage. GA took static measurements for fixed contamination and wipe measurements for removeable contamination at any scan area locations where the activity exceeded the action level. Samples for analysis were also taken in each survey unit.

The NRC-approved TRIGA Reactor Facility DP describes the measurement methodology for the conduct of surveys in Section 4.4.2, “Measurement Methodology for Conduct of Surveys.” Fixed contamination survey protocol is described in Section 4.4.3, “Fixed Contamination Survey Protocol,” and the removeable contamination protocol is described in Section 4.4.4, “Removeable Contamination Survey Protocol.” MARSSIM Section 5.5.2.2, “Contaminant Present in Background – Determining Numbers of Data Points for Statistical Tests,” Section 5.5.2.3, “Contaminant Not Present in Background – Determining Numbers of Data Points for Statistical Tests,” and Section 5.5.2.4, “Determining Data Points for Small Areas of Elevated Activity,” define the methodology for calculating the minimum number of samples needed for statistical confidence. The appropriate MARSSIM tables can also be used to determine the minimum number of samples needed. The number of direct measurements to be obtained in each survey for use in the Sign Test is determined from MARSSIM Table 5.5, “Values of N for Use with the Sign Test,” which is based on MARSSIM Equation 5-2. GA used the Sign Test because the TRIGA radionuclides of concern are not present in background.

The NRC staff reviewed the protocol and methodology for scanning, static, and removeable activity measurements that were used to calculate the minimum number of radiological samples needed in the TRIGA Reactor Facility survey units and determined that GA followed the provisions of the DP and applicable provisions of MARSSIM Section 5.5.2, and therefore used acceptable methods for conducting radiological measurements as described in the GA TRIGA FSSR. The NRC staff also concluded that the minimum number of samples/measurements for each survey unit was correctly calculated by following the applicable MARSSIM protocol. Attachment G, “Gamma Spectroscopy Results,” of the GA TRIGA FSSR contains the gamma spectroscopy results, while Attachment H, “Statics, Wipes, and Exposure Rate Results,” includes the statics, removeable activity, and exposure rate measurement results. The staff’s review of the actual number of samples/measurements taken during the TRIGA Reactor Facility FSS verified that GA took significantly more than the minimum number required within each survey unit.

3.3.5 Determination of Sample Locations

MARSSIM Section 4.8.5, “Reference Coordinate System,” describes a reference coordinate system for relating survey measurements and sample locations to a specific starting point within a survey unit. Using the minimum number of samples determined using MARSSIM Table 5.5, as well as the necessary spacing of sample locations based on MARSSIM Equation 5-7 and Equation 5-8, GA established a random start location to systematically map the sample locations for Class 1 and Class 2 survey units. Class 3 survey unit sample locations were biased towards areas with the highest potential for residual contamination and employed a random measurement pattern as recommended by the guidance in MARSSIM Section 5.5.2.5, “Determining Survey Locations.” In addition to the systematic locations, a minimum of five judgmental sample locations were added to each survey unit based on process knowledge.

Section 3.17, “Determination of Sample Locations,” of the GA TRIGA FSSR addresses the determination of sample locations in Class 1, Class 2, and Class 3 survey units. The NRC staff also notes that GA took significantly more than the minimum number of samples/measurements determined from MARRSIM in each GA survey unit. Accordingly, the NRC staff determined that the sample locations selected are consistent with the guidance in MARSSIM and are therefore acceptable.

3.3.6 Survey Action Levels

GA established investigation or action levels based on radiological measurement results in order to verify that proper survey unit classifications and adequate survey methods were established. Survey unit locations where the measured activity exceeded the investigation levels were marked to determine the concentration, area, and extent of potential contamination. Class 1 investigation levels for scan or static measurements were any measurement greater than 50 percent of the DCGL. Class 2 investigation levels for scan or static measurements were any measurement greater than 20 percent of the DCGL. Class 3 investigation levels for scan or static measurements were any measurement greater than the MDC. The investigation level for removable contamination in any Class 1, Class 2, or Class 3 survey unit was any measurement exceeding 100 disintegrations per minute per 100 square centimeters (100 dpm/100 cm^2).

MARSSIM Section 5.5.2.6, “Determining Investigation Levels,” provides the basis for determining FSS investigation levels to indicate when additional radiological investigations may be necessary. The survey investigation levels presented in Table 3.6, “Survey Investigation Levels,” of the GA TRIGA FSSR are more restrictive than the example investigation levels from Table 5.8, “Example Final Status Survey Investigation Levels,” of MARSSIM. Based on an evaluation of the information provided by the licensee in the GA TRIGA FSSR, the NRC staff concluded that the survey investigation levels established by GA are conservative and consistent with the guidance in MARSSIM and are therefore acceptable.

3.4 Survey Instrumentation

Instrumentation types, specifications, and operating conditions are discussed in Section 4.4.1, “Instrumentation – Type, Specifications, and Operating Conditions,” of the TRIGA Reactor Facility DP. MARSSIM Section 6.5.3, “Instrument Selection,” Table 6.1, “Radiation Detectors with Applications to Alpha Surveys,” Table 6.2, “Radiation Detectors with Applications to Beta Surveys,” and Table 6.3, “Radiation Detectors with Applications to Gamma Surveys,” provide guidance on radiation detection instrumentation applicable to conducting FSS. Table 4.1, “Instruments Used at the TRIGA Facility,” and Table 4.2, “Typical Instrument Operating Parameters and Sensitivities,” of the GA TRIGA FSSR specify the types, operating parameters, and sensitivities of instruments used to conduct the FSS measurements.

The instrumentation used to perform FSS for the GA TRIGA Reactor Facility included the following: (1) gas flow proportional survey and floor monitors were used to perform surface surveys; (2) sodium iodide survey meters were used to perform soil and concrete scans; (3) scintillation detectors were used to measure total surface activity; (4) ion chambers were used to perform exposure rate measurements; (5) a laboratory gas flow proportional counter was used to measure removable activity from surface wipes; and (6) a cesium iodide scintillator was used as the pipe detector to measure activity in the buried pipe. The GA radiological survey instruments received daily response checks using National Institute of Standards and Technology (NIST) traceable sources; the daily background counts and source counts had an established acceptance range criterion of $\pm$ 20 percent; and the count times were sufficient to ensure statistical validity of the measurement data for total and removable contamination.

Based on the information provided by the licensee in the GA TRIGA FSSR and supporting documents, the NRC staff evaluated the instrument types and specifications required to perform FSS at the GA TRIGA Reactor Facility. Based on this review, the NRC staff determined that GA selected and used instruments to perform FSS at the GA site in accordance with MARSSIM. Accordingly, these instruments are acceptable to establish the final radiological status of the 17 GA TRIGA Reactor Facility survey units and demonstrate compliance with the radiological release requirements of 10 CFR Part 20, Subpart E.

3.4.1 Instrument Calibration

Section 4.1, "Instrument Calibration," of the GA TRIGA FSSR states that all FSS instruments (excluding the Ludlum Model 44-159-1 cesium iodide scintillator used for buried piping surveys) were calibrated every six months by GA following American National Standards Institute standard N323B-2003 and utilizing NIST traceable standards that align with the radiological characteristics of the radionuclides of concern. The pipe detector used by GA is the Ludlum Model 44-159-1 cesium iodide scintillator coupled with a Ludlum Model 2350-1 portable survey instrument using a 50-foot cord. Calibration of the pipe detector was performed by Ludlum and the detector sensitivity was determined at the GA Nuclear Calibration Laboratory.

MARSSIM Section 6.5.4, "Instrument Calibration," provides guidance on instrument calibration. The NRC staff reviewed the instrument calibration protocols and procedures for the survey instruments used to perform FSS at the GA TRIGA Reactor Facility and determined that they are consistent with the MARSSIM guidance. The NRC staff therefore determined that the calibration protocols and procedures for radiological measurement instrumentation used during the GA FSS are acceptable to determine the final radiological status of the 17 survey units.

3.4.2 Instrument and Source Efficiency Determination

The GA instrument efficiency calculations were consistent with International Organization for Standardization (ISO) 7503.1, "Evaluation of Surface Contamination," and were determined with the use of reference radiation sources with known emission rates per unit area. Instrument efficiency is defined as the net count rate divided by the activity of a known source, while the total source activity is defined as the net count rate divided by (instrument efficiency x size of probe area x surface efficiency). The surface efficiency for Sr-90 and Cs-137 was 0.5, which is recommended in ISO 7503.1. Alpha emitters and low energy beta emitters used the recommended source efficiency of 0.25. NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," also provides source efficiencies for common surface materials.

Section 6.5.4 of MARSSIM provides guidance regarding the calculation of instrument efficiency. The NRC staff reviewed the instrument and source efficiency calculations for the instruments used to perform FSS at the GA TRIGA Reactor Facility and determined that they conform to the MARSSIM guidance. The NRC staff therefore determined that the instrument and source efficiency calculations for radiological measurement instrumentation used during the GA FSS are acceptable to evaluate surface contamination and calculate instrument MDCs.

3.4.3 Determination of Counting Times, MDCs, and Counting Uncertainties

MARSSIM Section 4.7.1, "Selection of Instruments," recommends MDCs between 10 and 50 percent of the DCGL. GA selected background and survey measurement minimum counting times for the GA TRIGA Reactor Facility FSS to achieve MDCs at or below 50 percent of the DCGL for the hardest to detect radionuclide. The counting times and MDCs for each detector instrument and measurement type are identified in Table 4.1 and Table 4.2 of the GA TRIGA FSSR. GA calculated the static measurement and smear counting MDCs at the 95 percent confidence level based on NUREG-1507. GA calculated the ratemeter scanning for surface and structure MDCs at the 95 percent confidence level based on a combination of MARSSIM Equation 6-8, Equation 6-9, and Equation 6-10. The scan MDCs for soil areas are based on the area of elevated activity, depth of contamination, and the radionuclide using the calculations found in MARSSIM Section 6, "Field Measurement Methods and Instrumentation."

The counting uncertainty for total and removable contamination measurements at the GA TRIGA Reactor Facility followed MARSSIM Equation 6-15, as well as the uncertainty propagation formula from MARSSIM Section 6.8.3, "Uncertainty Propagation," and data from MARSSIM Table 6.9, "Areas Under Various Intervals About the Mean of a Normal Distribution." Using this approach, GA reported the total activity measurements in dpm/100 cm^2 $\pm$ 1.96 sigma or total measurement uncertainty. Based on this information, the NRC staff determined that the GA followed the guidance for calculating counting times, MDCs, and counting uncertainties found in MARSSIM and NUREG-1507, which is acceptable for reporting the data results in the GA TRIGA FSSR.

3.5 Final Status Survey Results

There are 17 survey units identified in the GA TRIGA FSSR: a total of eleven surveys units are identified as Class 1, five as Class 2, and one as Class 3 impacted areas. One of the Class 1 survey units included a buried pipe, which was discovered after the start of the GA FSS activities and identified as an area of concern that did not fit the previously approved release criteria. GA addressed this situation by performing a dose assessment to demonstrate that the potential yearly TEDE to a representative member of the critical group following license termination is well below the release criteria in 10 CFR Part 20, Subpart E. This buried pipe dose assessment and the NRC staff's evaluation are discussed in Section 3.6 below.

The GA FSS methods were based on survey units that contained surfaces, structures, and open land/soil areas. The survey data for surfaces and structures included wipe samples for removable contamination, scanning and static measurements for total activity, and exposure rate measurements. The survey data for open land/soil areas included volumetric sampling and analysis. The NRC staff reviewed the analytical results for all survey units, which were provided in Attachment G and Attachment H of the GA TRIGA FSSR, and determined that with the exception of the survey unit containing the buried pipe, all the results for survey unit radiological assessment were well below the screening values equivalent to 25 mrem/yr in NUREG-1757, Volume 2, Appendix H, Table H.2, as well as the more restrictive screening value for Eu-152 from the NRC/EPA MOU. The NRC staff determined that the analytical methods used for the survey units were adequate to detect the radionuclides of concern at or below their investigation levels. The NRC staff also reviewed the results of the gamma spectroscopy and Sr-90 analyses and confirmed that the only radionuclides of concern that were detected were those listed by the licensee as a radionuclide of concern in Section 5.3 of the FSSR. For the above reasons, the NRC staff concluded that the survey unit areas identified in the GA TRIGA FSSR meet the radiological release criteria of 10 CFR Part 20, Subpart E.

3.6 Buried Pipe Dose Assessment

As is described above, residual contamination in a buried drainpipe was not consistent with the previously approved release criteria for the site. To demonstrate that the calculated TEDE to the representative person of the critical group from residual radioactive material contained in the buried drainpipe at the GA TRIGA Mark F reactor is well below the release criteria stated in 10 CFR Part 20 Subpart E (i.e., 25 mrem/yr and ALARA), the licensee submitted a dose assessment of the material remaining in a buried drainpipe at the GA TRIGA Mark F Reactor as part of its FSSR on December 14, 2020. In emails from the licensee dated February 26 and May 18, 2021, the licensee provided additional information on the dose assessment in response to NRC questions as referenced in § 1.0 of this safety evaluation.

GA based the exposure scenarios it considered in the dose assessment on the site continuing to be used for its current purpose. According to the FSSR, the GA site contains nearly one million square feet (~ 93,000 m^2) of office space, including engineering, sophisticated test facilities, precision manufacturing installations, and advanced technology laboratories. The FSSR states that the site, along with much of the surrounding area, is designated for scientific research by the City of San Diego. The FSSR further indicates that the there is no significant agricultural activity in the local area and that no significant freshwater recreation areas exist within the local hydrological area.

GA considered three exposure scenarios in the dose assessment for the buried pipe: dose from a buried pipe located at depth, dose from a buried pipe in the surface soil, and external exposure dose to a worker.

3.6.1 Background

During a survey by NRC's contractor, Oak Ridge Associated Universities (ORAU), GA and ORAU personnel found a drain in the trench on the east side of the Mark F reactor pit. A count of 350,000 cpm was measured in the drain after the drain plug was removed. A small section of concrete was initially excavated to determine the extent of contamination. Once this concrete was removed, it was discovered that the drainpipe was broken and there was significant amount of soil settling (approximately 18 to 24 inches [46 to 61 cm]) under the Mark F floor. A larger section of the concrete floor was then removed to find the remaining drain line and to determine the level of soil contamination.

The pipe was located at a depth of 1.38 m below the reactor room floor. The pipe is cast iron with a 3-inch (7.6 cm) outer diameter and 1/8-inch (0.32 cm) wall thickness. A video of the inside of the pipe found considerable pipe scale that had flaked off and could be easily removed. The licensee stated that the inside of the pipe was cleaned until the reduction in detector count rates plateaued and diminishing returns were seen in the removal of contamination. The initial counts in the pipe were reported to be in the range of 20,000 to 40,000 cpm and were reported to be reduced to less than 10,000 cpm with an average reading of 3,300 cpm, including background. In October 2020, the licensee filled the pipe with a concrete slurry to fix removable contamination and to add shielding to reduce external exposure. The soil contamination under the broken pipe was evaluated separately as part of Survey Unit 5A as described above. According to Table 6.4 in the FSSR, the maximum SOF observed for this survey unit was 0.387, which would correspond to a dose of 9.68 mrem/yr.

3.6.2 Buried Pipe Source Term

GA determined the buried pipe source term by measuring the count rate in the pipe approximately every 6 inches (15.24 cm) using a Ludlum Model 44-159-1 cesium iodide scintillator. The average net count for these measurements was 2,200 cpm and the maximum net count was 5,280 cpm. Background readings were determined by taking measurements in a clean section of cast iron pipe located at depth in non-impacted soil on the site. The average reading for the background measurements was 936.1 cpm. GA determined the "in the field" detector efficiency using the calibrated GA irradiator range at the GA Nuclear Calibration lab. The detector sensitivity was determined to be 102 cpm/µR/hr. Based on this detector efficiency and the count rates measured in the contaminated pipe, GA determined the average and maximum exposure rates in the pipe to be 21.57 µR/hr and 51.76 µR/hr. GA also analyzed two samples of pipe scale using gamma spectroscopy and also analyzed the samples for Sr-90. Cs-137 and Sr-90 were the only radionuclides detected in the pipe scale.

GA used the MicroShield code to determine the Cs-137 activity on the inside of the pipe that would correspond to the average and maximum exposure rates estimated for the pipe. GA calculated that the average exposure rate would correspond to 7.922 µCi and the maximum exposure rate would correspond to 19.01 µCi. GA determined that the total mass of the pipe is 85.72 kg. From the above information, GA estimated an average Cs-137 concentration over the pipe of 92.41 pCi/g and a maximum concentration of 221.8 pCi/g.

In GA's initial FSSR submittal, a scaling factor of 0.803 (Sr-90 to Cs-137) was used to calculate Sr-90 concentrations of 74.25 pCi/g and 178.2 pCi/g based on the estimated average and maximum concentrations of Cs-137, respectively. GA based this scaling factor on a scaling factor previously developed for Sr-90 at the site. GA subsequently developed a scaling factor based on measurements of Sr-90 and Cs-137 in the vicinity of the pipe. GA sent eleven soil samples to TestAmerica for analysis of Cs-137 and Sr-90. GA initially provided the results of these analyses to the NRC in the February 2021 email and provided a revised version in May 2021 to correct a typo. Of the 9 samples that had Cs-137 and Sr-90 activities above the MDC, the ratio of Sr-90 to Cs-137 ranged from 0.38 to 7.07 with an average ratio of 1.848 and a standard deviation of 2.16. GA used the average ratio (i.e., 1.848) as a scaling factor to calculate an average Sr-90 concentration of 170.7 pCi/g based on the average Cs-137 concentration described above.

The NRC staff has reviewed GA's estimation of the source term in the buried pipe. The NRC staff agrees that the primary radionuclides are Cs-137 and Sr-90 based on the radionuclides of concern for the site and that these were the only radionuclides observed in the laboratory measurements. The NRC staff also concludes that the use of measured count rates and MicroShield modeling was an appropriate method for estimating the concentration of Cs-137 in the pipe given that MicroShield is designed to estimate gamma emission. The NRC staff performed independent MicroShield calculations and obtained comparable results. The NRC staff also reviewed the calculation of the Sr-90 scaling factors and finds that deriving these values from measured Cs-137 and Sr-90 activities in soil samples in the vicinity of the pipe is a reasonable approach, though the NRC staff notes that the expected transport of these two radionuclides in soil can be different and that the ratio might not remain constant throughout the soil. The use of the average ratio to calculate the Sr-90 activities does not appear to fully capture the uncertainty in the ratios seen in the soil samples. However, the average ratio in the soils is much higher than the ratio seen in the pipe scale and therefore the use of the soil measurements for modeling the pipe scale inventory appears to be conservative. Additionally, as described below, the NRC staff also performed a sensitivity analysis of the dose consequences if the Sr-90 was calculated using the maximum measured ratio and the resulting calculated dose was still below 25 mrem per year.

3.6.3 Exposure Scenarios

GA evaluated the potential dose from the residual contamination in the buried pipe using three different exposure scenarios: dose from a buried pipe located at depth (Scenario 1), dose from a buried pipe in the surface soil (Scenario 2), and external exposure dose to a worker (Scenario 3).

In Scenario 1, buried pipe at depth, the pipe was assumed to be located at its current depth (1.38 m below the TRIGA building floor), while in Scenario 2, buried pipe in surface soil, the pipe was assumed to be located at a depth of only 5 cm. In both Scenario 1 and Scenario 2, the pipe was assumed to instantaneously disintegrate, and the medium was assumed to become soil. No credit was taken in this analysis for the concrete slurry that was placed in the pipe. The volume of contaminated material was assumed to be equal to the pipe volume. According to the FSSR, the pathways analyzed for Scenarios 1 and 2 were external gamma, inhalation, and soil ingestion. However, as noted in the May 18, 2021 email, the Scenario 1 residual radioactivity (RESRAD) calculation inadvertently included all pathways except radon.

Scenario 3 evaluated the potential dose to a construction or utility worker who might be exposed to the residual radioactivity in the pipe during demolition of the TRIGA building. In Scenario 3, the pipe was not assumed to disintegrate, and the pipe was modeled as being filled with a concrete slurry. The external gamma pathway dose was the only pathway considered for Scenario 3.

The NRC staff reviewed the exposure scenarios and pathways analyzed. In view of the above, including projected future site use and the acceptability of detection and analysis methods, the NRC staff makes the following conclusions:

- The exposure scenarios selected are consistent with the current and expected future land uses for the site (i.e., land is used for office space and scientific research facilities).

- The evaluation of the pipe in both a degraded (Scenarios 1 and 2) and intact (Scenario 3) form appropriately considers the potential dose from the residual contamination in the pipe for any potential future degradation state of the pipe.

- The calculation of the potential dose from the residual radioactivity in the pipe at both its current depth (Scenario 1) and at the surface (Scenarios 2 and 3) appropriately considers the potential dose from the residual radioactivity in the pipe in its current depth as well as the potential dose if the pipe is brought to the surface in the future. Further, given the nuclides detected, the pathways selected (i.e., external gamma, inhalation, and soil ingestion) for Scenarios 1 and 2 are appropriate for the exposure scenario considered (i.e., a worker at a scientific facility who is exposed to a degraded pipe).

- The inadvertent inclusion of other scenarios in the Scenario 1 calculation was conservative and resulted in an overestimation of the potential dose.

- It was only appropriate to only consider the external dose pathway for Scenario 3 (i.e., a worker exposure to residual radioactivity in an intact pipe) because there would not be a dose associated with other pathways (e.g., inhalation or soil ingestion) if the pipe remains intact.

3.6.4 Dose Assessment Calculations

GA evaluated the potential dose from Scenarios 1 and 2 (buried pipe at depth and buried pipe in surface soil, respectively) using the RESRAD-ONSITE code (Version 7.2). As was discussed in the previous section, in both of these scenarios the pipe was assumed to disintegrate instantaneously and subsequently have the properties of soil. In Scenario 1, the contaminated pipe media was modeled as soil with a volume of 6,583 cm^3 (i.e., the volume of the pipe). The contaminated area was modeled as having a length equal to the pipe length (360 inches [914.4 cm]), a thickness equal to the pipe thickness (0.25 inches [0.635 cm]), and a width of 11.33 cm. The contamination was assumed to have 1.38 m of clean cover above it. GA based the concentrations derived for Cs-137 and Sr-90 on the average exposure rate (see Section 3.6.2). The total dose calculated by RESRAD for Scenario 1 was 1.854×10^{-6} mrem/yr (Table 1). GA used the same assumptions for Scenario 2 as for Scenario 1 except that GA modeled the cover depth for Scenario 2 as being 5 cm thick. The projected dose in the FSSR was reported as 6.647 mrem/yr, but the licensee subsequently identified that the Sr-90 concentration assumed in this calculation was not correct. A revised calculation using an updated Sr-90 concentration was provided in the May 2021 email and the final dose calculated for Scenario 2 was 6.672 mrem/yr (Table 1).

The projected dose from Scenario 3 (external exposure to worker) was calculated by the licensee using the MicroShield code. In this calculation, GA assumed the residual radioactivity to be on the inside surface of the pipe. GA modeled the pipe as being made of iron and being filled with a concrete slurry. GA calculated the external dose rate at a distance of 1 ft (30.5 cm) from the pipe. GA performed the dose calculation for a concrete cylinder that had residual radioactivity on its surface. GA modeled the material around the cylinder as iron with a thickness of 0.125 inches (0.3175 cm). GA modeled the pipe as being 30 ft (9.144 m) long with a radius of 1.4 inches (3.556 cm). GA used a total Cs-137 activity in the pipe of 19.01 µCi (see Section 3.6.2 above). The licensee reported that the results of this MicroShield run was 2.94×10^{-3} mR/hr and that the effective dose rate based on an anterior/posterior geometry, with buildup, was 4.05×10^{-3} mrem/hr. The licensee calculated the total annual dose based on assumed exposure times of 40 hrs/yr and 1,040 hrs/yr, which resulted in projected doses of 0.162 mrem/yr and 4.21 mrem/yr respectively (Table 1).

Table 1: Buried Pipe Dose Assessment Results

Scenario	Dose (mrem/yr)	Dose (mSv/yr)
Buried Pipe at Depth	1.854×10^{-6}	1.854×10^{-8}
Buried Pipe in Surface Soil	6.672	6.672×10^{-2}
External Exposure to Worker (1,040 hrs/yr)	4.21	4.21×10^{-2}
External Exposure to Worker (40 hrs/yr)	0.162	1.62×10^{-3}

The NRC staff reviewed the dose assessment calculations performed by GA and concluded that the input parameters assumed for the calculations are appropriate because they are consistent with the configuration and activity of the residual radioactivity. The NRC staff also performed independent RESRAD and MicroShield calculations and obtained results comparable to the licensee's results. The staff's independent calculations included sensitivity analyses to determine the dose consequences if the concentrations of Cs-137 and Sr-90 were higher than assumed. The NRC staff calculated the projected dose using concentrations of Cs-137 and Sr-90 determined using the maximum exposure rate and maximum scaling factor for Sr-90. The projected dose in all cases was less than 25 mrem/yr. For these reasons, the NRC staff concludes that the dose assessments performed by GA were adequate to predict dose rates from the buried pipe.

3.6.5 Conclusions for Buried Pipe Dose Assessment

The NRC staff has reviewed the dose modeling analyses for the residual radioactivity in the buried pipe remaining under the floor of the GA TRIGA Mark F Reactor building using the Consolidated Decommissioning Guidance (NUREG-1757), Volume 2, Section 5.2 (Unrestricted Release Using Site-Specific Information). GA submitted this dose modeling analysis as part of its FSSR for this site to demonstrate that the final site conditions meet the unrestricted release criteria in 10 CFR 20.1402.

For the reasons described above, the NRC staff concludes that the dose modeling completed for the buried pipe is reasonable to predict dose rates for the exposure scenarios under consideration. In addition, the dose estimate provides reasonable assurance that the dose to the average member of the critical group is not likely to exceed the 0.25 mSv (25 mrem) annual dose criterion in 10 CFR 20.1402. This conclusion is based on the modeling effort performed by the licensee and the independent analysis performed by the staff. The NRC staff further concludes that in determining the dose, the licensee has used a combination of the conceptual model, exposure scenario, mathematical model, and input parameters to calculate a reasonable estimate of dose and that the licensee has adequately considered the uncertainties inherent in the modeling analysis. Because the limiting (most conservative) exposure scenario is buried pipe in surface soil, and this scenario results in an estimated maximum dose of 6.6 mrem/year, which is well below the 25 mrem/year release criteria, the NRC staff concludes that the GA buried pipe dose assessment provides reasonable assurance that the residual radioactivity in the pipe is consistent with unrestricted release.

3.8 Final Dose Conclusions

The NRC staff evaluated the licensee's FSSR to examine the licensee's assessment of the doses resulting from exposure to residual radioactivity remaining at the GA site now that the decommissioning process is complete. This review was conducted in accordance with the regulatory guidance and acceptance criteria contained in NUREG-1757, Volume 2, Revision 1, and NUREG-1700, Revision 2. Based on the discussion provided in this safety evaluation, the NRC staff finds that the GA FSSR provides reasonable assurance that the licensee performed adequate surveys to demonstrate compliance with the criteria for unrestricted use, as specified in 10 CFR 20.1402.

The licensee stated that the site release criteria for the GA site correspond to the 10 CFR 20.1402 criteria for unrestricted use. The residual radioactivity, including that from groundwater sources, that is distinguishable from background, must not cause the TEDE to an average member of the critical group to exceed 25 mrem/yr. The residual radioactivity must also be reduced to levels that are ALARA.

As described above, the licensee stated that the FSSR sections that support this release conclusion demonstrate that the radiological surveys were conducted in a manner consistent with the GA FSS Plan and that the survey units passed the FSS. As described in Section 3.5, with the exception of the survey unit containing the buried pipe, all the results for survey unit radiological assessment were well below the screening values equivalent to 25 mrem/yr in NUREG-1757, Volume 2, Appendix H, Table H.2, as well as the more restrictive screening value for Eu-152 from the NRC/EPA MOU. Additionally, the maximum dose the licensee calculated for the buried pipe was 6.672 mrem/yr for the scenarios analyzed. The maximum SOF for the soil in the vicinity of the pipe was 0.387, which would correspond to a dose of 9.68 mrem/yr. So, even if an individual were to receive the maximum dose from both the pipe and soil, the combined dose would still be less than 25 mrem/yr.

Therefore, the NRC staff concludes that the GA TRIGA facility, as described by License No. R-38 and License No. R-67, meets the radiological criteria for unrestricted release and may be removed from the requirements for future NRC oversight.

4.0 NRC INSPECTIONS AND CONFIRMATORY SURVEYS

NRC inspectors and survey contractors from the Oak Ridge Institute for Science and Education (ORISE) performed multiple inspection activities, as well as in-process and confirmatory surveys of the radiological conditions at the GA facility, throughout the decommissioning process. ORISE also performed confirmatory analysis of samples collected from the site. These samples included both volumetric and swipe (removable contamination) samples. The samples were tested for Cobalt-60, Cesium-137, Strontium-90, gross alpha and gross beta activity, and other gamma-emitting radionuclides associated with the GA TRIGA reactors.

NRC inspectors reviewed the licensee’s survey results, survey methodology, and plans for demonstrating that the survey results would confirm that the remaining structures and areas at GA met the acceptable radiological levels for unrestricted release. The NRC staff also reviewed confirmatory and in-process radiation and contamination surveys conducted by ORISE. Confirmatory surveys provide confidence that the licensee’s survey results are representative of the conditions for that survey unit. In-process surveys provide confidence that the licensee’s survey results are accurate. Based on the data review, discussions, and observations, the NRC inspectors observed that the licensee had in place methods for demonstrating compliance with the unrestricted release criteria. The NRC inspectors also found that the licensee had in place a methodology in which the survey results were used to assess the radiological condition of the onsite structures in order to determine whether further remediation was required, or the structures were suitable for demolition. Based on the results of the ORISE survey, the licensee conducted additional remediation of certain areas within the facility. An NRC inspector conducted a follow up inspection to verify that the licensee had effectively remediated these areas.

In addition to the independent in-process surveys, the NRC inspectors also split several samples with the licensee to assess the licensee's capability to characterize various areas and structures of the site. The samples were analyzed by the NRC's contract laboratory, the U.S. Department of Energy's Radiological and Environmental Sciences Laboratory. The results indicated that the licensee correctly characterized the decommissioning systems and structures at GA. In summary, NRC inspections and ORISE confirmatory surveys corroborated that the radiological conditions of the portion of the GA site proposed to be released met the approved site-specific DCGLs, and that GA's laboratory data were consistent and in agreement with the ORISE analytical results.

Throughout the decommissioning process, inspectors from the NRC's Region IV office in Arlington, Texas, conducted routine safety inspections at the GA TRIGA Reactor Facility, as documented in the following NRC Inspection Reports (IRs), which took place during and after removal of the TRIGA irradiated fuel elements in 2010: IR 50-163/2010-01; 50 89/2010-01 (ADAMS Accession No. ML103060034), IR 50 163/2012-01; 50 89/2012 01 (ADAMS Accession No. ML12321A127), IR 50 163/2013-01; 50 89/2013-01 (ADAMS Accession No. ML13338A864), IR 50 163/2015-01; 50 89/2015-01 (ADAMS Accession No. ML15328A527), IR 50 163/2018-01; 50 89/2018-01 (ADAMS Accession No ML18319A137), IR 50 163/2019-01; 50 89/2019-01 (ADAMS Accession No. ML19247C512), and IR 50 163/2020-01; 50 89/2020-01 (ADAMS Accession No. ML20090B701).

The inspections consisted of observations by the NRC inspectors, interviews with GA and contractor personnel, confirmatory surface measurements, collection of soil samples, and a review of decommissioning work plans and work instructions. The NRC inspections also verified that radioactive waste associated with the decommissioning project had been appropriately shipped offsite and that the decommissioning and FSS activities were being conducted safely and in accordance with the appropriate regulatory requirements, licensee commitments, and the NRC-approved GA DP. No health or safety concerns were identified during the NRC inspections.

During the period of August 5-8, 2019, ORISE performed confirmatory surveys in support of the GA FSS and decommissioning activities, which included gamma surface scans, gamma direct measurements, alpha-plus-beta scans, alpha-plus-beta direct measurements, smear sampling, and soil/volumetric sampling within Building G21 and associated land areas, as applicable. The areas investigated included the following survey units: Mark I reactor pit, Mark F reactor pit and canal, Mark I reactor room (floor and lower walls), Mark F reactor room (floors and lower walls), the soil lab, mezzanine 1, mezzanine 2, TRIGA waste yard, TRIGA front yard, TRIGA back yard, and room 112, as well as a small section of the TRIGA Reactor Facility roof. ORISE provided the results of the confirmatory survey in a report dated November 26, 2019 (ADAMS Accession No. ML19337D382). The ORISE survey data indicated that additional remediation was necessary in selected areas. The licensee subsequently remediated these areas. During the March 2020 inspection (ADAMS Accession No. ML20090B701), the NRC inspector confirmed that the licensee had effectively remediated all areas except the excavated pit in the Mark F reactor room. The licensee completed the remediation of the excavated pit by July 2020. The licensee submitted its final survey results for the excavated pit to the NRC by letter dated December 14, 2020 (ADAMS Accession Nos. ML21012A273, ML21012A274, ML21012A275, and ML21012A276). The results of the NRC confirmatory surveys and the results of the licensee's final survey support the conclusion that the residual radioactivity levels satisfy the criteria for license termination set forth in the GA licenses, and as established in the previously submitted DP and FSS Plan.

Based on observations during the NRC inspections and ORISE confirmatory survey activities, decommissioning activities have been carried out by GA in accordance with the approved TRIGA Reactor Facility DP. Additionally, the NRC staff evaluated the licensee's FSSR, and the results of the independent confirmatory survey conducted by ORISE. Based on the NRC staff's evaluation of the GA FSSR sampling and scanning data, NRC staff inspections, ORISE confirmatory analyses, and comparison to the TRIGA Reactor Facility DP and FSS Plan criteria, the NRC staff concludes that the GA TRIGA Reactor Facility decommissioning has been performed and completed in accordance with the approved DP, and that the facility and site are suitable for unrestricted release in accordance with the radiological criteria for license termination in 10 CFR part 20, subpart E, "Radiological Criteria for License Termination."

5.0 EVALUATION OF THE NEED FOR NRC/EPA LEVEL 2 CONSULTATION

5.1 Background

The NRC and EPA entered into a MOU for "Consultation and Finality on Decommissioning and Decontamination of Contaminated Sites" on October 9, 2002. The MOU provides that, unless an NRC-licensed site exceeds any of three trigger criteria contained in the MOU, the EPA agrees to a policy of deferral to the NRC for decision making on decommissioning, without the need for consultation.

For sites that trigger the criteria in the MOU, the NRC will consult with the EPA at two points in the decommissioning process: (1) prior to NRC approval of the licensee's license termination plan or DP, which the NRC terms Level 1 consultation; and (2) following completion of the FSS, which the NRC terms Level 2 consultation.

5.2 Evaluation

The NRC reviewed and approved the GA DP in 1999, before the MOU was in effect. Although the GA TRIGA reactors were not evaluated for Level 1 consultation, the NRC subsequently evaluated the DCGLs contained in the FSSR to determine whether they would have triggered the need for consultation. Based on this review, the NRC staff concluded that a Level 1 consultation would not have been needed because the approved DCGLs are below the Level 1 consultation trigger soil concentrations for sites that intend to pursue industrial/commercial use.

The NRC staff reviewed the data in the GA FSSR and compared the residual radioactivity levels to the trigger values for soil in Table 1 of the EPA MOU related to the residential and industrial use scenarios. Table 1 states that, except for Radium-226, Thorium-232, or total uranium, soil concentrations should be aggregated using a SOF approach to determine the site-specific consultation trigger concentrations. Consistent with the MOU, the residual radioactive material concentrations for Cobalt-60, Cesium-137, Europium-152, and Europium-154 (as determined from the sample analyses) were aggregated using the SOF approach to determine the site-specific consultation trigger values for each of the 17 GA survey units. The NRC staff determined that none of the "as left" survey unit average concentrations exceed the SOF trigger value when compared to the Table 1 values for industrial use in the EPA MOU. After evaluating this information, the NRC determined that a Level 2 consultation with the EPA under the MOU is not required.

6.0 CONCLUSIONS

The requirements at 10 CFR 50.82(b)(6) establish the criteria to be used by the NRC for terminating the license of a non-power reactor facility that has an approved DP. These criteria include: (1) the decommissioning has been performed in accordance with the approved DP, and (2) the final radiation survey and associated documentation demonstrate that the facility and site have met the criteria for decommissioning in 10 CFR Part 20, Subpart E.

The NRC has concluded that all decommissioning and dismantlement activities have been completed in the survey units to be released from the GA licenses. The FSS activities have confirmed that the residual radioactivity in each of the survey units meets the criteria established in the GA DP. The NRC's review of the licensee's DP determined that the proposed DCGLs would ensure that the 10 CFR Part 20, Subpart E, release criteria would be met. The NRC's subsequent review of the GA FSSR determined that the final survey reports were consistent, and demonstrated compliance, with the GA DP. Therefore, the FSS results demonstrate that the survey areas to be released meet the radiological criteria for unrestricted release. As described above, the dose remaining in the survey units is less than the NRC unrestricted release criteria of 25 mrem/yr and is therefore acceptable.

The NRC also concluded that a Level 2 consultation with the EPA was not required based on the as-left residual radioactivity concentrations. In addition, multiple NRC inspections and confirmatory measurements substantiated that the licensee's decommissioning and FSS programs adequately assessed the radiological conditions at the site. Therefore, the NRC approves the termination of NRC License Nos. R-38 and R-67 for the GA TRIGA facility and the release of the site for unrestricted use, as specified in GA's December 14, 2020, request.

Principal Contributors: Marlayna Doell
Stephen Giebel
Karen Pinkston
Kim Conway

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

Docket Nos. 50-089 and 50-163

AMENDMENT TO INDEMNITY AGREEMENT NO. B-9
AMENDMENT NO. 33

Effective______________________, 2021, Indemnity Agreement No. B-9, between General Atomic Company, and the US Nuclear Regulatory Commission, dated April 20, 1962, as amended, is hereby terminated.

FOR THE UNITED STATES NUCLEAR REGULATORY COMMISSION

Frederick Miller Digitally signed by Frederick Miller
Date: 2021.12.02 11:35:33 -05'00'

Fred R. Miller, Chief
Financial Assessment Branch
Division of Rulemaking, Environmental, and Financial Support
Office of Nuclear Material Safety and Safeguards

Accepted_____________________________, 2021

By________________________________ GENERAL ATOMIC COMPANY

Enclosure 2