FORT CALHOUN STATION DECOMMISSIONING PROJECT
LICENSE TERMINATION PLAN

CHAPTER 1
GENERAL INFORMATION

REVISION 0

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ABREVIATIONS

AEC	Atomic Energy Commission
ADAMS	Agencywide Documents Access and Management Systems
ALARA	as low as is reasonably achievable
AMCG	average member of the critical group
AMSL	above mean sea level
bgs	below ground surface
CE	Combustion Engineering
CNW	Chicago and Northwestern
CCP	calculations and position paper
DA	deconstruction area
DCGL	derived concentration guideline level
DQO	data quality objectives
DSAR	Defueled Safety Analysis Report
EIS	environmental impact statement
FCS	Fort Calhoun Station
FSS	final status survey
gpm	gallons per minute
GTCC	greater than class C
HSA	historical site assessment
ISFSI	Independent Spent Fuel Storage Installation
LTP	license termination plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
OCA	Owner Controlled Area
ODCM	Off-Site Dose Calculation Manual
OPPD	Omaha Public Power District
PWR	pressurized water reactor
RESRAD	RESidual RADioactive
RO	reverse osmosis
TEDE	total effective dose equivalent

1 GENERAL INFORMATION

Omaha Public Power District (OPPD) is submitting this license termination plan (LTP) for Fort Calhoun Station (FCS). The following provides the licensee name, address, license number, and docket number for FCS.

Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102
License No. DPR-40
Docket No. 50-285

The name and address of the facility is:

Fort Calhoun Station
9610 Power Lane
Blair, NE 68008

The spent nuclear fuel will be stored in the Independent Spent Fuel Storage Installation (ISFSI) which will be maintained under a 10 CFR Part 72 license, Certificate of Compliance No. 72-1004, and Docket No. 72-054.

1.1 Purpose

The objective of decommissioning FCS is to reduce the level of residual radioactivity to levels that permit the release of the site for unrestricted use and allow for the termination of the 10 CFR Part 50 license, excluding the ISFSI area. This LTP satisfies the requirement of 10 CFR 50.82(a)(9) to submit an LTP for U.S. Nuclear Regulatory Commission (NRC) approval. This LTP was written following the guidance in Regulatory Guide 1.179, “Standard Format and Contents for License Termination Plans for Nuclear Power Reactors” [1], NUREG-1700, “Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans” [2], NUREG-1575, “Multi-Agency Radiation Survey and Site Investigation Manual” (MARSSIM) [3], and NUREG-1757, Volume 2, “Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria” [4]. To satisfy the requirements of 10 CFR 50.82(a)(10), this LTP is accompanied by a proposed license amendment that establishes the criteria for when changes to the LTP require NRC approval.

The LTP describes the decommissioning activities that will be performed, the process for performing the final status surveys (FSS), and the method for performing dose assessments and as low as reasonably achievable (ALARA) evaluations to demonstrate that the FCS site meets the criteria for unrestricted use. The LTP contains specific information on the following:

- the Historical Site Assessment (HSA) and site characterization to ensure that FSS covers all areas where contamination existed, remains, or has the potential to exist or remain;
- identification of remaining dismantlement activities;
- plans for site remediation;
- detailed plans for the final radiation survey;

- compliance with the radiological criteria for license termination;
- an updated site-specific estimate of remaining decommissioning costs;
- a supplement to the environmental impact statement (EIS) describing any new information or significant environmental change associated with proposed termination activities; and
- identification of parts, if any, of the FCS site that were released for use before approval of the LTP.

Section 1.5 discusses the purpose and content of each LTP chapter. Section 1.6 discusses the process for making changes to the LTP.

1.2 Decommissioning Objective

The decommissioning objective is to conduct remediation and survey operations such that OPPD can submit a request to the NRC for the release of the site (other than the remaining licensed ISFSI facility) after meeting the unrestricted release requirements of 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use." The LTP documents the process that will be used to demonstrate that the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are ALARA.

1.3 Historical Background and Site Description

1.3.1 Historical Background

FCS Unit 1 was a Combustion Engineering (CE) 2-loop pressurized water reactor (PWR) rated at nominal 533.7 megawatts electrical. The station is comprised of a CE pressurized water reactor with supporting facilities. The primary coolant system consisted of two heat transfer loops. Each loop contained one steam generator and two reactor coolant pumps with associated piping and valves. In addition, the primary coolant system included a pressurizer, pressurizer relief tank, interconnecting piping, and the instrumentation necessary for operational control. All major components of the primary coolant system are located within the Containment Building.

Plant construction began in 1968. The first fuel assembly was loaded into the reactor May 24, 1973. The final fuel assembly was loaded on June 8, 1973 and the final core verification was completed on June 9, 1973. The NRC issued an operating license on August 9, 1973. The plant officially went online on September 1, 1973, with commercial operation beginning on September 26, 1973. On June 24, 2016, and updated on August 25, 2016, OPPD submitted the Certifications of Permanent Cessation of Power Operations in accordance with 10 CFR Part 50.82(a)(i). The plant went offline on October 24, 2016.

OPPD began actively decommissioning FCS on April 29, 2019, and completed the transfer of all spent nuclear fuel to the ISFSI in May 2020. The ISFSI is located in the protected area designated for dry storage of spent fuel and greater than class C (GTCC) waste. The ISFSI is licensed under Subpart K of 10 CFR Part 72.

1.3.2 Site Description

The FCS site is located on the west bank of the Missouri River at river mile 646.0 on approximately 540 acres, approximately 19.4 miles north of Omaha, Nebraska. Figure 1-1 shows the geographical location of the site relative to nearby towns, cities, and the river. OPPD has a perpetual easement on approximately 117 acres of land primarily on the east bank of the river directly opposite the plant buildings.

In June of 2018, OPPD requested approval from the NRC to remove approximately 120 acres of land on the northwest portion of the Owner-Controlled Area (OCA) from the Part 50 License (Agencywide Documents Access and Management Systems (ADAMS) Accension No. ML 18215A187). In November of 2018, OPPD submitted a request for partial site release of the 475-acre property in Iowa that was subject to a perpetual easement (ADAMS Accession No. ML 18316A036), and the NRC approved the releases in April of 2019 (ADAMS Accession No. ML 19074A301). Figure 1-2 shows the boundaries of the OCA as well as the areas that were subjected to partial site release.

On the southern part of the site, the ground rises sharply about 60 feet to a higher, level area which is bounded on the south by U.S. Highway 75, formerly U.S. Highway 73. Figure 1-3 shows the FCS site layout.

A majority of the site is being farmed at the present time and it is planned that farming will continue. On-site farming consists primarily of grain, with approximately 354 acres under cultivation. The area adjoining the site is farmland and sparsely populated. Cargill Industries property borders the FCS site to the northwest. The minimum exclusion distance is at a 1,525 feet radius circle centered on the Auxiliary Building stack. The nearest privately owned land is farmland and is approximately 0.5 miles from the site. The nearest population center area of more than 25,000 is formed by adjacent cities of Omaha, Nebraska and Council Bluffs, Iowa.

A rail spur from the Chicago and Northwestern (CNW) Railway was constructed to serve the construction of FCS. The original CNW tracks and rail spur have since been removed. In 1994, a permanent easement was granted to allow the construction of a new rail spur in the approximate location of the old CNW railway to allow trains to serve the Cargill industrial facility located north of FCS. In late 2020, OPPD added three lines (Lines A, B, and C approximately 1,360, 1,595, and 2,230 feet in length, respectively) to the rail spur to allow for the direct loading of waste into rail cars. A 100 feet by 400 feet waste processing structure has been erected at the rail spur to allow for the size reduction and loading of radioactive waste in a controlled environment.

1.3.3 Topography

As stated in the "Defueled Safety Analysis Report" (DSAR) [5], the FCS site is situated within parts of Sections 16, 17, 20 and 21, Township 18 North, Range 12 East of Washington County, Nebraska in the Modale quadrangle. The site is part of the Missouri River bottomland, which is a nearly level plain about 15 miles wide at Blair, Nebraska, 8 miles wide at the site and narrowing to 3 miles wide in the vicinity of Omaha-Council Bluffs. The elevation of this plain averages about 1,000 feet above mean sea level (AMSL) at the site.

The surface of the land, starting from the Missouri River at about elevation 997 feet AMSL, falls to an old channel of the river before rising again to approximately 1,004 feet. Beyond this point, the land then gradually falls off to about 1,000 feet, rises again to approximately 1,020 feet, and then rises approximately 60 feet to a higher plateau at elevation 1,080 feet. Figure 1-4 shows the FCS site topography.

The Missouri River, which flows generally north to south, forms the northeast to southeast site boundary. This part of the river is referred to by the Corps of Engineers as the Blair Bend. The river limits are under control of the Corps of Engineers who have established a structure azimuth line which acts as another site boundary.

1.3.4 Meteorology and Climatology

Nebraska is located midway between two distinctive climate zones, the humid east and the dry west. Cyclic weather conditions representative of either zone, or combinations of both, occur. Changes in weather result from the invasion of large masses of air with dissimilar properties. These air masses tend to get their characteristics from either the warm and humid south-southeast, the warm and dry southwest, the cool and dry north-northwest, or the cold continental polar air of the north. The region is also affected by many storms or cyclones (areas of low pressure) which travel across the country, generally from west to east. Periodic and rapid changes in the weather are normal, especially in the winter.

Annual average precipitation for the region is about 29.9 inches, but annual amounts vary widely from year to year. About 75 percent of the precipitation occurs during showers and thunderstorms from April through September. Snowfall amounts to about 30 inches of snow as the annual average, but total annual amounts vary widely from year to year.

The surface wind direction and speed are quite varied during all seasons of the year. The prevailing wind direction from May through December is from south-southeast; north-northwesterly winds prevail during the remainder of the year. The annual wind speed is 8.5 miles per hour based on the mid-point of the range reported in the DSAR.

According to the Limited Site Non-Radiological Characterization Survey Report [6] the mean annual temperature for the region is 51.1 degrees Fahrenheit. The January monthly mean is 20.2 degrees Fahrenheit, while that for July is 77.7 degrees Fahrenheit. Relative humidity ranges from an average of about 78 percent for the period midnight to noon and about 59 percent from the period noon to midnight. The mean percentage of possible sunshine over the area is about 50 percent in winter and about 75 percent in summer.

1.3.5 Geology and Seismology

The soils below FCS include thick beds of limestone, dolomite, shale, sandstone, and thin layers of coal beds. The deeper formations were deposited in marine depositional environments with the shallow soils from the lateral migration of the paleo river channel. The major tectonic features of the mid-continent region began to develop late in the Paleozoic Era, and probably most of the important structural features of the Nebraska Iowa Missouri River Valley area had already developed or were developing by the end of Permian period. However, there is no record of movement of the fault in historic times, or any indication of activity in recent geologic time.

At the beginning of the Pleistocene period, the Missouri River Valley and its main tributaries were established in their approximate present positions. Subsequently, under successive glacial movements, the valleys were filled and re-opened several times. During this period, the peorian loess was deposited on the terraces and adjacent uplands. It is probable that only the upper part of the alluvium in the Missouri River Valley is actually of recent age and that deeper deposits are mostly of Pleistocene age.

According to the [5] and [6], unconsolidated sediments at the plant site generally range from 65 to 75 feet in thickness. The soils are typically interstratified and cross-bedded. These soils may be grouped generally into two units:

- an upper fine-grained sandy clay with silt approximately 20 to 50 feet thick
- an underlying carbonate bedrock surface at a depth of approximately 65 to 75 feet below ground surface (bgs).

The upper units were representing former river deposits and are not likely continuous, but rather have preferential channels formed by paleo-oxbow deposits.

Pennsylvanian-aged limestone and shale (bedrock) of the Kansas City Formation are encountered below the overburden soils. The bedrock below the site consists of various types of limestone formations.

1.3.6 Hydrology

According to EA12-010, "Post 2011 Flood Geotechnical Assessment" [7], groundwater at the site is in hydraulic communication with the adjacent Missouri River, with the water table ranging from 2 to 20 feet bgs depending on the river stage. However, according to the "Review of the Groundwater Protection Program at the Fort Calhoun Nuclear Station" [8], under typical conditions, the depth to groundwater is approximately 15 to 20 feet bgs. Both soil units identified in Section 1.3.5 are water bearing with the deeper unit exhibiting a higher hydraulic conductivity. The hydraulic gradients below the site are relatively flat with relatively slow groundwater velocity.

Water levels taken at the site show that the groundwater gradients are nearly flat, with only a gentle slope toward the river, about 10 feet below the ground surface. Water levels at the site varied from elevations 993.7 to 992.4 feet, while the river levels recorded during this same period ranged from elevations 993.2 to 992.4 feet. Groundwater levels vary with changes in the river level. The rate of groundwater flow in the alluvial soils varies with the permeability; however, the groundwater flow rate, or velocity, is very slow due to the low gradients. The coefficient of permeability varied from about one-half to three feet per day in the upper sandy silt and silty sand. In the lower fine-to-coarse sands and gravels, coefficients of permeability as high as 20 feet per day were measured.

According to the DSAR, groundwater flow directions have been reported to be both toward the Missouri River (northeasterly) and away from the Missouri River (south-southwesterly). Flow directions towards the river appear to represent times when Missouri River levels are relatively low (e.g., during the spring, summer, and early fall, when most precipitation occurs, and the river flow is relatively high). Flow directions away from the river appear to represent times when

Missouri River levels are relatively high, causing bank storage effects (e.g., during late fall and winter when the river recedes).

Locally, next to the plant structures, a reverse osmosis (RO) water treatment plant withdrew groundwater, to be used at the plant. This groundwater withdrawal caused a cone of depression and altered groundwater flow. The extraction well was located at the northwest corner of the old warehouse slab. Testing during the well installation documented that the aquifer may produce approximately 500 gallons per minute (gpm). The production well was in service as of August 2007, continuously pumping about 200 gpm and was subsequently removed from service in 2018. Measurements of groundwater flow indicated that within 600 feet of the RO well, groundwater flow was toward the RO well (i.e., southwest).

A search for active wells was conducted of nearby properties and concentrated in the flood plain approximately 2.5 miles upstream and 4.5 miles downstream from the FCS site. A review of the well databases revealed that there are 31 domestic/private wells, seven geothermal heating wells, 13 irrigation wells, six public water wells (for recreation areas), 27 environmental monitoring wells, and four commercial/industrial wells, within the search area (Haley & Aldrich, Inc. memorandum to Energy*Solutions*, April 13, 2021) [9].

1.3.7 Surface Waters

The FCS site is bounded on the north by a portion of the Blair Bend of the Missouri River. The Corps of Engineers maintains river structures to prevent further meandering of the channel within the alluvial flood plain; the structures take the form of pile dikes and revetments. There are six dams upstream that control the river flow. The nearest structure to the site is at Gavin's Point, and the most distant structure is Fort Peck. There are no dams, locks, or similar structures on the Missouri River downstream of the plant site.

Fish Creek is an intermittent drainage stream that runs along the west side of the ISFSI and switchyard. This stream discharges into a larger wetland, before flowing in the Missouri River.

Two sanitary lagoons are located in the east portion of the site. An application area is located to the south of the lagoons for disposal of lagoon water. Although detectable Cs-137 has been found in the sludge from these lagoons, licensed radioactivity has not been detected in the lagoon water. The FCS site surface water features are shown on Figure 1-5.

The site has been flooded several times, with the most recent events occurring in 2019. The most impactful flooding occurred in 2011, where the river overflowed its banks for several months. The flood stage from that event and its impacts to the site are shown on Figure 1-6.

In 2020, OPPD installed a HESCO barrier around the site to mitigate flooding during decommissioning. The barriers are sand and gravel filled fabric-lined metal cages that range from 5 to 10 feet in height. Approximately 8,000 linear feet of barrier was installed at the locations identified on Figure 1-7. The portion of the barrier that runs between the Intake Structure and Turbine Building will be erected in the future if necessary. The HESCO barrier will likely be dismantled upon the completion of decommissioning activities.

1.3.8 Environs and Natural Resources

The plant site is located on the alluvial plain of the Missouri River in a predominantly agricultural region roughly 19.4 miles north of the Omaha metropolitan area.

There are no residences within one half mile of the location. The seven nearest residences are from 3,000 to 4,000 feet distant. These are located generally along Highway 75, the southern boundary of the site. There are no schools, hospitals, prisons, motels, or hotels in the immediate vicinity of the site. An industrial park is located northwest of the plant property. Industries include a large corn processing facility, agricultural fertilizer storage facilities, and various other light industrial plants.

The DeSoto National Wildlife Refuge occupies approximately 7,821 acres east of the plant site. This area is open to the public for day use year-round. Visitors to the refuge generally use areas from two to five miles from the plant. Estimates by the U.S. Fish and Wildlife Service place annual usage of the facility at approximately 120,000 for the Visitors Center and 400,000 for the refuge. The expected maximum daily usage of the facility has been placed at 2,500 visitors for a winter weekday and 5,000 on a summer weekend. The Boyer Chute Federal Recreation Area is a day use facility occupying approximately 2,000 acres southeast of the plant site. Visitors to the recreation area generally use areas seven to ten miles from the plant. The estimates for annual usage of this facility are approximately 50,000 visitors.

The State of Nebraska operates the Fort Atkinson State Historic Park five and one-half miles southeast of the plant site. This day use facility is mostly seasonal and estimates place annual usage at 60,000. The State of Iowa maintains Wilson Island State Park with 42 camping spaces south of the DeSoto National Wildlife Refuge and four miles southeast of the plant site. The estimates for usage of this facility range from 500 on a winter weekend to 1,000 on a summer weekend.

Two private facilities lie to the north of the plant along the Missouri River. The Cottonwood Cove Marina & RV Resort is located approximately four and one-half miles from the plant. Estimates place summer weekend usage at 200 people. River View Park Resort & Marina is a private campground lying directly to the south of Cottonwood Marina and ranging from four to four and one-half miles from the plant. The campground has approximately 282 campsites and is open from April to October.

The nearest municipality is the city of Blair, Nebraska, about three miles northwest, with a population of 7,990 per the 2010 census.

Fort Calhoun, Nebraska, is about five miles southeast of the facility. The 2010 census reported a population of 908 in Fort Calhoun and 167 in Kennard Village, about seven miles from the plant site. The 2010 population of Washington County is 20,234.

Missouri Valley, Iowa, about 11 miles east, has a 2010 population of 2,838 as compared to the 2000 population of 2,982. Harrison County, Iowa is located east of Blair, Nebraska and has a population of 14,928 according to the 2010 census.

The Omaha metropolitan area includes the cities of Omaha, Nebraska, and Council Bluffs, Iowa, and the adjoining areas of Douglas, Washington, and Sarpy Counties, Nebraska, and

Pottawattamie County, Iowa. The area lies 10 to 25 miles south of the site, with the main concentration of population beyond the 15-mile radius.

The 2010 U.S. Census data shows an increase in population in the Omaha metropolitan area and in most of the nearby cities but a decrease in the rural and farm population. While it is probable that the area around the plant site outside of the Omaha metropolitan area will remain largely agricultural and that the population will increase slowly, a general decline of the rural population will continue, reflecting the movement of people into towns and cities. The expansion of the Omaha metropolitan area has been generally south and westward, coinciding with the interstate highway. It is expected that future growth of the metropolitan area will continue south and west and also northwestward. Thus, it is probable that the area surrounding the plant site will continue to remain largely agricultural.

1.4 Operational Background

The NRC issued an operating license on August 9, 1973. The plant began commercial operation on September 26, 1973, initially at a maximum power level of 1,420 megawatts thermal, corresponding to an electrical output of 481 megawatts electrical. A license amendment was granted, permitting maximum reactor output to not more than 1,500 megawatts thermal.

Other milestones related to the operational history of FCS are as follows:

- On May 15, 1984, a steam generator tube rupture due to intergranular stress corrosion cracking occurred during the hydrostatic pressure testing of the primary coolant system.
- In 1993, 2011, and again in 2019, the site experienced flooding events as a result of historic high Missouri River levels.
- On November 4, 2003, the NRC granted an operating license extension for an additional 20 years, to 2033.
- In the spring of 2011, flooding of the site caused an extended outage of the reactor.
- In December 2013, the reactor returned to operation after recovery from the flood of 2011.
- On June 24, 2016, and updated on August 25, 2016, OPPD submitted the Certifications of Permanent Cessation of Power Operations in accordance with 10 CFR Part 50.82(a)(i).
- On October 24, 2016, the reactor was shut down for the final time, thus initiating the decommissioning process.
- On November 13, 2016, the final reactor fuel off-load was completed.
- On October 11, 2018, the OPPD Board authorized the move toward decommissioning FCS in accordance with the DECON alternative.
- On May 13, 2020, transfer of the spent nuclear fuel to the ISFSI was completed.

1.5 License Termination Plan Summary

1.5.1 General Information

The FCS LTP describes the process used to meet the requirements for terminating the 10 CFR Part 50 license and to release the site for unrestricted use. The LTP has been prepared in accordance with the requirements in 10 CFR 50.82(a)(9) and is submitted as a supplement to the DSAR. The LTP submittal is accompanied by a proposed license amendment that establishes the criteria for when changes to the LTP require prior NRC approval.

The subsections below provide a brief summary of the seven chapters of the LTP that address the requirements in 10 CFR 50.82(a)(9)(ii) as well as compliance with the radiological criteria for license termination.

1.5.2 Site Characterization

LTP Chapter 2 discusses the site characterization that has been conducted to determine the extent and range of radioactive contamination on site prior to remediation, to include structures (including systems and embedded piping) or portions of structures (e.g., basements), soils, and groundwater. The results from the initial radiological characterization are presented in OPPD Calculations and Position Paper (CPP) FC-20-012, “Fort Calhoun Station Decommissioning Project Radiological Characterization Report” [10]. The LTP will identify the instrumentation and supporting quality assurance practices used during characterization, and will identify the background levels used. Based on the results of the site characterization, OPPD will plan remediation and FSS in areas determined to be impacted by the operation of FCS.

The “Historical Site Assessment for Fort Calhoun Station” (HSA) [11] provided the initial foundation for site characterization and the basis for dividing the site into survey units. The survey units were selected and classified using the criteria specified in MARSSIM.

Some areas of the site were inaccessible during the initial site characterization (e.g., reactor cavity, spent fuel pool, holdup tanks rooms, embedded drain lines). These locations will be characterized as they become accessible. Data from subsequent characterization will be presented in a supplemental report and may be used to change the original classification of a survey unit, within the requirements of this LTP, up to the time of FSS, as long as the classification reflects the level of residual activity existing prior to any remediation in the survey unit.

LTP Chapter 2 contains a summary of information contained in [10] and [11]. Additional characterization information and confirmation will continue throughout the decommissioning as part of the FSS process. The LTP will generally not be updated to include this additional characterization.

1.5.3 Identification of Remaining Site Dismantlement Activities

LTP Chapter 3 identifies the remaining site dismantlement and decontamination activities. The information provided in Chapter 3 includes:

- a description of the areas and equipment that need further remediation,

- a description of completed and ongoing decommissioning activities,
- a description of future decommissioning activities and tasks,
- a summary of radiological conditions that may be encountered,
- radiation exposure projections for decommissioning,
- an estimate of the types and quantities of radioactive material generated for release and disposal,
- descriptions of proposed control mechanisms to ensure areas are not re-contaminated, and
- general project milestones.

OPPD is decommissioning FCS in accordance with the DECON alternative described in NUREG-0586, Supplement 1, Volume 1, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" [12]. The decommissioning activities will be conducted in accordance with the OPPD Health and Safety Program, Radiation Protection Program, Radioactive Waste Program, Process Control Program, Off-Site Dose Calculation Manual (ODCM), and plant administrative, work control, and decommissioning implementation procedures. These are established programs that are routinely inspected by the NRC.

Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during plant operations. The radiological risk associated with decommissioning activities is bounded by previously analyzed radiological risk for former operating activities that occurred during major maintenance and outage activities.

The primary control mechanism used to ensure areas are not cross-contaminated during decommissioning operations is the use of dedicated waste haul routes for radioactive and non-radioactive waste, as shown in Figure 1-8. A large temporary enclosure will be erected outside the Containment Building equipment hatch where radioactive waste will be loaded directly into trucks/containers. The trucks will then transport the waste to a waste load-out enclosure located at the rail spur. Once inside the enclosure, the waste will be size reduced as necessary before being loaded into rail cars for transport to Clive, Utah. All work in the enclosures will be done under negative pressure and periodic surveys will be performed within the enclosures and on the roadways, as necessary, to ensure that control is maintained.

The information provided in Chapter 3 supports the assessment of impacts considered in other sections of the LTP and provides sufficient detail to identify resources needed during the remaining dismantlement activities.

1.5.4 Remediation Plans

LTP Chapter 4 discusses the various remediation techniques that may be used during decommissioning to reduce residual contamination to levels that comply with the release criteria in 10 CFR 20.1402. This chapter also discusses the required ALARA evaluation and the impact of remediation activities on the Radiation Protection Program.

The selected remediation methods used are dependent upon the contaminated material and extent of contamination. The principal materials that may be subject to remediation are structural surfaces. Very limited soil contamination is expected, and minimal groundwater or surface water contamination has been identified to date. Remediation techniques that may be used for structural surfaces include scabbling and shaving, chipping, sponge and abrasive blasting, standard and pressure washing, wiping, grit blasting, mechanical fracturing and cutting, and other methods. Surface and subsurface soil with activity levels in excess of the appropriate derived concentration guideline level (DCGL) will be removed and disposed as radioactive waste. Soil remediation equipment will include, but not be limited to, back and track hoe excavators. Remediation of soils will include the use of established excavation safety and environmental control procedures as well as appropriate work package instructions to ensure adequate erosion, sediment, and air emission controls during soil remediation.

OPPD will use the guidance for conducting ALARA analyses in Appendix N of NUREG-1757, Volume 2, which describes acceptable methods for determining when further reduction of residual radioactivity is required to concentrations below the levels necessary to satisfy the 25 mrem/year dose criterion.

1.5.5 Final Radiation Survey Plan

LTP Chapter 5 presents the FSS Plan which will be used to develop the site procedures, survey packages, and instructions to perform the FSS of the FCS site. The FSS Plan describes the final survey process used to demonstrate that the FCS facility and site comply with radiological criteria for unrestricted use specified in 10 CFR 20.1402 (i.e., annual dose limit of 25 mrem to an average member of the critical group [AMCG] plus ALARA for all dose pathways). Another objective of the FSS Plan will be to demonstrate that the potential dose from small areas of elevated activity is below the release criterion for each survey unit.

The FSS Plan describes the development of the survey plan, survey design and data quality objectives (DQO), survey methods and instrumentation, data collection and processing, data assessment and compliance, and reporting of results. The FSS Plan also describes the methods used to implement and maintain isolation and control measures to prevent contaminating remediated areas and areas that previously underwent FSS. The plan allows for the use of advanced technologies so long as the survey quality is demonstrated to be equal to or better than traditional methods described in MARSSIM.

The FSS Plan incorporates measures to ensure that final survey activities are planned and communicated to regulatory agencies to allow the scheduling of inspection activities by those agencies if so desired.

The FSS Plan addresses FCS basement structures (including embedded piping and penetrations), above-grade structures, buried piping, and land areas that are identified as impacted. The adjacent areas that were classified as non-impacted and the ISFSI, which will still remain a licensed area, will not be subject to FSS.

1.5.6 Compliance with the Radiological Criteria for License Termination

LTP Chapter 6 presents the information, methods, and calculations used to demonstrate compliance with the radiological criteria for license termination and release of the site for unrestricted use. Chapter 6 provides the site-specific initial suite of radionuclides and mixture fractions, future land use scenarios, AMCG, environmental and exposure pathways, conceptual models and mathematical models used for the dose assessment of backfilled basements (surfaces, embedded pipe, penetrations), soil, buried pipe, above-ground structures and existing groundwater.

The reasonably foreseeable scenarios are assessed and the bounding Resident Farmer scenario is selected as the compliance scenario. Less likely but plausible scenarios are also assessed to ensure that, if land uses other than the reasonably foreseeable land uses were to occur in the future, unacceptably high risks would not result.

The mathematical model selected for the dose assessment is RESRAD-ONSITE version 7.2. Probabilistic uncertainty analysis is performed to ensure that prudently conservative or site-specific values are selected for parameters that significantly affect dose. The screening values in Table H.1 of [4] are applied to above-ground buildings. DCGLs and area factors ($DCGL_{EMC}$) are calculated and adjusted to account for the dose from insignificant radionuclides that are eliminated from detailed assessment.

1.5.7 Update of the Site-Specific Decommissioning Costs

In accordance with 10 CFR 50.82(a)(9)(ii)(F), LTP Chapter 7 provides an updated estimate of the remaining decommissioning costs for releasing the FCS site for unrestricted use. This chapter also compares the estimated remaining cost with the funds currently available in the decommissioning trust fund.

1.5.8 Supplement to the Environmental Report

LTP Chapter 8 updates the environmental report for FCS with new information and any significant environmental impacts associated with the site’s decommissioning and license termination activities. This section of the LTP is prepared pursuant to 10 CFR 51.53(d) and 10 CFR 50.82(a)(9)(ii)(G). In accordance with 10 CFR 51.53(d), OPPD considers Chapter 8 as a supplement to the Environmental Report addressing the actual or potential environmental impacts associated with the execution of the described decommissioning activities.

LTP Chapter 8 compares the described decommissioning attributes to those identified in NUREG-0586, which provides a generic environmental assessment for the decommissioning of a reference nuclear facility. The environmental assessment performed by OPPD determined that the environmental effects for decommissioning FCS are minimal and there are no adverse effects outside the bounds of NUREG-0586. Additionally, the U.S. Atomic Energy Commission (AEC) “Final Environmental Statement related to operation of Fort Calhoun Station Unit 1” [13], and NUREG-1437, Supplement 12, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Fort Calhoun Station Unit 1 – Final Report” [14] are reviewed to verify that the decommissioning activities do not invalidate the conclusions of these documents.

1.6 Regulatory Notifications of Changes

OPPD is submitting the LTP as a supplement to the DSAR. Accordingly, OPPD will update the LTP in accordance with 10 CFR 50.71(e). Once approved, OPPD may make changes to the LTP, without prior NRC approval, in accordance with the criteria in 10 CFR 50.59, 10 CFR 50.82(a)(6), and 10 CFR 50.82(a)(7).

OPPD is also submitting a proposed amendment to the FCS license that adds a license condition that establishes the criteria for determining when changes to the LTP require prior NRC approval. Changes to the LTP require prior NRC approval when the change:

- requires Commission approval pursuant to 10 CFR 50.59,
- results in significant environmental impacts not previously reviewed,
- detracts or negates the reasonable assurance that adequate funds will be available for decommissioning,
- decreases a survey unit area classification (i.e., impacted to not impacted, Class 1 to Class 2, Class 2 to Class 3, or Class 1 to Class 3 without providing NRC a minimum 14-day notification prior to implementing the change in classification),
- increases the DCGLs and related minimum detectable concentrations (for both scan and fixed measurement methods),
- increases in the radioactivity level, relative to the applicable DCGL, at which an investigation occurs,
- changes the statistical test applied to one other than the Sign test, or
- increases in the Type I decision error.

The LTP will also be updated every two years. The contact for LTP information, including any submitted changes and updates, is:

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1.7 References

[1] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.179, Revision 2, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," 2019.

[2] U.S. Nuclear Regulatory Commission, "NUREG-1700, Revision 2, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," 2018.

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

[4] U.S. Nuclear Regulatory Commission, "NUREG-1757, Volume 2, Revision 2, Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria - Draft Report for Comment," 2020.

[5] Omaha Public Power District, Fort Calhoun Station Unit 1, “Defueled Safety Analysis Report”.

[6] TSSD Services, "Fort Calhoun Station Limited Site Non-Radiological Characterization Survey," 2016.

[7] Omaha Public Power District, "Fort Calhoun Station, Flood Recovery Action Plan".

[8] Omaha Public Power District, "Review of the Groundwater Protection Program at the Fort Calhoun Nuclear Station, Revision 1," 2008.

[9] Haley & Aldrich, "Memorandum to EnergySolutions, File No. 0127960-006, Fort Calhoun Nuclear Power Station - Well Research," April 13, 2021.

[10] Omaha Public Power District, "FC-20-012, Fort Calhoun Station Decommissioning Project Radiological Characterization Report".

[11] Radiation Safety and Control Services, "TSD 20-001, Historical Site Assessment for Fort Calhoun Station," 2020.

[12] U.S. Nuclear Regulatory Commission, "NUREG-0586, Supplement 1, Volume 1, Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," 2002.

[13] Omaha Public Power District, "Final Environmental Statement related to operation of Fort Calhoun Station Unit 1," 1972.

[14] U.S. Nuclear Regulatory Commission, "NUREG-1437, Supplement 12, Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Fort Calhoun Station Unit 1 – Final Report," 2003.

Figure 1-1 FCS Site Location

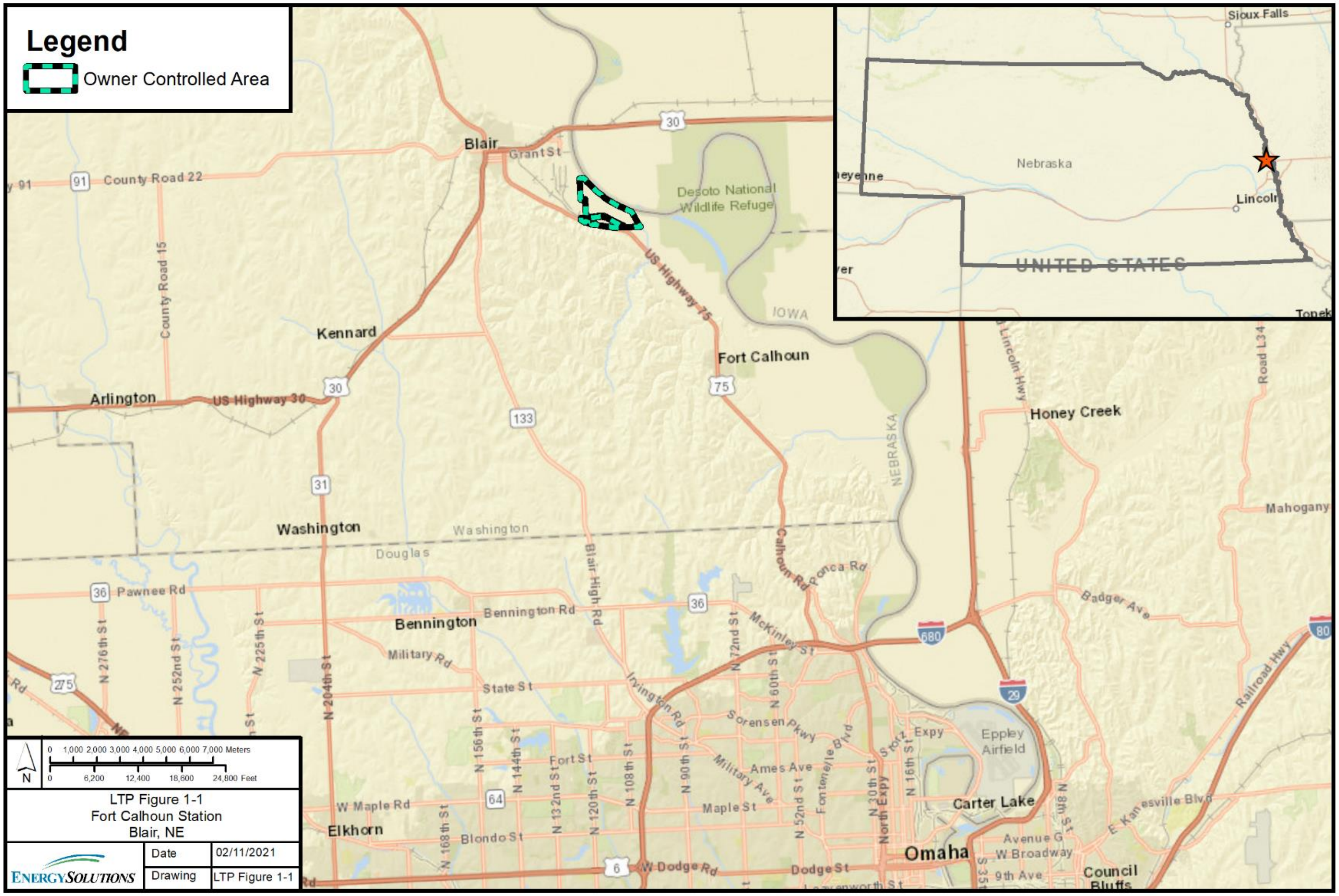


Figure 1-2 FCS Owner Controlled Areas

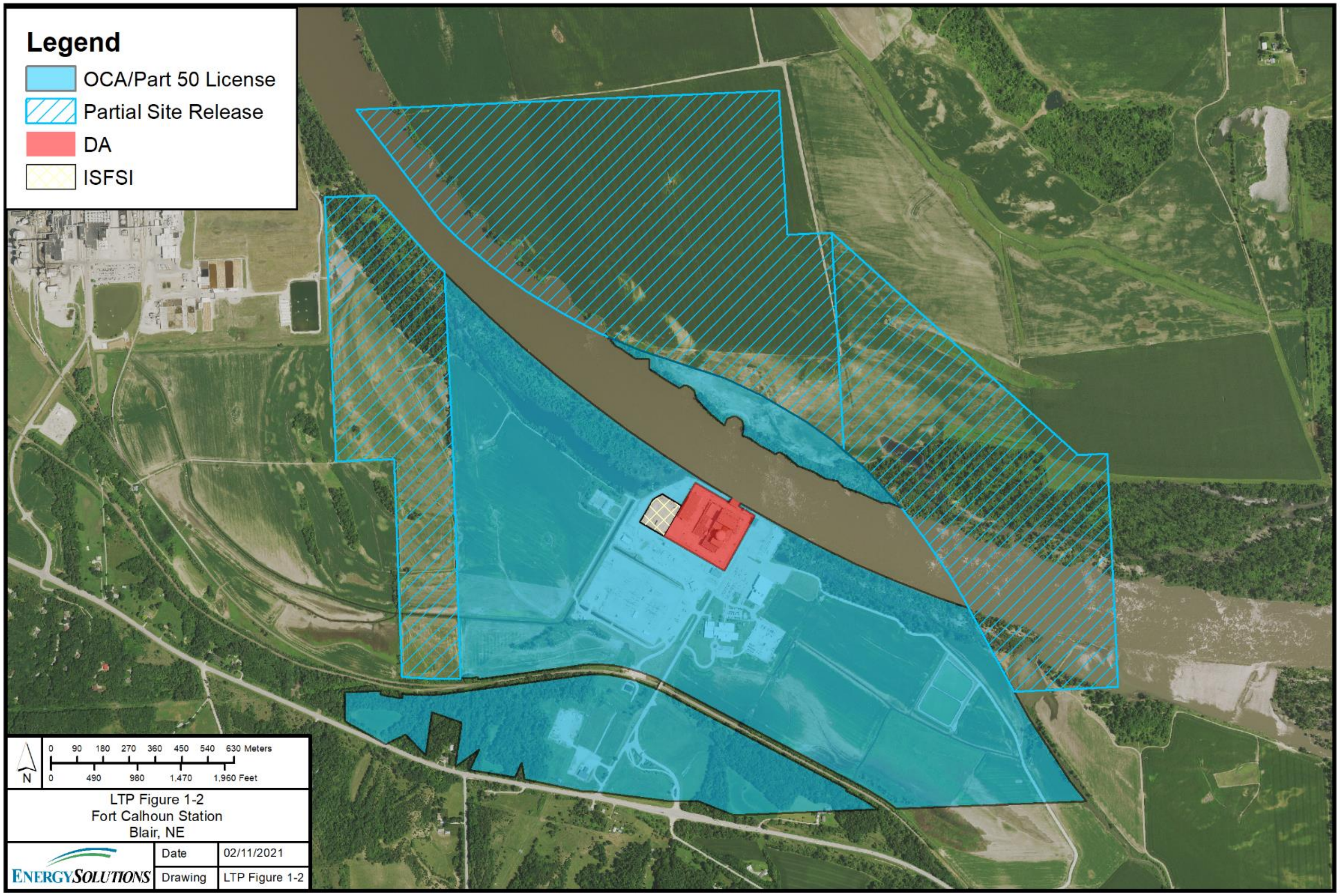


Figure 1-3 FCS Site Layout

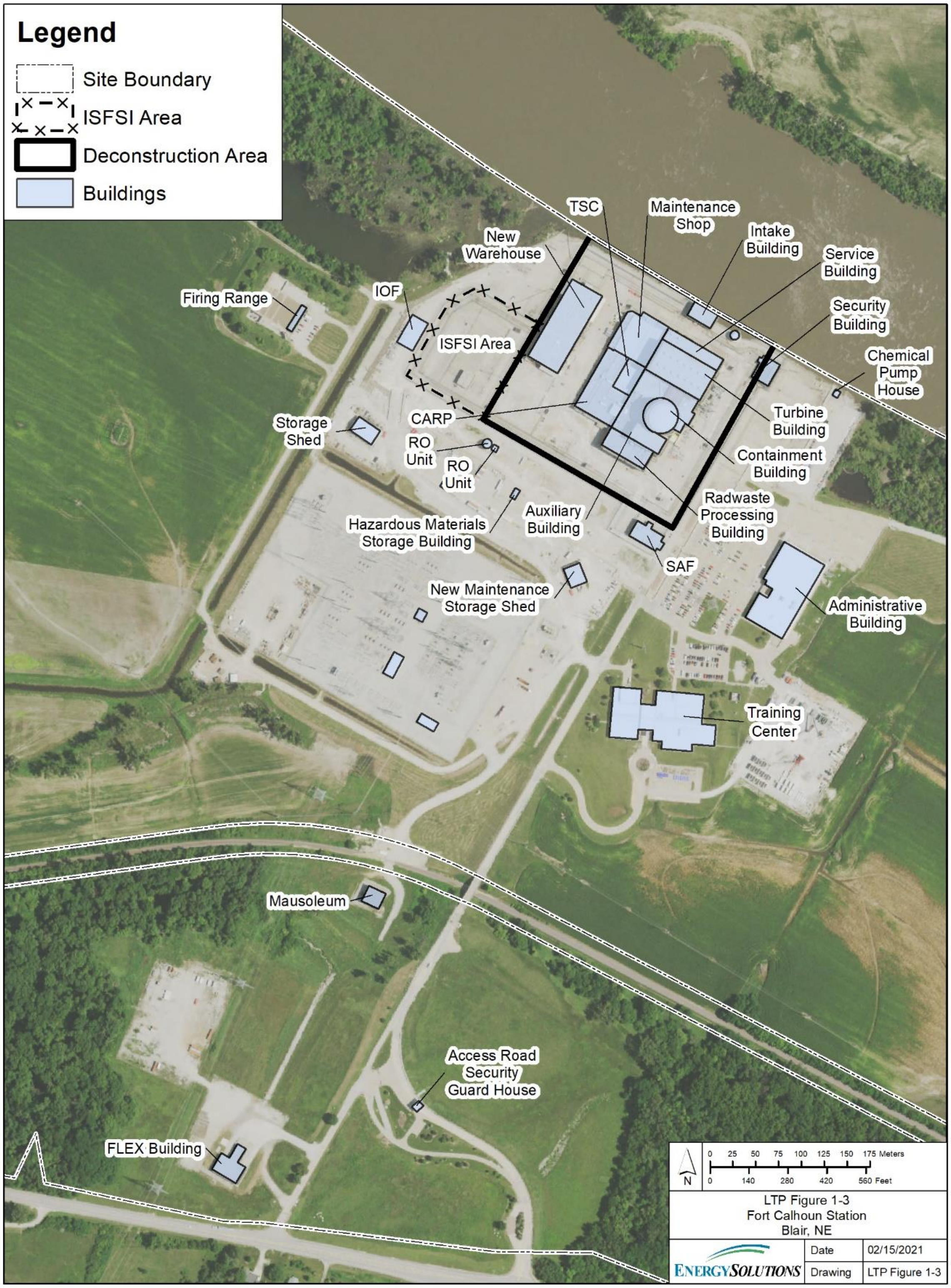


Figure 1-4 FCS Site Topography

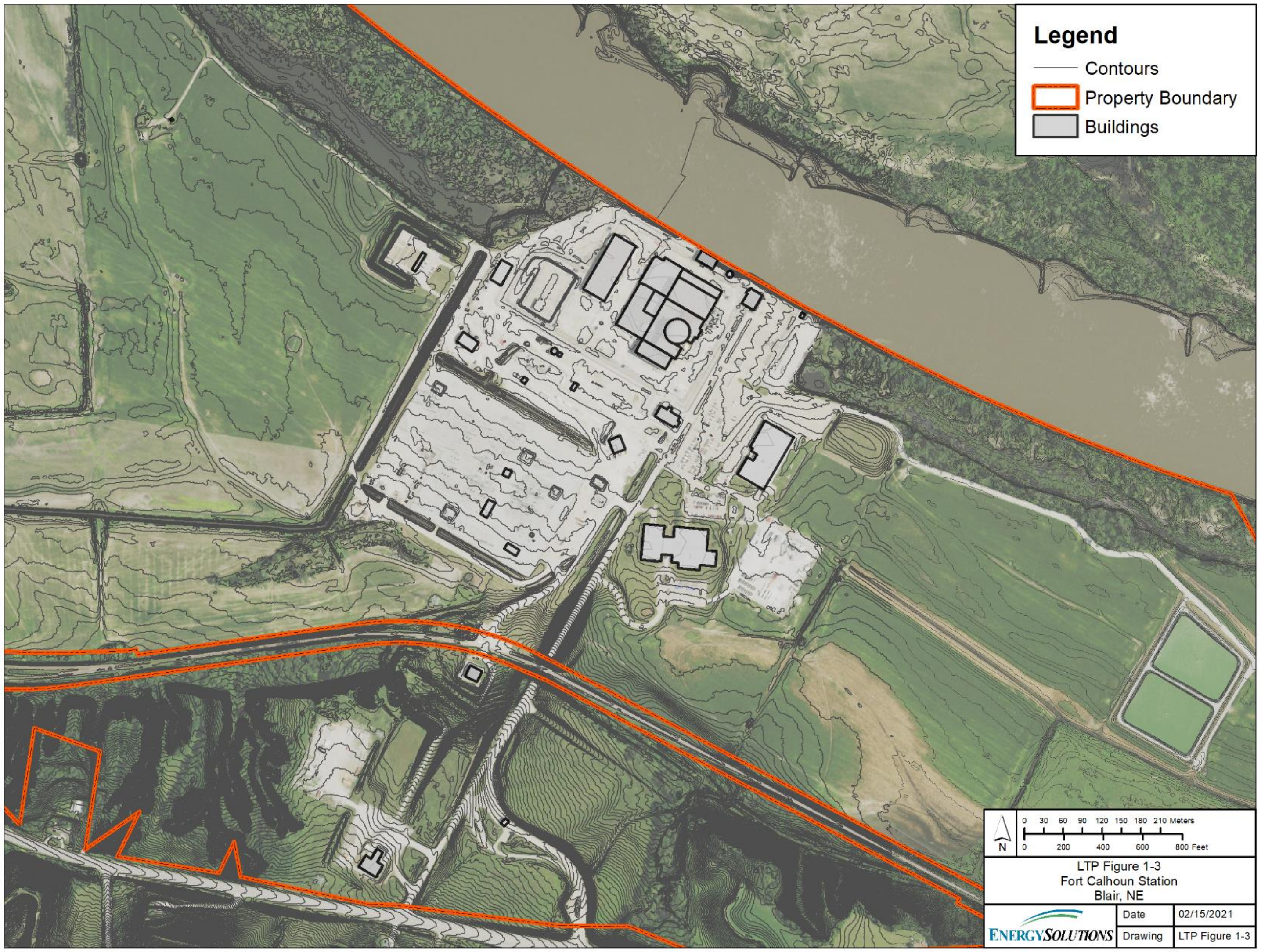


Figure 1-5 FCS Site Water Features

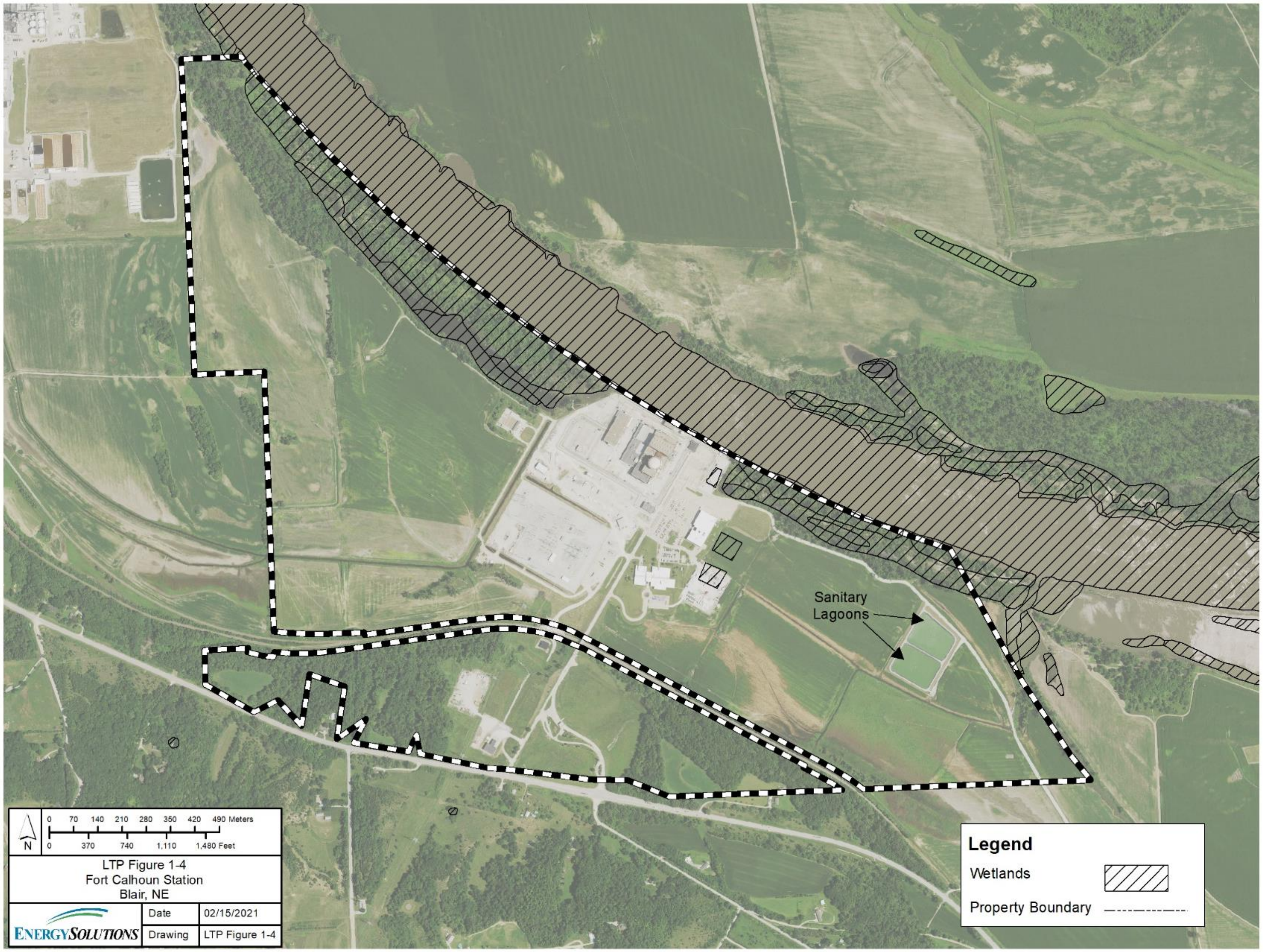


Figure 1-6 Extent of 2011 Flooding Incident

Figure 1-7 HESCO Barrier

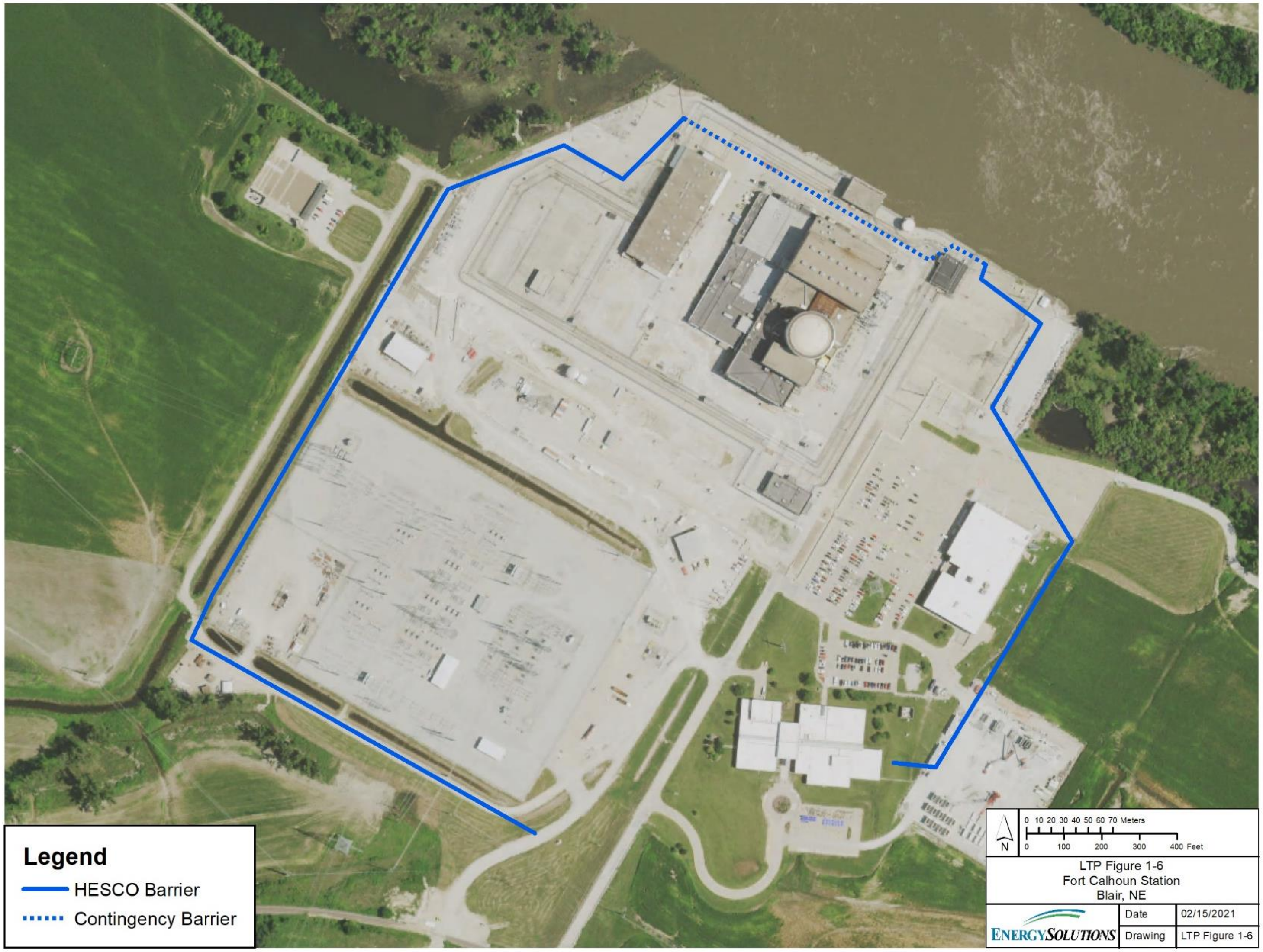
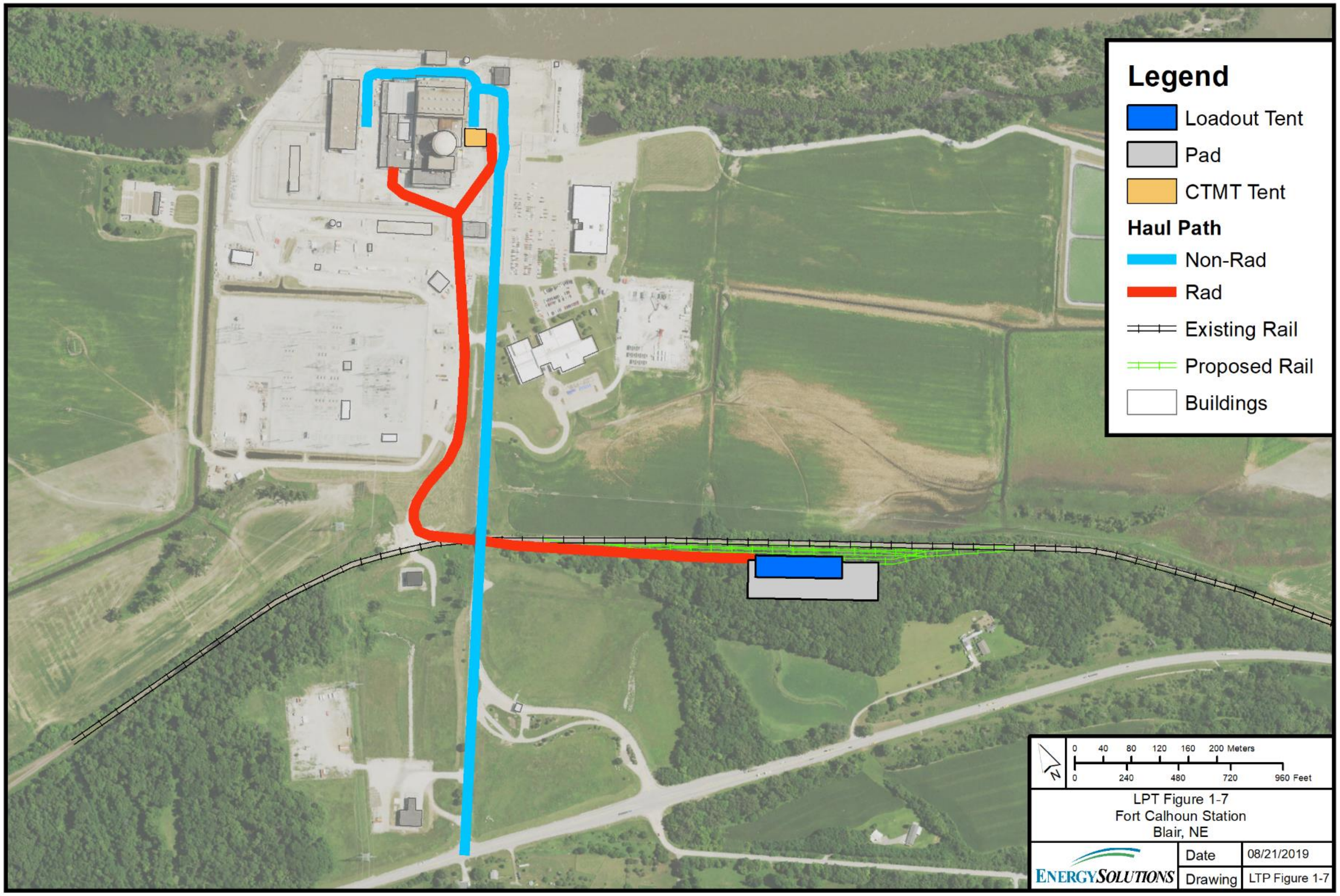


Figure 1-8 Waste Enclosures and Haul Paths



FORT CALHOUN STATION DECOMMISSIONING PROJECT
LICENSE TERMINATION PLAN

CHAPTER 2
SITE CHARACTERIZATION

REVISION 0

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ABBREVIATIONS

AB	Auxiliary Building
AMSL	above mean sea level
BOP	balance of plant
CB	Containment Building
DA	deconstruction area
DCGL	derived concentration guideline level
DOE	U.S. Department of Energy
DQO	data quality objective
EPA	U.S. Environmental Protection Agency
FCS	Fort Calhoun Station
FSS	final status survey
GTCC	greater than Class C
HSA	Historical Site Assessment
HTD	hard-to-detect
ISFSI	Independent Spent Fuel Storage Installation
LTP	license termination plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
NQEQ	Nebraska Department of Environmental Quality
NRC	U.S. Nuclear Regulatory Commission
OCA	Owner Controlled Area
ODCM	Offsite Dose Calculation Manual
OPPD	Omaha Public Power District
PCB	polychlorinated biphenyl
QAPP	quality assurance project plan
RCA	radiologically controlled area
REMP	Radiological Environmental Monitoring Program
RGPP	Radioactive Groundwater Protection Program
ROC	radionuclides of concern

RP	radiation protection
SAF	Security Access Facility
SIRWT	Safety Injection Refueling Water Tank
SFP	Spent Fuel Pool
SOF	sum of fractions
SSC	structure, system, and component
TB	Turbine Building
TLD	thermoluminescent dosimeter
TSC	Technical Support Center
URS	unconditional release survey

2 SITE CHARACTERIZATION

In accordance with the requirements of 10 CFR 50.82 (a)(9)(ii)(A) and the guidance of Regulatory Guide 1.179, "Standard Format and Contents for License Termination Plans for Nuclear Power Reactors" [1], this chapter provides a description of the radiological characterization performed at the Fort Calhoun Station (FCS) site.

The purpose of site characterization is to ensure that the final status survey (FSS) will be conducted in all areas where contamination existed, remains, or has the potential to exist or remain. The results of the characterization survey, along with the information in the "Historical Site Assessment for Fort Calhoun Station" (HSA) [2], demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected. There are areas known to potentially contain contamination that were inaccessible during the initial characterization which will be surveyed during continuing characterization as access is gained (see Section 2.5).

The site characterization incorporates the results of investigations and surveys conducted to quantify the extent and nature of contamination at the FCS site. In addition, the results of site characterization surveys and analyses have been and continue to be used to identify areas of the site that will require remediation, as well as to plan remediation methodologies, develop waste classification and volumes, and estimate costs.

The characterization surveys were designed and executed using the guidance provided in NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM) [3] and NUREG-1757, Volume 2, "Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria" [4]. In addition, surveys were designed and executed in accordance with the FCSD-RA-LT-200, "Characterization Survey Plan" [5], and FCSD-RA-LT-100, "Quality Assurance Project Plan for the License Termination Plan Development, Site Characterization and Final Status Survey Projects at Fort Calhoun Station" (QAPP) [6], which describes policy, organization, functional activities, the data quality objectives (DQO) process, and measures necessary to achieve quality data. The information obtained from the characterization provides guidance for decontamination and remediation planning. Materials which were shown to be contaminated with radioactive material at concentrations greater than the unrestricted release criteria have been and will continue to be removed and properly packaged for shipment and disposal.

The site characterization of FCS includes the information requirements listed in NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans" [7] and [4]. Measurements and samples were collected in each accessible area, along with the historical information, to provide a clear picture of the residual radioactive materials and its vertical and lateral extent at the site. Using appropriate DQOs, surface soil samples, subsurface soil samples, sediment samples, concrete samples, gross beta measurements, and smear samples were collected to provide a profile of the residual radioactivity at the site. Volumetric samples were analyzed for the applicable radionuclides with detection limits that provided the level of detail necessary for decommissioning planning. Based upon the volume of characterization data collected and an assessment of the characterization results, the characterization survey is considered adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected.

As part of the decommissioning process, all spent reactor fuel was loaded into casks and transferred to an Independent Spent Fuel Storage Installation (ISFSI). Greater than Class C (GTCC) waste will be loaded into casks and brought to the ISFSI once the reactor vessel segmentation is complete. The fuel and GTCC waste will remain on-site in dry storage within the ISFSI until it is transferred to the Department of Energy (DOE). The ISFSI has been constructed in the northeast corner of the FCS site, immediately north of the deconstruction area (DA), formerly a portion of the protected area.

2.1 Historical Site Assessment

In accordance with guidance provided in Section 3.0 of MARSSIM, an HSA was issued in February 2020. Historical information, including any 10 CFR 50.75(g) files, employee interviews, radiological incident reports, pre-operational survey data, spill reports, special surveys (e.g., site aerial surveys, marine fauna and sediment surveys), operational survey records, and Annual Radiological Environmental Reports (including sampling of air, groundwater, estuary water, milk, invertebrates, fish and surface vegetation) to the NRC were reviewed and compiled for this investigation.

2.1.1 Objectives

The HSA is a detailed investigation that collects existing information (from the start of FCS activities related to radioactive materials or other contaminants) for the site and its surroundings. The HSA focuses on historical events and routine operational processes that resulted in contamination of plant systems, on-site buildings, surface soils, and subsurface soils within the Owner Controlled Area (OCA). The information compiled by the HSA was used, in conjunction with [5], to establish initial survey units and their MARSSIM classifications. This information was used as input into the development of site-specific derived concentration guideline levels (DCGL), remediation plans, and the design of the FSS. The scope of the HSA included potential contamination from radioactive materials, hazardous materials, and other regulated materials.

The objectives of the HSA are to:

- identify potential, likely, or known sources of radioactive and chemical contaminants based on existing or derived information,
- distinguish portions of the site that may need further action from those that pose little or no threat to human health,
- provide an assessment of the likelihood of contaminant migration,
- provide information useful to subsequent characterization surveys,
- provide a graded initial classification for impacted soils and structures in accordance with MARSSIM guidance, and
- delineate initial survey unit boundaries and areas based upon the initial classification.

2.1.2 Methodology

The objective of the HSA records search and interview process was the identification of events posing a significant probability of impacting the hazardous material or radiological status of FCS site land areas and structures. These events include system, structure, or area contamination from system failures resulting in airborne releases, liquid spills or releases, or the loss of control over solid material. Depending upon previous site operations and processes, the potential for residual contamination varies by area. In order to facilitate effective characterization surveys to guide future decontamination activities and provide sufficient data for the design of FSS, land areas and structures are classified based upon their potential for contamination.

Each incident identified that posed a realistic potential to impact the characterization of the site was further investigated. This investigation focused on the scope of contaminant sampling and analysis, any remedial actions taken to mitigate the situation, and any post-remedial action sampling, survey, and analysis in an attempt to identify the “as-left” condition of the incident location. Historical records archives provided the source of a vast majority of the documents inspected.

Also included in the research associated with the development of the HSA were:

- relevant excerpts from written reports and correspondences
- personnel interviews, including the use of questionnaires, of current, former, and retired plant personnel to confirm documented incidents and identify undocumented incidents
- site inspection, utilizing site drawings, photographs, prints, and diagrams to identify, locate, confirm, and document areas of concern

Information from this research was used in the HSA development, including the compilation of data, evaluation of results, documentation of findings, and the identification of initial survey unit classifications.

2.1.2.1 Preliminary Classification

The HSA investigation was designed to obtain sufficient information to provide initial classification of the site land areas and structures as impacted or non-impacted. Impacted areas have a potential for contamination (based on historical data) or contain known contamination (based on past or preliminary radiological surveillance). Non-impacted areas are identified through knowledge of site history or previous survey information and are areas where there is no reasonable possibility for residual radioactive contamination. Areas were classified as impacted from a radiological perspective. Potential chemical hazards incidents on owner-controlled areas were also documented, including the confirmed presence of petroleum products, asbestos, or other hazardous materials.

If a land area or structure was classified as impacted, then a determination of the initial impacted area classification (e.g., Class 1, Class 2, or Class 3), in accordance with Section 2.2 of MARSSIM, was made based upon the information obtained.

Initial classification of FCS areas was based on historical information and available historical radiological survey data. Classifying a survey area has a minimum of two stages: initial classification and final classification. Initial classification of most areas is performed at the time of identification of the survey unit, using the information available when the HSA was prepared. Final classification is performed and verified as a DQO during FSS design. Radiological survey data from characterization surveys, operational surveys in support of decommissioning, routine surveillance, and any other applicable survey data may cause an increase in survey area classifications (e.g., from Class 3 to Class 2 and from Class 2 to Class 1) until the time of commencement of the FSS.

2.1.2.2 Documents Reviewed

Records maintained to satisfy the requirements of 10 CFR Part 50.75(g)(1) provided a major source of documentation for the HSA records review process. During the conduct of the HSA for FCS, many record types were evaluated, including paper, microform, and electronic media. A complete listing of documents reviewed is provided in Appendix A of the HSA. A summary of the types of records reviewed include:

- interviews of long-tenured employees knowledgeable of site operations
- records from the Nebraska Department of Environment and Energy
- FCS incident files such as Condition Reports, Incident Reports, Radiological Occurrence Reports, and Licensee Event Reports
- FCS special survey and operational radiological survey records
- engineering reports of environmental assessments and subsurface investigations at FCS
- the FCS file maintained in compliance with federal regulation 10 CFR 50.75(g)
- the FCS Offsite Dose Calculation Manual (ODCM) [8]
- the FCS Updated Safety Analysis Reports
- the FCS Spill Prevention, Control and Countermeasures Plan
- the FCS Annual Radioactive Effluent Release Reports
- the FCS Annual Radiological Environmental Operating Reports
- the TSSD Services, Inc., "Fort Calhoun Nuclear Station Historical Site Assessment Report" [9]
- the TSSD Services, Inc., "Fort Calhoun Nuclear Station Limited Non-Radiological Characterization Survey" [10]

2.1.2.3 Licenses, Permits, and Authorizations

FCS was operated in accordance with several Federal and State of Nebraska licenses and permits. The NRC site License for FCS and support Technical Specifications allowed Omaha Public Power District (OPPD) to use any quantity of radioactive material at the site to support operations during its lifetime and to implement decommissioning activities.

The U.S. Environmental Protection Agency (EPA) and the Nebraska Department of Environmental Quality (NDEQ) maintain files on a variety of environmental programs that are applicable to FCS. These include permit applications and monitoring results with information on specific waste types and quantities, sources, type of site operations, and operating status of the facility or site.

The following denotes the licenses and permits relevant to the development of the HSA:

- NRC – Docket No. 50-285, Facility Operating License No. DPR-40
- EPA – ID No. 110007129623

From August 9, 1973, to the present, the facility has been operated under NRC License No. DPR-40, Docket No. 50-285. This period includes power operations to 2016, permanent shutdown for decommissioning in 2016, and ISFSI construction and spent fuel load-off which was completed in 2020.

2.1.2.4 Personnel Interviews

Long term employees with historical knowledge of station operations were interviewed regarding the plant operating history in November and December of 2019. The intent of the employee interviews was to capture the institutional knowledge of those familiar with plant operation and construction. As detailed in the conclusions of the HSA, based on those interviews, there do not appear to be any undocumented incidents of significant contamination at the station. Further, none of the identified impacted areas or structures, systems, and components (SSC) are an imminent threat to human health or the environment.

2.1.3 Operational History

The station was granted a construction permit by the U.S. Atomic Energy Commission in 1968, and plant construction began that year. The first fuel assembly was loaded into the reactor May 24, 1973. The final fuel assembly was loaded on June 8, 1973 and the final core verification was completed on June 9, 1973. The NRC issued an operating license on August 9, 1973. The plant officially went online on September 1, 1973, with commercial operation beginning on September 26, 1973. On June 24, 2016, and updated on August 25, 2016, OPPD submitted the Certifications of Permanent Cessation of Power Operations in accordance with 10 CFR Part 50.82(a)(i). The plant went offline on October 24, 2016.

OPPD began actively decommissioning FCS on April 29, 2019, and completed the transfer of all spent nuclear fuel to the ISFSI in May 2020. The ISFSI is located in the protected area designated for dry storage of spent fuel and GTCC waste. The ISFSI is licensed under Subpart K of 10 CFR Part 72.

2.1.4 Incidents

2.1.4.1 Radiological Events

FCS experienced several radiological events over the operating years, including a steam generator tube rupture and an overflow event involving the Safety Injection Refueling Water Tank (SIRWT). The steam generator tube rupture, which resulted in radioactive contamination in interior rooms and floors in the Turbine Building and Auxiliary Building, occurred in 1984. This event is significant in that the main steam and condensate, feedwater, and blowdown systems were contaminated by reactor coolant. The primary to secondary leakage was confirmed by Chemistry personnel during the interview process. Records providing evidence of environmental contamination resulting from the primary to secondary leaks were not found. Annual radioactive effluent release reports confirmed that effluent releases during this time complied with regulatory release limits, and annual Radiological Environmental Operating Reports contain no evidence of radiological impact to the environment during this time.

The SIRWT over flowed to outside railroad siding door. The SIRWT overflow event also caused an overflow of the Spent Fuel Pool (SFP). Most of the SFP water was confined inside the Auxiliary Building; however, some did escape to the adjacent outdoors area. The DA area affected by the SIRWT event was excavated immediately following the event until analysis confirmed the contamination had been removed. The excavated soil was disposed of as radioactive waste.

During the late 1980s and early 1990s, new buildings, including the Radwaste Processing Building, Chemistry and Radiation Protection Facility, Cafeteria, were constructed within the DA, whereas the Training Center and Administration Building were constructed outside the DA. Construction of the Administration Building required the closure of two sanitary lagoons. Sludge samples from the lagoons showed detectable Cs-137 (approximately 0.1 pCi/g). The old lagoons were replaced by the current operating sanitary lagoons located southeast of the plant. Current practices include quarterly sampling of the sludge and water. Although detectable concentrations of Cs-137 (<0.2 pCi/g), Cs-134 (<0.06 pCi/g), and Co-60 (<0.025 pCi/g) has been found in the sludge samples from both the west and east lagoons, licensed radioactivity has not been detected in the lagoon water, which is disposed by land application in an application area adjacent to the lagoons.

Waste processing at FCS occurred inside the Radwaste Processing Building, which contains provisions for dry active waste processing, liquid radwaste cleanup (using filtration and ion exchange equipment), and radwaste solidification equipment. Gaseous and liquid waste treatment processes for effluent, as well as solid waste handling processes, are described in the FCS ODCM and in the Defueled Safety Analysis Report (DSAR) [11]. Information acquired during interviews revealed that free in/out flow of floodwater entering the Radwaste Processing Building bay area during the 2011 flood was prevented through use of water stops. The captured floodwater was directed to radioactive waste processing prior to release via the liquid effluent discharge system. This action reduced the potential of contaminating the DA soil.

General radiation protection (RP) practices for decontamination activities, including the decontamination of the steam generators in 2006, included the use of lining material (e.g., herculite or equivalent material) at laydown areas to control the spread of contamination to the underlying media and routine RP surveys. These activities were performed in accordance with site RP procedures.

Sealand containers with outage equipment, radioactive material, or radioactive waste were staged inside and outside the DA. There are no records of any container leakage or loss of container integrity.

During the 2006 refueling outage, OPPD replaced major components, such as steam generators, pressurizer, reactor vessel head, low-pressure turbine, and main transformers. The removed components were staged for decontamination within a tented concrete area located in the southwest corner of the DA. During the decontamination effort, FCS RP personnel performed frequent radiation and contamination surveys to control personnel exposure and the spread of radioactive contamination. After dismantlement of the tent, RP personnel surveyed the concrete base pad for release.

In early 2011, FCS received warning of the pending flood from the U.S. Army Corp of Engineers, who had to control water from a record snowmelt via systematic releases through a network of upstream dams on the Missouri River. FCS prepared for the flood by constructing aqua berms and raised walkways to protect the plant components. On June 6, 2011, when the reactor was shut down for scheduled refueling, FCS declared a Notification of Unusual Event due to the Missouri River flooding. The majority of the FCS site, including the sanitary lagoons, remained flooded for approximately three months. Plant personnel interviewed during the HSA recalled a site-wide 3- to 4-foot water level with a strong current. While preparing for the flood, a large earthen berm, consisting of soil from the site, was constructed around the switchyard. Following the flood, the soil was returned to the hill on the west side of the property from where it came. Sand bags and silt were also removed from the parking lots and then placed on the hill. According to interviewees, samples from sandbags from inside the PA were surveyed for release; some samples showed detectable Cs-137. However, interviewees also stated that sand showing detectable Cs- 137 was disposed of as radioactive waste. Following flood recovery and special oversight of the NRC, the plant was allowed to restart and return to service in December 2013. Figure 1-6 in Chapter 1 shows the devastating impact of the 2011 flooding.

A review of the annual Radiological Environmental Operating Reports for the years 2005 through 2015 revealed that no licensed radioactivity was identified in any of the off-site environmental samples, which included sediment, vegetation, surface water, ground water, well water, milk, air, fish, and food crops. Laboratory analyses were performed to achieve the lower limits of detection (LLD) established by the FCS ODCM.

2.1.4.2 Non-Radiological Events

Areas of interest were identified for non-radiological (chemical) use based on historical use and operations. However, there have been two non-radiological events that may have released non-radiological constituents to the environment: an exothermic reaction within the piping from the water treatment plant to the chemical treatment lagoons and a transformer being dropped during its transport via a goldhofer trailer.

The first event is the exothermic reaction, related to the original water treatment plant when the treated water line had both caustic and acid within the line causing a number of manhole covers to blow off. It is unknown if this reaction created a release to the environment.

The second event occurred when the spare main transformer was dropped while being moved on a goldhofer trailer. The trailer suffered a hydraulic line break, which caused the levelizing feature on the trailer to fail and dump the transformer. It was reported that the spare main transformer load was dropped, potentially causing a minor release.

Additionally, other non-radiological chemical events most likely occurred, but were related to general site operations. These included water treatment operation, chemical storage, hazardous waste storage, and water treatment lagoons maintenance. Other activities at the FCS that could have significant non-radiological impacts, include fire prevention training and target practice.

2.1.5 Findings and Conclusions

The HSA was completed in accordance with guidance provided in MARSSIM. As expected, operational activities at FCS from initial power generation in September 1973 to the present have resulted in areas that have been impacted with radiological and/or non-radiological contaminants. A general conclusion that can be drawn from the record reviews, personnel interviews, and site walk-downs that were part of HSA development is that FCS had an excellent operating history in that operations have resulted in very low radiological and non-radiological potential impacts to the environment beyond the DA. No identified areas of contamination are a current or expected threat to human health or the environment that would warrant immediate corrective action.

The information developed in the HSA indicates that the areas and SSCs with a high probability of requiring remediation (Class 1) are located within radiologically controlled areas (RCAs). Migration of surface contamination from the RCA appears to be limited, as has been determined from frequent site surveys conducted inside the DA.

Ordinarily, historic events that resulted in contamination were remediated immediately at the time of their discovery. In these instances, the remediation was accomplished to the point of no significant measured radioactivity above background, within the counting capabilities of field and lab instrumentation available at the time. These instances and impacted areas were appropriately documented in historical files and are cited within the HSA. It is presumed that modern detection and counting capabilities would either corroborate those results, or further identify residual radioactivity.

The following conclusions are presented for consideration and to clearly state important observations:

- Known incidents of contamination have been remeditated and none of the impacted areas or SSCs are an imminent threat to human health or the environment.
- Historic releases at the station have been managed in accordance with applicable radiological and non-radiological regulations.
- Each area identified as potentially impacted will require further characterization as it becomes more accessible during decommissioning to determine the extent to which it may have been impacted, if at all.
- No new impacted areas that were not previously known have been identified by the HSA.
- Where lead-based paint, asbestos containing material, or components containing mercury or polychlorinated biphenyls (PCB) are present, the areas are located within buildings, are not exposed to the environment, and are being managed in accordance with site procedures (e.g., use of negative pressure, enclosures, and designated haul roads). The current management practices for these areas are sufficient to ensure the safety of site workers until the materials of concern are permanently removed.

- FCS has implemented the guidance prescribed by NEI 07-07, "Industry Ground Water Protection Initiative – Final Guidance Document" [12] and has established an on-going groundwater monitoring program.
- Groundwater is not used as a source of drinking water on or within 0.5 miles of the station.
- Groundwater flow beneath the station is primarily easterly towards the Missouri River; however, bank flow conditions give rise to seasonal westerly flow near the river.
- The only radionuclide contaminants identified in groundwater are tritium and Sr-90.

2.1.6 Initial Survey Units and Classification

2.1.6.1 Survey Units

The entire 540 acre FCS site (plus 26 acres of property to the east across the river) was divided into preliminary survey units and assigned initial classifications based on operational history and the incidents and processes documented for a specific survey unit. The survey units established by the HSA and FCSD-RA-LT-200, "Characterization Survey Plan" were used as initial survey units for characterization. A survey unit is a structure, portion of a structure, open land area, or system that is surveyed and evaluated as a single entity. Survey units were delineated to physical areas with similar operational history or similar potential for residual radioactivity. To the extent practical, survey units were established with relatively compact shapes, and highly irregular shapes were avoided unless the unusual shape was appropriate for the site operational history or the site topography. Additionally, survey unit sizes were established to adhere to the suggested sizes provided in MARSSIM. A summary of the initial survey unit classifications is presented in the following sections.

2.1.6.1.1 Class 1 Structures

Table 2-1 below presents the Class 1 structures: the Containment Building, the Auxiliary Building, and the Radwaste Processing Building. The structures contain the nuclear reactor, primary reactor systems, reactor support systems, radioactive waste systems, and nuclear fuel handling and storage systems. During operations, radioactive material was routinely handled, transferred, and stored within these buildings. Throughout facility operations, these structures were subjected to spills of radioactive liquids, the spread of loose surface contamination, and airborne radioactive material.

The end state configuration of the Class 1 structures is discussed in Chapter 3.

Table 2-1 Class 1 Structure Survey Units

Survey Unit ID No.	Survey Unit Description	Classification
1000	Unit 1 Containment Building (CB)	
1100	CB 977' Elevation – Under Vessel Area	1
1200	CB 995'/996' Elevation G/A	1

Survey Unit ID No.	Survey Unit Description	Classification
1201	CB 995' Elevation – 'A' S/G Enclosure	1
1202	CB 995' Elevation – 'B' S/G Enclosure	1
1203	CB 996' Elevation – Reactor Cavity	1
1204	CB 996' Elevation – Fuel Transfer Canal	1
1300	CB 1013' Elevation G/A	1
1400	CB 1045' Elevation G/A	1
1500	CB 1060' Elevation G/A	1
1600	CB Exterior Surfaces	-
1601	CB Roof	2
1602	CB Exterior Walls	3
2000	Auxiliary Building (AB)	
2100	AB 971' Elevation G/A	1
2200	AB 989' Elevation G/A	1
2201	AB 994' Elevation – Spent Fuel Pool	1
2300	AB 1007' Elevation G/A	1
2400	AB 1011' Elevation G/A	1
2500	AB 1013' Elevation G/A	1
2600	AB 1025' Elevation G/A	1
2700	AB 1036' Elevation G/A	1
2800	AB 1039' Elevation G/A	1
2900	AB Exterior Surfaces	-
2901	AB Roof	2
2902	AB Exterior Walls	3
4100	Radwaste Processing Building	1

2.1.6.1.2 Class 2 and 3 Structures

The primary functions of the Class 2 and 3 structures are to house the secondary side steam systems or electrical generating systems, or to act as office, warehouse, or security space. Table 2-2 below presents a list of the Class 2 and 3 structures with relation to the FCS DA, formerly a portion of the protected area. The majority of these structures did not routinely house radioactive systems or materials during operations. However, it was possible, due to their physical proximity to effluent release pathways, radioactive contamination of secondary side systems, temporary storage and transport of radioactive materials in and through these buildings, and past incidents involving the loss of control of radioactive material that residual radioactive material could be found in, on, or around these structures. Consequently, the initial Class 2 or 3 designations are justified. All of the structure survey units are depicted in Figure 2-1.

The end state configuration of the Class 2 and 3 structures is discussed in Chapter 3.

Table 2-2 Class 2 and 3 Structure Survey Units

Survey Unit ID No.	Survey Unit Description	Classification
3000	Turbine Building (TB)	
3100	TB 990’ Elevation G/A	3
3200	TB 1000’ Elevation G/A	3
3300	TB 1011’ Elevation G/A	3
3400	TB 1036’ Elevation G/A	3
4000	Balance of Plant (BOP) Buildings Inside DA	
4200	Intake Building	3
4300	Security Building	3
4400	Security Access Facility (SAF)	3
4500	Service Building	3
4600	Maintenance Shop	3
4700	Technical Support Center (TSC)	3
4800	Chemistry and Radiation Protection Facility	3
4900	New Warehouse	3
5000	BOP Buildings Outside DA	
5100	Administrative Office Building	3
5200	Training Center	3
5300	Mausoleum (Original Steam Generator Storage Facility)	2
5400	FLEX Building	3
5500	New Maintenance Storage Shed	3
5600	Chemical Pump House (Neutralization Basin)	3
5700	Storage Shed	3
5800	Sanitary Lift Stations	3
9000	Miscellaneous SSC	
9001	Blast and Ballistic Rated Enclosures	3
9002	Reverse Osmosis Units	3
9003	Firing Range	3

Survey Unit ID No.	Survey Unit Description	Classification
9004	Hazardous Materials Storage Building	3
9005	Access Road Security Guardhouse (OCA Entrance Building)	3
9006	3451 Old Building	3
9007	3451 New Building	3
9008	1251 Control and Switchgear Building	3

2.1.6.1.3 <u>Class 1 Open Land Areas</u>

Table 2-3 presents the open land areas that have been initially classified as impacted Class 1. The bases for the initial classification are the open land proximity to Class 1 structures within the DA, an overflow event in 1983, which was documented in the HSA and resulted in localized soil contamination outside the Auxiliary Building, and the location of the waste haul path and waste loadout structure.

Table 2-3 Class 1 Open Land Survey Units

Survey Unit ID No.	Survey Unit Description	Classification
7000	Land Areas Inside DA	
7100	Northwest Land Areas I/S DA Fence	-
7101		1
7102		1
7103		1
7104		1
7105		1
7200	Southwest Land Area I/S DA Fence	-
7201		1
7202		1
7203		1
7204		1
7205		1
7300	Southeast Land Area I/S DA Fence	-
7301		1
7302		1
7303		1
7304		1

Survey Unit ID No.	Survey Unit Description	Classification
7400	Northeast Land Area I/S DA Fence	-
7401		1
7402		1
7403		1
7404		1
7405		1
7406		1
7500	Primary Plant Land Area	-
7501		1
7502		1
7503		1
7504		1
7505		1
7506		1
7507		1
8500	Waste Loadout Containment Structure	-
8501		1[a]
8502		1[a]
8503		1[a]
8504		1[a]
8505		1[a]
8506		1[a]
8507		1[a]
8508		1[a]
8600	Waste Haul Path	-
8601		1[a]
8602		1[a]
8603		1[a]
8604		1[a]
8605		1[a]
8606		1[a]
8607		1[a]

(a) The survey unit classifications listed for survey units 8500 and 8600 are intended as future classifications for FSS. Surveys conducted before the waste haul path and waste loadout structure are constructed may treat these survey units as Class 3.

Based on an assessment of historical incidents and events, it was anticipated that soils in these areas could possibly contain residual radioactive material in excess of the unrestricted release criteria. The Class 1 open land survey units are illustrated in Figure 2-4 and Figure 2-5.

2.1.6.1.4 Class 2 Open Land Areas

Table 2-4 presents the open land areas that have been initially classified as impacted Class 2. Based upon a review of the historical information contained in the HSA and future use, there was a potential for residual radioactive contamination to exceed the unrestricted release criteria.

Table 2-4 Class 2 Open Land Survey Units

Survey Unit ID No.	Survey Unit Description	Classification
8000	OCA Outside the DA	
8400	Waste Haul Path Buffer Zone	-
8401		2[a]
8402		2[a]
8403		2[a]
8404		2[a]
8405		2[a]
8406		2[a]
8700	Sewage Lagoon	-
8701		2
8702		2
8703		2

(a) The survey unit classification listed for survey unit 8400 is intended as a future classification for FSS. Surveys conducted before the waste haul path and waste loadout structure are constructed may treat this survey unit as Class 3.

The Class 2 open land survey units are illustrated in Figure 2-3.

2.1.6.1.5 Class 3 Open Land Areas

Table 2-5 presents the open land areas have been initially classified as impacted Class 3. Historical information contained in the HSA indicated that the presence of residual radioactivity in concentrations in excess of the unrestricted release criteria was not expected.

Table 2-5 Class 3 Open Land Survey Units

Survey Unit ID No.	Survey Unit Description	Classification
8000	OCA Outside the DA	

Survey Unit ID No.	Survey Unit Description	Classification
8100	North Owner Controlled Areas	-
8101		3
8102		3
8103		3
8104		3
8105		3
8106		3
8107		3
8108		3
8109		3
8110		3
8200	West Owner Controlled Area	-
8201		3
8202		3
8203		3
8204		3
8205		3
8206		3
8207		3
8300	South Owner Controlled Area	-
8301		3
8302		3
8303		3
8304		3
8305		3
8306		3
8307		3
8308		3
8309		3
8310		3

The Class 3 open land survey units are illustrated in Figure 2-2.

2.2 Characterization Approach

Site characterization of the FCS was performed in accordance with [5], which was developed to provide guidance and direction to the personnel responsible for designing and implementing characterization survey activities. FCSD-RA-LT-200 works in conjunction with implementing procedures and survey unit specific survey instructions (sample plans) that were developed to safely and effectively acquire the requisite characterization data.

Characterization data acquired through the execution of the sample plan was used to meet three primary objectives:

- Provide radiological inputs necessary for the design of FSS.
- Develop the required inputs for the License Termination Plan (LTP) and dose assessments.
- Support the evaluation of remediation alternatives and technologies.

Characterization efforts focused on open land areas and remaining structures that will be subjected to FSS. Extensive characterization of equipment, systems, or structures that will be removed prior to the performance of FSS is not required in accordance with Appendix O of [4].

A significant study question that must be answered by the characterization is whether or not a survey unit is classified correctly. The appropriate classification of a survey unit is critical to the survey design for FSS. A classification which underestimates the potential for contamination could result in a survey design that does not obtain adequate information to demonstrate that the survey unit meets the release criteria. In some cases, this can increase the potential for making decision errors.

As site-specific DCGLs were not yet established for the FCS decommissioning at the time the characterization surveys were performed, alternate action levels were selected. The screening values presented in Tables H.1 and H.2 of [4] were selected as the alternate action levels.

As stated in the guidance, the models, scenarios, and parameters used to develop the screening values were intended to be conservative. The lack of information about a site warrants the use of conservative models and default conditions to ensure that the derived dose is not underestimated. Subsequently, use of screening values as action levels during characterization provided reasonable assurance that a survey unit was conservatively classified. The screening values used for soil characterization at FCS are reproduced below in Table 2-6.

Table 2-6 Screening Values for Soil

Radionuclide	Screening Value (pCi/g)	Radionuclide	Screening Value (pCi/g)
H-3	1.10E+02	Cs-137	1.10E+01
Fe-55	1.00E+04	Pu-238	2.50E+00
Co-60	3.80E+00	Pu-239/240	2.30E+00
Ni-63	2.10E+03	Pu-241	7.20E+01
Sr-90	1.70E+00	Am-241	2.10E+00
Cs-134	5.70E+00	Cm-243/244	3.20E+00

For structures, the gross screening level that was used during characterization as an action level to evaluate the classification of survey units was the nuclide-specific screening value of 7,100 dpm/100 cm^2 total gross beta-gamma surface activity, based on Co-60 from [4]. The use of the Co-60 screening value was conservative, as it was anticipated that the radionuclide distribution for surface contamination would be principally Co-60 and Cs-137.

The initial suite of potential radionuclides of concern (ROC) from FC-18-002, "Potential Radionuclides of Concern During the Decommissioning of Fort Calhoun Station" [13] was used during characterization. The final document that presents the ROC with mixture percentages and supersedes FC-18-002 is technical support document (TSD) 21-043, "Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan" [14].

2.2.1 Potential Radionuclides of Concern

TSD 21-043 establishes the basis for an initial suite of potential ROC for decommissioning. Based on the elimination of some of the theoretical neutron activation products, noble gases, and radionuclides with a half-life less than two years (with the exception of Ce-144), an initial suite of potential ROC for the decommissioning of the FCS was prepared. The initial suite of ROC is listed below in Table 2-7.

Table 2-7 Initial Suite of Radionuclides

Radionuclide	Half-Life (Years)	Radionuclide	Half-Life (Years)
Am-241	4.32E+02	Fe-55	2.70E+00
C-14	5.73E+03	H-3	1.24E+01
Ce-144	0.78E+00	Ni-59	7.50E+04
Cm-243	1.81E+01	Ni-63	9.60E+01
Cm-244	1.81E+01	Np-237	2.10E+06
Co-58	0.19E+00	Pu-238	8.78E+01
Co-60	5.27E+00	Pu-239	2.41E+04
Cs-134	2.00E+00	Pu-240	6.60E+03
Cs-137	3.00E+01	Pu-241	1.44E+01
Eu-152	1.33E+01	Sb-125	2.75E+03
Eu-154	8.80E+00	Sr-90	2.91E+01
Eu-155	4.76E+00	Tc-99	2.13E+05

2.2.2 Data Quality Objectives

DQOs were implemented for characterization surveys in a similar manner as anticipated for FSS. However, the goal of characterization is contamination quantification and delineation of the nuclide suite, whereas the FSS goal is comparison of data against the null hypothesis of the Sign test as defined by MARSSIM. Characterization inspections and surveys of sufficient quality and quantity were performed to determine the nature, extent, and range of radioactive contamination in each applicable survey unit, including applicable structures, residues, soils, and surface water.

The seven steps in the DQO development process are as follows:

1) State the problem.
2) Identify the decision.
3) Identify inputs to the decision.
4) Define the study boundaries.
5) Develop a decision rule.
6) Specify limits on decision errors.
7) Optimize the design for obtaining data.

The DQOs for site characterization included identifying the types and quantities of media to collect. Soils and structural concrete were sampled volumetrically. Sufficient measurements were obtained to achieve statistically significant results so that the mean and maximum activity, as well as the sample standard deviation, could be determined. Direct measurements and scans of structural surfaces and surface soils were also made using the same instruments and minimum detectable concentrations (MDC) as will be employed for FSS. Volumetric samples that exhibited the highest activity were sent to an off-site laboratory for analysis of hard-to-detect (HTD) radionuclides.

2.2.3 Survey Design

Characterization surveys were designed and performed in accordance with all applicable, approved procedures and FCSD-RA-LT-200, “Characterization Survey Plan.” Survey design incorporated a graded approach based upon the DQOs for each survey unit.

There are three approaches used for survey design: judgmental (biased), systematic, or random. Judgmental survey designs use known information or process knowledge to select locations for static measurements or samples. Systematic survey design, usually only utilized for FSS, selects static measurement or sample locations by using a systematic sampling design (typically a square or triangular grid) with a random start. Random survey design selects static measurements or sample locations randomly. The decision of whether to perform survey design using a judgmental, systematic, or random approach was addressed by the DQO process. A judgmental approach was warranted when the characterization effort was designed to delineate the extent of an area that requires remediation. Alternatively, a systematic or random approach was warranted if the characterization effort was designed to verify the basis for classification. Characterization surveys of Class 2 and Class 3 survey units utilize a combination of random and judgmental approaches to survey design. For Class 1 survey units, samples are primarily judgmental (as shown in Table 2-8).

2.2.3.1 Number of Static Measurements or Samples

The number of measurements or samples that were taken in each survey unit was determined by assessing the population size necessary to satisfy the DQOs.

For the characterization of Class 1 survey units, the number of static measurements or samples were of sufficient quantity to satisfy the DQO decision in the professional judgment of the responsible LT/FSS Specialist.

For the characterization of Class 2 survey units, the minimum number of systematic or random static measurements or samples taken in the survey unit were commensurate with the probability of the presence of residual radioactive contamination in the survey unit. The sample size selected was sufficiently robust to provide a statistically defensible mean and assessment of variability.

For Class 3 survey units, the primary characterization DQO is to establish the basis for the Class 3 or non-impacted classification. Consequently, the population of random measurements or samples was sufficiently robust so that the basis for the classification presented a high degree of confidence that no licensee-generated radioactive material resides in these areas. Since the recommended survey unit size for a Class 3 is unlimited, additional measurements or samples above the minimum population calculated may have be necessary to address this DQO.

Table 2-8 provides the recommended judgmental and random measurement or sample population sizes. These are recommended sample population sizes; the actual number of judgmental and random measurements or samples collected in a survey unit was at the discretion of the responsible LT/FSS Specialist.

Table 2-8 Recommended Sample Population Size

Classification	Population Type	Recommended Population Size
Class 1	Judgmental	8
	Random	Not required
Class 2	Judgmental	30
	Random	15
Class 3	Judgmental	13
	Random	14

2.2.3.2 Determination of Survey Locations

For impacted survey units, the location of measurements or samples to be taken in each survey unit (or group of survey units with similar geography and classification) was be determined by the responsible LT/FSS Specialist. For judgmental measurement or sample locations, consideration was be given to locations that exhibited measurable radioactivity (identified during the scan survey), depressions, discolored areas, cracks, low point gravity drain points, actual and potential spill locations, or areas where the ground had been disturbed. Historical information found in the HSA aided in judgmental location selection.

For Class 1 survey units, the location of measurements or samples was biased to suspect areas. For Class 2 and 3 survey units, the location of measurements or samples was chosen at random and augmented with biased measurements or samples, as necessary. Sample locations were determined by generating random pairs of coordinates that correspond to specific locations within a survey unit. This was accomplished through the use of a random number generator or through the use of the computer software Visual Sample Plan.

2.2.3.3 Scan Coverage

Survey units were scanned in accordance with their classification. The area scanned in each survey unit was determined by the professional judgment of the responsible LT/FSS Specialist during the survey design process. Table 10 is a list of recommended scan coverage guidelines that were used.

Table 2-9 Recommended Scan Coverage

Classification	Recommended Characterization Scan Coverage
Class 1	No scanning required unless compelled by a specific survey objective
Class 2	50% to 100%, concentrating on areas with an increased probability of exhibiting elevated activity (such as Class 1 boundaries, vehicle transit routes, etc.)
Class 3	5% to 50%, with emphasis on areas that were used for plant activities during operation and areas downwind or downstream of known effluent release points

2.2.4 Types of Measurements and Samples

The characterization survey of building or piping surfaces consisted of surface scans (beta and gamma), static beta measurements, material samples, and smears. The characterization survey of any concrete or asphalt-paved open land area consisted of surface scans (beta and gamma), static beta measurements, and volumetric samples. The characterization survey of the open land areas consisted of gamma scans and the sampling of surface and subsurface soil, sediment, and surface water for isotopic analysis. The following is a description of the different types of measurements and samples that were utilized.

2.2.4.1 Static Measurements

Static measurements were performed to detect direct levels of total surface contamination on structural surfaces of the buildings or on concrete or asphalt paved areas. These measurements were performed using a 126 cm^2 scintillation detector, the Ludlum Model 44-116.

Static measurements were conducted by placing the detector on or very near the surface to be counted and acquiring data over a pre-determined count time. A count time of one minute was typically used for surface measurements and generally provides detection levels well below the action level. Instrument count times could be adjusted, as appropriate, to achieve an acceptable MDC for static measurements. Adjustments to count times were controlled procedurally with management oversight.

2.2.4.2 Beta Surface Scans

Scanning was performed in order to locate areas of residual activity above the investigation level. Beta scans were performed over accessible structural surfaces including, but not limited to, floors, walls, ceilings, roofs, asphalt, and concrete paved areas. Hand-held beta scintillation detectors (i.e., Ludlum Model 44-116) were used for beta surface scans.

Beta scanning was performed with the detector position maintained within 1.27 cm (0.5 inch) of the surface and with a scanning speed of one detector active window per second. If surface conditions prevented scanning at the specified distance, the detection sensitivity for an alternate distance was determined, and the scanning technique adjusted accordingly. Scanning speed is calculated *a priori* to ensure the MDC for scanning was appropriate for the stated objective of the survey. Adjustments to scan speed and distance could be made when necessary. Adjustments to scan speed and distance were controlled procedurally with management oversight.

Technicians monitored the audible response of the survey instrument to identify locations of elevated activity that required further evaluation. All areas of elevated contamination located during scan surveys were identified for further investigation.

2.2.4.3 Gamma Surface Scans

Gamma scans were performed over open land surfaces to identify locations of residual surface activity. Sodium iodide (NaI) gamma scintillation detectors (i.e., Ludlum Model 44-10) were used for these scans.

Scanning was performed by moving the detector in a serpentine pattern, while advancing at a rate not to exceed 0.5 m (20 in) per second. The distance between the detector and the surface was maintained within 7.5 cm (3 in) of the surface. Audible signals were monitored, and locations of elevated direct levels were flagged for further investigation.

2.2.4.4 Removable Surface Contamination

If applicable, removable beta contamination or smear surveys were performed to verify that loose surface contamination was less than the action level. A 100 cm^2 surface area was wiped with a circular cloth or paper filter, using moderate pressure. Smears were then analyzed for the presence of gross beta and gross alpha activity. Counting was performed using a proportional counting system or equivalent.

2.2.4.5 Concrete Sampling

Volumetric sampling of concrete, as opposed to static measurements, was necessary when gross static measurements were not sufficient to address the survey unit specific DQOs. As an alternative to core boring, a patented procedure that uses a hollow drill bit was used to obtain exact volumes of concrete material at certain depths while utilizing a vacuum collection system. Material from each of the incremental depths at a location were captured in a separate container for each depth increment via use of the vacuum system.

2.2.4.6 Material Sampling

Samples of soil and sediment were obtained from designed judgmental, randomly selected, or systematic sample locations, as well as other biased locations in areas exhibiting elevated activity that were identified by scanning. Surface soil is usually defined as the top 15 cm (6 in) layer of soil, while subsurface soil is usually defined as soil below the top 15 cm layer in 1-meter increments. Surface soil was collected using a split spoon sampling system or by using hand trowels, bucket augers, or other suitable sampling tools.

Subsurface soil was sampled by direct push sampling systems (i.e., Geoprobe) or by the use of hand augers. Subsurface soil sampling was performed, as necessary, to address the DQOs for the survey unit.

An adequate amount of material (may range from 0.5 liters up to two liters) was collected at each location. Sample preparation included the removal of extraneous material and the homogenization and drying of the soil for analysis. Separate containers were used for each sample, and accountability for each container was present throughout the analysis process as specified in the QAPP. Samples were split as specified in the QAPP when required.

2.2.5 Instrument Selection, Use, and Minimum Detectable Concentrations

The radiation detection and measurement instrumentation for characterization was selected to provide both reliable operation and adequate sensitivity to detect the ROC identified for the decommissioning of the FCS at levels sufficiently below the established action levels. Detector selection was based on detection sensitivity, operating characteristics, and expected performance in the field. In all cases, the instruments and detectors selected for static measurements and analysis were capable of detecting the anticipated ROC at an MDC of 50% of the applicable action level.

Commercially available portable and laboratory instruments and detectors were used to perform the three basic survey measurements: (1) surface scanning, (2) static measurements, and (3) analysis of material samples.

Instrumentation and nominal MDC values that were employed during characterization are listed in Table 2-10 below. As the project proceeds, other measurement instruments or technologies, such as continuous data collection scan devices, may be added.

Table 2-10 Example of Instrument Types and Nominal MDC

Detector Model	Instrument Model[a]	Application	Nominal Detection Sensitivity[b]	
			MDC_{scan} (dpm/100cm^2)	MDC_{static} (dpm/100cm^2)
Ludlum 44-9	Ludlum 2350-1	β static & scan	2900	985
Ludlum 43-5	Ludlum 2350-1	α static & scan	150	75

Detector Model	Instrument Model[a]	Application	Nominal Detection Sensitivity[b]	
			MDC_{scan} (dpm/100cm^2)	MDC_{static} (dpm/100cm^2)
Ludlum 43-68 β mode	Ludlum 2350-1	β static & scan	1050	330
Ludlum 43-68 α mode	Ludlum 2350-1	α static & scan	170	70
Ludlum 44-116	Ludlum 2350-1	β static & scan	1300	415
Ludlum 43-90	Ludlum 2350-1	α static & scan	130	55
Ludlum 44-10	Ludlum 2350-1	γ scan	3.5 pCi/g Co-60 6.5 pCi/g Cs-137	N/A
Ludlum 43-37	Ludlum 2350-1	β scan	1000	N/A
Tennelec LB5100 proportional counting system	N/A	α and/or β smear	N/A	18
HPGe gamma spectroscopy system	N/A	γ Analysis	N/A	~0.15 pCi/g for Co-60 and Cs-137

(a) Functional instrument equivalent may be used (e.g., Ludlum 2221 or Ludlum 3001).
(b) Based on 1-minute count time and default values for surface efficiencies, $\in_s$, as specified in International Standard ISO 7503-1, Part 1, “Evaluation of Surface Contamination, Beta-Emitters (maximum beta energy greater than 0.15 MeV) and Alpha-Emitters” [15]

2.2.5.1 Instrument Calibrations

All data loggers, associated detectors, and all other portable instrumentation that were used for characterization were calibrated on an annual basis using National Institute of Standards and Technology (NIST) traceable sources. The calibration of instruments used for characterization is addressed in Section 4.7 of the QAPP.

2.2.5.2 Instrument Use and Control

The receipt, inspection, issue, controls, and accountability of portable radiological instrumentation used for characterization is performed in accordance with the procedure that governs the issue, control, and accountability of characterization and FSS portable radiological instrumentation. The issue and control of instruments used for characterization is addressed in Section 4.6 of the QAPP.

2.2.5.3 Laboratory Instrument Methods and Sensitivities

Gamma spectroscopy was primarily performed by the on-site radiological laboratory. Gas proportional counting and liquid scintillation analysis was performed by an approved vendor laboratory, GEL Laboratories, in accordance with approved laboratory procedures. Energy*Solutions* ensured that quality programs of the contracted off-site vendor laboratories that were used for the receipt, preparation, and analysis of characterization samples provided the same level of quality as the on-site laboratory under the QAPP.

In all cases, analytical methods were established to ensure that required MDC values were achieved. The analysis of radiological contaminants used standard approved and generally accepted methodologies or other comparable methodologies. Table 2-11 below provides the analytical methods employed and the typical laboratory MDCs achieved by GEL Laboratories.

Table 2-11 GEL Laboratories Methods, MDCs, and Reporting Limits

Test	Technique	Method	Typical MDC (pCi/g)	Reporting Limit (pCi/g)
Gamma	Gamma Spectroscopy	DOE HASL 300, 4.5.2.3/Ga-01-R	1.01	1
H-3	Liquid Scintillation	EPA 906.0 Modified	7.88	10
C-14	Liquid Scintillation	EPA EERF C-01 Modified	3.53	5
Fe-55	Liquid Scintillation	DOE RESL Fe-1 Modified	6.54	10
Ni-59	Low Energy Gamma Spectroscopy	DOE RESL Ni-1	2.14	5
Ni-63	Liquid Scintillation	DOE RESL Ni-1 Modified	2.96	5
Sr-90	Gas Flow Proportional	EPA 905.0 Modified/DOE RP501 Rev. 1 Modified	0.15	0.4
Tc-99	Liquid Scintillation	DOE EML HASL-300, Tc-02-RC Modified	1.34	2
Np-237	Alpha Spectroscopy	ASTM C 1475-00 Modified	0.01	0.01
Pu-238	Alpha Spectroscopy	DOE EML HASL-300, Pu-11-RC Modified	0.06	0.4
Pu-239/240	Alpha Spectroscopy	DOE EML HASL-300, Pu-11-RC Modified	0.06	0.4
Pu-241	Liquid Scintillation	DOE EML HASL-300, Pu-11-RC Modified	4.28	5
Am-241	Alpha Spectroscopy	DOE EML HASL-300, Am-05-RC Modified	0.06	1

Test	Technique	Method	Typical MDC (pCi/g)	Reporting Limit (pCi/g)
Cm-242	Alpha Spectroscopy	DOE EML HASL-300, Am-05-RC Modified	0.05	1
Cm-243/244	Alpha Spectroscopy	DOE EML HASL-300, Am-05-RC Modified	0.06	1

2.2.6 Quality Assurance

Section 2.2 of MARSSIM discusses the need for a quality system to ensure the adequacy of data used to demonstrate that site conditions are acceptable for unrestricted release. Laboratory quality for sample analyses taken to support characterization and FSS is discussed in NUREG-1576, "Multi-Agency Radiological Laboratory analytical Protocols Manual" (MARLAP) [16] and Regulatory Guide 4.15, "Quality Assurance of Radiological Monitoring Programs (Inception through Normal Operations to License Termination) – Effluent Streams and the Environment" [17]. Further, MARSSIM and MARLAP both indicate that a QAPP may be used in addition to, or in lieu of, existing quality systems to ensure data quality is achieved.

The QAPP was prepared and implemented to ensure the adequacy of data being developed and used during the site characterization and FSS process. The QAPP describes policy, organization, functional activities, the DQO process, and measures necessary to achieve quality data. It supplements the quality requirements and quality concepts presented in NO-FC-10, "Quality Assurance Topical Report" [18],which adequately encompass other risk-significant decommissioning activities.

All characterization activities essential to data quality were implemented and performed using approved procedures. The effective implementation of characterization was verified through audit and surveillance activities, including field walk-downs by LT/FSS management and program self-assessments, as appropriate. Corrective actions were prescribed, implemented, and verified when deficiencies were identified. These measures applied to any applicable services provided by off-site vendors, as well as on-site subcontractors.

FCSD-RA-LT-200, "Characterization Survey Plan," was developed according to the essential elements of the quality assurance and quality control (QA/QC) program for the decommissioning of FCS and is subject to the QAPP. The QA/QC program elements applicable to characterization are as follows:

- establishment and implementation of plans, procedures, and protocols for field operations
- actions to ensure that the procedures are understood and followed by the implementing staff
- documentation of the data collected

Details of the QA/QC elements specific to characterization are presented in the QAPP as well as the procedures and sample plan instructions. Characterization operations and the associated data acquisition and recording was guided and conducted in compliance with these QA/QC requirements. The specific QA/QC program components for site characterization are as follows:

- personnel qualifications, experience, and training
- execution in accordance with approved procedures
- proper documentation of survey data and sample analyses
- selection of appropriate instruments to perform the surveys
- proper instrument calibration and daily functional checks
- management oversight of characterization activities relative to the adherence to procedures, protocols, and documentation requirements

2.3 Summary of Characterization Survey Results

Site characterization surveys commenced in November 2019 with the concrete sampling of the structures inside of the DA. In January 2020, characterization of the open land survey units commenced. Throughout 2019 and 2020, characterization activities were performed in parallel with dry fuel storage, radioactive commodity removal, and radioactive waste shipment activities at FCS. Consequently, access to certain areas to collect meaningful characterization survey data was prohibited. In these cases, characterization has been deferred until such time that radiological or physical conditions would allow for substantial survey. Characterization surveys ("continuing characterization") will continue throughout the decommissioning process. Section 2.5 provides more details on the planned continuing characterization.

The characterization results presented in this chapter were taken from FC-20-012, "Fort Calhoun Station Decommissioning Project Radiological Characterization Report" [19].

2.3.1 Volumetric Concrete Sampling

In November and December of 2019, Energy*Solutions* contracted New Millennium Nuclear Technologies International, Inc. (NMNT International) to collect volumetric concrete samples using their proprietary TruPro technology, which utilizes hollow-bit drilling as an effective alternative to traditional core boring. All of the buildings inside of the DA (see Table 2-1 and Table 2-2) were subject to volumetric concrete characterization, with primary focus on the Containment and Auxiliary Buildings. In total, 744 concrete samples were analyzed using the on-site gamma spectroscopy system. Table 2-12 and Table 2-13 summarize the locations and numbers of concrete samples collected for site characterization.

The locations selected for the concrete sampling were biased towards areas with elevated dose rates, count rates, proximity to radiological components, or by visual observations of floor and wall surfaces that indicated potential contamination (e.g., discoloration or standing water). The goal was to identify, to the extent possible, the locations that exhibited the highest potential of contamination representing the worst case bounding radiological condition for concrete. This judgmental sampling approach also ensured there was sufficient source term in the samples to achieve the sensitivities required to determine the radionuclide distributions of gamma emitters as well as HTD radionuclides.

Using the TruPro technology, NMNT International was able to collect a total of 810 samples from 178 locations. Specific locations were identified for sampling to 1.5-inch depth or 6-inch depth. For a location specified to be 1.5 inches in depth, three samples increments (each representing 0.5 inches in depth down to 1.5 inches) were collected. For a location specified to be 6 inches in depth, 6 sample increments (the first four were 0.5, 1, 1.5, and 2 inches deep, then the next two were 4 and 6 inches deep) were collected. The tubes, drill bits, and sample containers were changed out for each new increment obtained, ensuring that no cross-contamination occurred.

Sample increments were counted starting with the top-most increment and increments of a sample location were analyzed until all ROC were less than MDC. Because this was the case, only 744 of the total 810 concrete samples collected were counted by the on-site gamma spectroscopy system.

In May 2020, traditional concrete core samples were acquired in the Containment Building exterior concrete on the 1,045 feet above mean sea level (AMSL) elevation. These samples were collected to investigate the positive identifications of plant-derived radionuclides at depth in concrete on the exterior side of the steel liner that was expected to have no contamination. The conclusion was reached that the original samples, collected utilizing the TruPro methodology, were inadvertently cross-contaminated, as shown by the gamma spectroscopy results of the traditional concrete core samples.

2.3.1.1 Volumetric Concrete Sampling Results in the Containment Building

Surveys for the sampling of volumetric concrete in the Containment Building, survey unit 1000, were performed under characterization sample plan 1000-C.

In the 977 feet, 996 feet, 1,013 feet, and 1,045 feet AMSL elevations floors and walls of the Containment Building, 429 concrete samples from 71 locations were collected. The locations where the samples were collected are depicted in Figure 2-7 through Figure 2-10. All 429 samples collected were analyzed using the on-site gamma spectroscopy system. A summary of the gamma spectroscopy results are provided in Table 2-14 through Table 2-21. The plant-derived gamma-emitting radionuclides identified in the on-site analyses of the concrete samples were Co-60, Cs-134, Cs-137, Eu-152, Eu-154, Eu-155, and Am-241.

Seventeen concrete samples from the Containment Building were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-47. Significant HTD radionuclides identified by the analysis of the concrete samples include H-3, C-14, Ni-63, and Sr-90.

2.3.1.1.1 Concrete Core Sampling Results in the Containment Building Exterior

As stated in Section 2.3.1, traditional concrete core samples were collected in the Containment Building exterior concrete to verify cross-contamination of the original samples. The concrete core samples were collected under an addendum to sample plan 1000-C.

Three 6-inch deep cores, utilizing a 2-inch diameter bit, were collected from sample locations 6, 9, and 20. The locations where the samples were collected are depicted in Figure 2-10. The top ½-inch disc from each core was analyzed using the on-site gamma spectroscopy system.

Summaries of the gamma spectroscopy results are provided in Table 2-22 and Table 2-23. No plant-derived ROC were identified in the on-site analysis of the concrete core samples.

It was concluded that the samples from locations 1, 6, 9, 19, 20, and 26 were cross-contaminated, and the data presented in Table 2-20 and Table 2-21 for the six locations is not considered valid data. Additional concrete samples of the exterior of the Containment Building will be collected during continuing characterization to further validate that the exterior concrete is uncontaminated. If plant-related radionuclides are identified above background, the concrete will be controlled as radioactive material and disposed of appropriately.

2.3.1.2 Volumetric Concrete Sampling Results in the Auxiliary Building

Surveys for the sampling of volumetric concrete in the Auxiliary Building, survey unit 2000, were performed under characterization sample plan 2000-C.

In the 971 feet, 989 feet, 1,007 feet, and 1,025 feet AMSL elevations floors and walls of the Auxiliary Building, 261 concrete samples from 67 locations were collected. The locations where the samples were collected are depicted in Figure 2-11 through Figure 2-14. All 261 samples collected were analyzed using the on-site gamma spectroscopy system. A summary of the gamma spectroscopy results are provided in Table 2-24 through Table 2-31. The plant-derived gamma-emitting radionuclides identified in the on-site analyses of the concrete samples were Co-60, Cs-134, and Cs-137.

Sixteen concrete samples from the Auxiliary Building were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-47. Significant HTD radionuclides identified by the analysis of the concrete samples include Ni-63, Sr-90, and H-3.

2.3.1.3 Volumetric Concrete Sampling Results in the Turbine Building

Surveys for the sampling of volumetric concrete in the Turbine Building, survey unit 3000, were performed under characterization sample plan 3000-C.

In the 990 feet AMSL elevation floor and walls of the Turbine Building, 54 concrete samples from 18 locations were collected. The locations where the samples were collected are depicted in Figure 2-15. Of the 54 samples collected, 18 were analyzed using the on-site gamma spectroscopy system. A summary of the gamma spectroscopy results are provided in Table 2-32 and Table 2-33. No plant-derived gamma-emitting radionuclides were identified in the on-site analyses of the concrete samples for the Turbine Building.

Two concrete samples from the Turbine Building were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-47. No HTD radionuclides were identified in the off-site analyses of concrete samples from the Turbine Building.

2.3.1.4 Volumetric Concrete Sampling Results in the BOP Buildings Inside the DA

Surveys for the sampling of volumetric concrete in the BOP buildings inside the DA, survey unit 4000, were performed under characterization sample plan 4000-C.

In the floors and walls of the BOP buildings inside the DA, 66 concrete samples from 22 locations were taken. The locations where the samples were collected are depicted in Figure 2-16 through Figure 2-25. Of the 66 samples collected, 36 were analyzed using the on-site gamma spectroscopy system. Summaries of the gamma spectroscopy results are provided in Table 2-34 through Table 2-46. The plant-derived gamma-emitting radionuclides identified in the on-site analyses of the concrete samples were Co-60 and Cs-137.

Two concrete samples from the BOP buildings inside the DA were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-47. Significant HTD radionuclides identified by the analysis of the concrete samples include Sr-90.

2.3.2 Open Land Area Soil Sampling and Scanning

Each surface and subsurface soil sample was analyzed by the on-site gamma spectroscopy system. Analysis count times were adjusted as necessary to achieve an MDC equal to or less than 0.5 pCi/g for Cs-137 and Co-60. In accordance with procedure, a minimum of 10 percent of the soil samples were sent off-site to GEL Laboratories for full-suite ROC analysis.

2.3.2.1 Survey Results for Class 3 Open Land Survey Units

The Class 3 open land areas at FCS total 2,102,462 m^2 of surface area. The Class 3 open land surface area was broken into three survey units in accordance with the area descriptions, sizes, and boundaries presented in FCSD-RA-LT-200, “Characterization Survey Plan.” The Class 3 open land survey units are illustrated by Figure 2-2.

2.3.2.1.1 Survey Unit 8100

Class 3 open land survey unit 8100 is located in the northern area of the site, with a total size of 867,259 m^2. Additionally, survey unit 8100 is broken up into 10 smaller sub-units (8101-8110), mainly consisting of farmland, wetlands, and a creek, and houses the Firing Range, Storage Shed, and the Switchyard.

The survey design for survey unit 8100 called for the acquisition of 39 judgmental surface soil samples and 42 random surface soil samples. Approximately 5 percent of the total surface area in the survey unit, 43,363 m^2, was scanned using a Ludlum Model 2221 instrument coupled to a Ludlum Model 44-10 NaI detector.

For the area scanned, the average observed background in the survey unit was 9,057 cpm. The average minimum observed scan measurement was 7,799 cpm, and the average maximum observed scan measurement was 10,616 cpm. The average number of points greater than the action level was 13. After investigating areas that were above the action level, one judgmental surface soil sample was collected.

In total, 82 surface soil samples and 10 QC split or duplicate samples were acquired and analyzed by the on-site gamma spectroscopy system. Thirty-five samples were above MDC for Cs-137, with a maximum concentration of 3.16E-01 pCi/g. No other ROC were identified. The maximum sum of fractions (SOF) was 0.0993.

The locations of scan areas and surface soil samples collected in survey unit 8100 are illustrated on Figure 2-26 and Figure 2-27. The results of surface soil sample analyses are presented in Table 2-48 through Table 2-51.

Seven soil samples from survey unit 8100 were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site samples.

2.3.2.1.2 ***Survey Unit 8200***

Class 3 open land survey unit 8200 is located in the western area of the site, with a total size of 470,118 m^2. Additionally, survey unit 8200 is broken up into seven smaller sub-units (8201-8207), mainly consisting of farmland, and houses the FLEX Building and a portion of the site access road.

The survey design for survey unit 8200 called for the acquisition of 26 judgmental surface soil samples 28 random surface soil samples. Approximately 5 percent of the total surface area in the survey unit, 23,506 m^2, was scanned using a Ludlum Model 2221 instrument coupled to a Ludlum Model 44-10 NaI detector.

For the area scanned, the average observed background in the survey unit was 9,901 cpm. The average minimum observed scan measurement was 9,022 cpm, and the average maximum observed scan measurement was 11,086 cpm. The average number of points greater than the action level was seven. After investigating areas that were above the action level, no area was above the alarm set point and thus no additional samples were collected.

In total, 54 surface soil samples and 6 QC split or duplicate samples were acquired and analyzed by the on-site gamma spectroscopy system. Fourteen samples were above MDC for Cs-137, with a maximum concentration of 2.11E-01 pCi/g. No other ROC were identified. The maximum SOF was 0.0594.

The locations of scan areas and surface soil samples collected in survey unit 8200 are illustrated on Figure 2-28 and Figure 2-29. The results of surface soil sample analyses are presented in Table 2-52 through Table 2-55.

Six soil samples from survey unit 8200 were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site samples.

2.3.2.1.3 ***Survey Unit 8300***

Class 3 open land survey unit 8300 is located in the southern area of the site, with a total size of 765,085 m^2. Additionally, survey unit 8300 is broken up into 10 smaller sub-units (8301-8310), mainly consisting of farmland, and houses the Administration Building, the Training Center, and the parking lots.

The survey design for survey unit 8300 called for the acquisition of 39 judgmental surface soil samples and 42 random surface soil samples. Approximately 5 percent of the total surface area in the survey unit, 38,254 m^2, was scanned using a Ludlum Model 2221 instrument coupled to a Ludlum Model 44-10 NaI detector.

For the area scanned, the average observed background in the survey unit was 9,569 cpm. The average minimum observed scan measurement was 8,379 cpm, and the average maximum observed scan measurement was 10,787 cpm. The average number of points greater than the action level was six. After investigating areas that were above the action level, nine judgmental surface soil samples were collected.

In total, 93 surface soil samples and 10 QC split or duplicate samples were acquired and analyzed by the on-site gamma spectroscopy system. Forty-eight samples were above MDC for Cs-137, with a maximum concentration of 3.27E-01 pCi/g. No other ROC were identified. The maximum SOF was 0.0980.

The locations of scan areas and surface soil samples collected in survey unit 8300 are illustrated on Figure 2-30 and Figure 2-31. The results of surface soil sample analyses are presented in Table 2-56 through Table 2-59.

Nine soil samples from survey unit 8300 were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site samples.

2.3.2.2 Survey Results for Class 2 Open Land Survey Units

The Class 2 open land areas at FCS total 92,012 m^2 of surface area. The Class 2 open land surface area was broken into two survey units in accordance with the area descriptions, sizes, and boundaries presented in the FCSD-RA-LT-200, "Characterization Survey Plan." The Class 2 open land survey units are illustrated by Figure 2-3.

2.3.2.2.1 Survey Unit 8400

Class 2 open land survey unit 8400 is designated as the Waste Haul Path Buffer Zone, with a total size of 61,133 m^2. Additionally, survey unit 8400 is broken up into six smaller sub-units (8401-8406) and houses the New Maintenance Storage Shed and the Mausoleum (OSGSF).

The survey design for survey unit 8400 called for the acquisition of 30 judgmental surface soil samples and 15 random surface soil samples. Approximately 50 percent of the total surface area in the survey unit, 30,567 m^2, was scanned using a Ludlum Model 2221 instrument coupled to a Ludlum Model 44-10 NaI detector.

For the area scanned, the average observed background in the survey unit was 8,923 cpm. The average minimum observed scan measurement was 7,501 cpm, and the average maximum observed scan measurement was 10,243 cpm. The average number of points greater than the action level was 11. After investigating areas that were above the action level, two judgmental surface soil samples were collected.

In total, 33 surface soil samples and 4 QC split or duplicate samples were acquired and analyzed by the on-site gamma spectroscopy system. Four samples were above MDC for Cs-137, with a maximum concentration of 2.49E-01 pCi/g. No other ROC were identified. The maximum SOF was 0.0826.

The locations of scan areas and surface soil samples collected in survey unit 8400 are illustrated on Figure 2-32 and Figure 2-33. The results of surface soil sample analyses are presented in Table 2-60 through Table 2-63.

Three soil samples from survey unit 8400 were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site samples.

2.3.2.2.2 __Survey Unit 8700__

Class 2 open land survey unit 8700 is designated as the Sewage Lagoons, with a total size of 30,879 m^2. Additionally, survey unit 8700 is broken up into three smaller sub-units (8701-8703) in the southeast corner of the site.

The survey design for survey unit 8700 called for the acquisition of 6 surface soil and 2 sediment samples taken at judgmental locations, and 15 random surface soil samples. 100 percent of the accessible land in the survey unit was scanned, equaling 30,879 m^2. This area was scanned using a Ludlum Model 2221 instrument coupled to a Ludlum Model 44-10 NaI detector. The portions of the survey unit covered in lagoon water were deemed inaccessible for scanning.

For the area scanned, the average observed background in the survey unit was 8,146 cpm. The average minimum observed scan measurement was 7,031 cpm, and the average maximum observed scan measurement was 9,044 cpm. The average number of points greater than the action level was one. After investigating areas that were above the action level, one judgmental surface soil sample was taken.

In total, 23 surface soil samples, 2 sediment samples, and 4 QC split or duplicate samples were acquired and analyzed by the on-site gamma spectroscopy system. One sample was above MDC for Cs-137, with a concentration of 1.42E-02 pCi/g. No other ROC were identified. The maximum SOF was 0.0433.

The locations of scan areas and surface soil samples collected in survey unit 8700 are illustrated on Figure 2-34 and Figure 2-35. The results of surface soil sample analyses are presented in Table 2-64 through Table 2-67.

Two soil samples from survey unit 8700 were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site samples.

2.3.2.3 Survey Results for Class 1 Open Land Survey Units

The Class 1 open land areas at FCS total 36,868 m^2 of surface area. The Class 1 open land surface area was broken into five survey units in accordance with the area descriptions, sizes, and boundaries presented in the FCSD-RA-LT-200, “Characterization Survey Plan.” The Class 1 open land survey units are illustrated by Figures 3 and 4. Class 1 survey units 7500, 8500, and 8600 were not subject to initial characterization due to inaccessibility (survey unit 7500 houses the main plant buildings) or decommissioning activities (construction of the waste haul path and waste containment structure).

2.3.2.3.1 <u>Survey Unit 7100</u>

Class 1 open land survey unit 7100 is located in the northwest corner inside the DA fence, with a total size of 9,429 m^2. Additionally, survey unit 7100 is broken up into five smaller sub-units (7101-7105).

The survey design for survey unit 7100 called for the acquisition of eight surface soil and two subsurface soil samples taken at judgmental locations. Surface scanning was not required for the characterization of this Class 1 survey unit.

In total, 10 soil samples and 2 QC split or duplicate samples were acquired and analyzed by the on-site gamma spectroscopy system. One sample was above MDC for Cs-137, with a concentration of 7.77E-02 pCi/g. No other ROC were identified. The maximum SOF was 0.0529.

The locations of the soil samples collected in survey unit 7100 are illustrated on Figure 2-36. The results of the soil sample analyses are presented in Table 2-68 through Table 2-70.

One (1) soil sample from survey unit 7100 was sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site sample.

2.3.2.3.2 <u>Survey Unit 7200</u>

Class 1 open land survey unit 7200 is located in the southwest corner inside the DA fence, with a total size of 8,906 m^2. Additionally, survey unit 7200 is broken up into five smaller sub-units (7201-7205).

The survey design for survey unit 7200 called for the acquisition of eight surface soil and two subsurface soil samples taken at judgmental locations. Surface scanning was not required for the characterization of this Class 1 survey unit.

In total, 10 soil samples and 2 QC split or duplicate samples were acquired and analyzed by the on-site gamma spectroscopy system. No ROC greater than their respective MDCs were detected in any sample in the survey unit. The maximum SOF was 0.0529.

The locations of the soil samples collected in survey unit 7200 are illustrated on Figure 2-36. The results of the soil sample analyses are presented in Table 2-71 through Table 2-73.

One soil sample from survey unit 7200 was sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site sample.

2.3.2.3.3 <u>Survey Unit 7300</u>

Class 1 open land survey unit 7300 is located in the southeast corner inside the DA fence, with a total size of 6,686 m^2. Additionally, survey unit 7300 is broken up into four smaller sub-units (7301-7304).

The survey design for survey unit 7300 called for the acquisition of eight surface soil and two subsurface soil samples taken at judgmental locations. Surface scanning was not required for the characterization of this Class 1 survey unit.

In total, 10 soil samples and 2 QC split or duplicate samples were acquired and analyzed by the on-site gamma spectroscopy system. No ROC greater than their respective MDCs were detected in any sample in the survey unit. The maximum SOF was 0.0610.

The locations of the soil samples collected in survey unit 7300 are illustrated on Figure 2-36. The results of the soil sample analyses are presented in Table 2-74 through Table 2-76.

One soil sample from survey unit 7300 was sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site sample.

2.3.2.3.4 Survey Unit 7400

Class 1 open land survey unit 7400 is located in the northeast corner inside the DA fence, with a total size of 11,847 m^2. Additionally, survey unit 7400 is broken up into six smaller sub-units (7401-7406).

The survey design for survey unit 7400 called for the acquisition of eight surface soil and two subsurface soil samples taken at judgmental locations. Surface scanning was not required for the characterization of this Class 1 survey unit.

In total, 10 soil samples and 2 QC split or duplicate samples were acquired and analyzed by the on-site gamma spectroscopy system. No ROC greater than their respective MDCs were detected in any sample in the survey unit. The maximum SOF was 0.0472.

The locations of the soil samples taken in survey unit 7400 are illustrated on Figure 2-36. The results of the soil sample analyses are presented in Table 2-77 through Table 2-79.

One soil sample from survey unit 7400 was sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site sample.

2.3.3 Subsurface Soil Sampling with GeoProbe

Surveys for the sampling of subsurface soil within the DA were performed under characterization sample plan 7000-C.

The survey design called for the collection of 75 vertical subsurface soil samples at 25 locations, with samples to be collected in 1-meter increments to a depth of 3 meters at each location. Additionally, the survey design called for the collection of 12 diagonal subsurface soil samples at 6 locations, with 2 samples to be collected in 1-meter increments from the 2 deepest depths at each location. The purpose of the diagonal borings was to assess the sub-slab soils beneath the Turbine, Auxiliary, and Containment Buildings. A total of 87 subsurface soil samples were planned for collection in the survey unit.

In reality, 82 subsurface soil samples were acquired and analyzed by the on-site gamma spectroscopy system. Four diagonal locations met refusal and were unable to reach the sub-slab soil for the Containment and Turbine Buildings; therefore, those soils will be assessed during continuing characterization. Three samples were above MDC for Cs-137, with a maximum concentration of 1.15E-01 pCi/g. No other ROC were identified. The maximum SOF was 0.1142.

The locations of the subsurface soil samples collected in survey unit 7000 are illustrated on Figure 2-37. The results of the subsurface soil sample analyses are presented in Table 2-80 and Table 2-81.

Eight subsurface soil samples from survey unit 7000 were sent to GEL Laboratories for off-site gamma spectroscopy and HTD radionuclide analyses. The results of the analyses are presented in Table 2-82. Only Cs-137 was positively identified in the off-site samples.

2.3.4 Survey Results for Class 2 and 3 Structures

2.3.4.1 Class 3 Structures Inside the DA

2.3.4.1.1 Turbine Building

Sample plan 3000-C #2 was designed for the surveying of the Turbine Building due to its size and potential for contamination. The Turbine Building is a Class 3 structure with a total accessible surface area of 9,397 m^2.

The survey design for the Turbine Building called for the acquisition of 42 judgmental static measurements, 42 random static measurements, and a smear at each static measurement location. Four tiles were also acquired as judgmental samples. 639 m^2, or approximately 7 percent of the total surface area in the survey unit, was scanned using a Ludlum Model 2350-1 instrument coupled to a Ludlum Model 44-116 detector. Additionally, as qualitative measurements, each floor drain opening in the 990 feet AMSL elevation was subjected to a 1-minute static measurement with a Ludlum 2350-1 and 44-10 probe.

For the area scanned, the average observed background in the survey unit was 318 cpm. The average observed scan measurement was 504 cpm, and the maximum observed scan measurement was 922 cpm. Four alarms were produced during scanning, and static measurements were taken at alarm locations. All static measurements collected at scan alarm locations were below the alarm set point. When scanning, readings are obtained at one-second intervals, and if one reading is above the alarm set point the alarm is triggered. When taking a static measurement, the readings are averaged over a one-minute time period, and typically the average reading is below the alarm set point. This conservative methodology ensures that elevated areas of contamination are not missed. The gamma scans of the floor drains were all indistinguishable from background.

Eighty-seven random and judgmental static measurements were collected in the survey unit. Additionally, 25 judgmental static measurements were acquired due to the abundance of exposed system internals and externals available in the survey unit. In total, 112 static measurements were collected. None of the static measurements exceeded 50 percent of the interim screening value, 7,100 dpm/100cm^2. The four tiles were analyzed by the on-site gamma spectroscopy system for radionuclide identification only, due to the geometry. No ROC were identified for any of the tiles. For smears, the maximum observed beta activity was 31 dpm/100cm^2.

The locations of scan areas and static measurements collected in the survey unit are illustrated on Figure 2-38 through Figure 2-42. The summary of the measurement results are presented in Table 2-83 through Table 2-86.

2.3.4.1.2 <u>The Balance of Plant Buildings Inside the DA</u>

Sample plan 4000-C #2 was designed for the surveying of the BOP buildings inside of the DA. The following is a list of Class 3 structures surveyed under this sample plan:

- Intake Building
- Security Building
- SAF
- Service Building
- Maintenance Shop
- TSC
- Chemistry and Radiation Protection Facility
- New Warehouse

The survey design called for the acquisition of 20 judgmental static measurements and a smear and 1 m^2 scan at each static location for each building, with the exception of the Intake Building. The Intake Building was subjected to a full characterization, while the remaining buildings were subjected to limited characterization due to being slated for unconditional release survey (URS).

For the Intake Building, the survey design called for the acquisition of 13 judgmental static measurements 14 random static measurements, and a smear at every static measurement location. 95 m^2, or approximately 5 percent of the total available surface area in the Intake Building, was scanned using a Ludlum Model 2350-1 instrument coupled to a Ludlum Model 44-116 detector.

For the area scanned, the average observed background in the survey unit was 337 cpm. The average observed scan measurement was 493 cpm, and the maximum observed scan measurement was 915 cpm. The scans did not identify any areas of elevated activity.

Fourteen random and 159 judgmental static measurements were collected, for a total of 173 static measurements. None of the static measurements exceeded 50 percent of the interim screening value, 7,100 dpm/100cm^2. For smears, the maximum observed beta activity was 75 dpm/100cm^2.

The locations of scan areas and static measurements collected in the survey unit are illustrated on Figure 2-43 through Figure 2-60. The summary of the measurement results are presented in Table 2-87 through Table 2-89.

2.3.4.2 Class 2 and 3 Structures Outside the DA

Sample plan 5000-C was designed for the surveying of the BOP buildings outside of the DA. The only Class 2 structure under this sample plan was the Mausoleum, which ultimately was not surveyed due to the presence of steam generators in the building at the time of survey. The following is a list of the Class 3 structures surveyed under this sample plan:

- Administrative Office Building
- Training Center

- FLEX Building
- New Maintenance Storage Shed
- Chemical Pump House
- Chemical Storage Shed
- Sanitary Lift Stations

The survey design called for the acquisition of 20 judgmental static measurements and a smear and 1 m^2 scan at each static location for each building, with the exception of the Administrative Office Building, Training Center, and the Sanitary Lift Stations. The Administrative Office Building and the Training Center were subjected to a full characterization, while the remaining buildings were subjected to limited characterization due to being slated for URS.

For the Administrative Office Building, the survey design called for the acquisition of 13 judgmental static measurements, 28 random static measurements, and a smear at every static measurement location.

For the Training Center, the survey design called for the acquisition of 13 judgmental static measurements, 28 random static measurements, and a smear at every static measurement location.

For the Sanitary Lift Stations, the survey design called for the acquisition of six judgmental static measurements and a smear at every static location.

The survey unit was scanned utilizing a Model 2350-1 coupled to a Model 44-116 detector. For the Administrative Office Building, a minimum of 5 percent of the building was scanned. This was achieved by scanning 29 m^2 around each of the judgmental and random static locations. For the Training Center, a minimum of 5 percent of the building was scanned. This was achieved by scanning 21 m^2 around each of the judgmental and random static locations. For all other structures, a 1 m^2 area around each judgmental measurement location was scanned. For the area scanned, the average observed background in the survey unit was 311 cpm. The average observed scan measurement was 462 cpm, and the maximum observed scan measurement was 764 cpm. The scans did not identify any areas of elevated activity.

Fifty-six random and 112 judgmental static measurements were collected, for a total of 168 static measurements. None of the static measurements exceeded 50 percent of the interim screening value, 7,100 dpm/100cm^2. For smears, the maximum observed beta activity was 52 dpm/100cm^2.

The locations of scan areas and static measurements collected in the survey unit are illustrated on Figure 2-61 through Figure 2-70. The summary of the scan results are presented in Table 2-90 through Table 2-92.

2.4 Surface and Groundwater

Chapter 1 and Chapter 8 of this LTP contain summary descriptions of the geology, hydrogeology, and hydrology of FCS and the environs.

2.4.1 On-Going Investigations

On-going monitoring of surface water and groundwater at FCS includes the Radiological Environmental Monitoring Program (REMP), the Radiological Groundwater Protection Program (RGPP) per NEI 07-07, and monitoring under the Nebraska Industrial Stormwater General Permit and National Pollutant Discharge Elimination System (NPDES) permit.

2.4.1.1 REMP

The REMP is a requirement of the FCS "Quality Assurance Topical Report" and was initiated prior to plant operation in 1973. The main purpose of the REMP is to ensure public safety by monitoring plant discharges and assessing the effect, if any, of plant operations on the environment. Samples are collected that would account for various exposure pathways such as ingestion, inhalation, adsorption, and direct exposure. Samples collected on a regular basis include air, surface water, groundwater, milk, vegetation, fish, sediment, and food crops. Direct radiation is measured by thermoluminescent dosimeters (TLDs). These samples and TLDs are sent to an independent vendor laboratory for analysis. The vendor uses analytical methods that are sensitive enough to detect a level of activity far below that which would be considered harmful. Locations for sample collection are based on radiological and meteorological data from the Annual Effluent Release Report and information obtained from the Environmental Land Use Survey.

Most samples, particularly indicator samples, are collected in a circular area within a five-mile radius of plant containment. However, control locations are usually outside of five miles. Sample locations are listed by number along with their respective distances and direction from plant containment, in the ODCM.

When assessing sample results, data from indicator locations (those most likely to be affected by plant operations) are compared to those from control locations (those least or not likely to be affected). Results from an indicator location which were significantly higher than those from a control location could indicate a plant-attributable effect and could require additional investigation.

2.4.1.2 RGPP

Routine monitoring under the RGPP per NEI 07-07 is completed through the sampling and analysis of groundwater from 23 on-site monitoring wells, 2 surface water sites, and 4 storm water headers. The latest depiction of the monitoring wells is presented in Figure 2-6.

2.4.1.3 NPDES Permit and Nebraska Industrial Stormwater General Permit

The storm drain system conveys drainage from roads, parking areas, roof drains, and grassy surfaces throughout the station to 11 outfalls discharging to the adjacent Missouri River and drainage ditches. These points are identified as monitored outfalls and are authorized under Nebraska Industrial Stormwater General Permit No. NER910677 [20].

Stormwater outfalls SW01 through SW03 and SW10 drain parking areas and do not require specific monitoring. All of the other stormwater outfalls require quarterly visual inspection and benchmark sampling. Storm drains (catch basins) are located along several of the piped portions of the storm drain system in both the DA and other portions of the site.

FCS has an authorization to discharge under the NPDES Permit No. NE0000418 [21]. Two NPDES outfalls are located at the site and include the facility discharge line (Outfall 001) and Outfall 003. The permit authorizes the facility to discharge wastewater and is subject to the limitations, requirements, prohibitions, and conditions set forth in the permit.

2.4.2 Summary of Analytical Results in Groundwater

The sampling results from 2011 through 2018 reveal that there have been sporadic detections of strontium-90 (Sr-90) up to 0.9 pCi/L, in several shallow wells, but no other hard-to-detect radionuclides. The episodic but low Sr-90 concentrations reported in the shallow monitoring wells within the DA suggest that a small Sr-90 release has occurred at the site.

Tritium (H-3) has been detected at concentrations less than 416 pCi/L in only one monitoring well (MW-6) over the 2011 through 2018 groundwater monitoring time period. The minimal H-3 detections and the minor sporadic detections of Sr-90 indicate that groundwater at the FCS site is not significantly impacted by radiological contaminants, and that a plume of H-3 or Sr-90 is not migrating off-site.

The H-3 detected in MW-6 has occurred on three occasions, 2nd and 3rd quarter 2014 (223 and 416 pCi/L, respectively) and 2nd quarter 2018 (241 pCi/L). The H-3 concentrations detected are very near the minimum detection limit, H-3 was not detected in all other monitoring wells, and no on-site activities performed during these times would have triggered an increase in H-3 levels. This suggests that the values are potentially false positive.

Plant-related gamma-emitting radionuclides have not been identified in any groundwater samples.

All sample results for residential well water from off-site locations have been at the lower limit of detection with the exception of gross beta results due to naturally-occurring radionuclide concentrations.

2.4.3 Summary of Analytical Results in Surface Water

Under the REMP, three surface water locations are sampled on monthly basis for gamma isotopic analysis and on a quarterly basis for H-3 analysis. To date, H-3 or plant-related gamma-emitting radionuclides have not been detected in surface water samples.

2.5 <u>Continuing Characterization</u>

The characterization of inaccessible or not readily accessible subsurface soils, buried pipes, or concrete surfaces has been deferred. These areas will be characterized (termed “continuing characterization”) as access is gained. The number and location of measurements and samples for continuing characterization will be determined by DQOs during survey design. The results of continuing characterization surveys will be addressed with revisions to the Radiological Characterization Report. The following are areas at FCS where continuing characterization will occur:

- The concrete of the Spent Fuel Pool and Fuel Transfer Canal. Continuing characterization will consist of the acquisition of volumetric concrete samples.
- The concrete walls and floor of the Under Vessel area in Containment. Continuing characterization will consist of the acquisition of additional volumetric concrete samples as augment to the existing volumetric concrete data.
- The concrete walls and floors of the 971 feet and 989 feet AMSL elevations of the Auxiliary Building. Continuing characterization will consist of the acquisition of additional volumetric concrete samples to bound the area and depth of elevated areas identified during the initial characterization.
- The concrete floors and walls of the Waste Hold-Up Tank rooms (Room 8). Continuing characterization will consist of the acquisition of volumetric concrete samples.
- The concrete of the Containment Building exterior. Continuing characterization will consist of the acquisition of volumetric concrete samples.
- The concrete of the Turbine Building basement. Continuing characterization will consist of the acquisition of volumetric concrete samples.
- The soils underneath the concrete of the Containment Building basement, Turbine Building basement, and Spent Fuel Pool. Continuing characterization will either consist of soil borings at the nearest locations along the foundation walls that can be feasibly accessed, angled soil bores to access the soils under the concrete, or soil samples collected after access is gained via core boring through the slab. Additional investigations and sampling will be performed in accordance with a sample plan if activity is positively identified.
- The embedded floor drain systems on the 971 feet and 989 feet AMSL elevations of the Auxiliary Building. Continuing characterization will consist of direct measurements on accessible portions of pipe and the acquisition of sediment or debris samples (if available) for analysis.
- The embedded floor drain system on the 990 feet AMSL elevation of the Turbine Building. Continuing characterization will consist of direct measurements on accessible portions of pipe and the acquisition of sediment or debris samples (if available) for analysis.
- The buried pipe systems that will be abandoned in place. Continuing characterization will consist of direct measurements on accessible portions of pipe and the acquisition of sediment or debris samples (if available) for analysis.

All surface soil at FCS has been adequately characterized; additional characterization of surface soil is not anticipated during continuing characterization. Radiological Assessments (RAs) will be performed in currently inaccessible soil areas that are exposed after removal of asphalt or concrete roadways and parking lots, rail lines, buried piping, or building foundation pads (slab on grade). Survey results for RAs will be presented in the relevant survey unit release records.

The decision to defer characterization was based on one or more of the following conditions:

- ALARA (as low as is reasonably achievable) considerations (e.g., the area is either a high radiation or high contamination area and additional data would likely not change the survey area or area classification of the location or surrounding areas)
- safety considerations
- historical data shows that the area could be classified without further characterization
- access for characterization would require significant deconstruction of adjacent systems, structures, or other obstacles where the removal could result in an unsafe condition or interfere with continued operation of operating systems
- the ability to use engineering judgment in assigning the area classification based on physical relationship to surrounding areas and the likelihood of the area to have radiological conditions represented by the conditions in these adjacent areas

Continuing characterization data will be used along with existing data to update the types of radionuclides present and update the variability in the radionuclide mix for both gamma-emitting and HTD radionuclides. In addition, as the decommissioning progresses, data from operational events caused by equipment failures or personnel errors, which may affect the radiological status of a survey unit will be captured. These events will be evaluated and, when appropriate, stored in the characterization database. All characterization data will be used in validating initial classifications and in planning for FSS.

2.6 References

[1] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.179, Revision 2, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," 2019.

[2] Radiation Safety and Control Services, "TSD 20-001, Historical Site Assessment for Fort Calhoun Station," 2020.

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

[4] U.S. Nuclear Regulatory Commission, "NUREG-1757, Volume 2, Revision 2, Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria - Draft Report for Comment," 2020.

[5] Omaha Public Power District, "FCSD-RA-LT-200, Characterization Survey Plan".

[6] Omaha Public Power District, "FCSD-RA-LT-100, Quality Assurance Project Plan for the License Termination Plan Development, Site Characterization and Final Status Survey Projects at Fort Calhoun Station".

[7] U.S. Nuclear Regulatory Commission, "NUREG-1700, Revision 2, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," 2018.

[8] Omaha Public Power District, "CH-ODCM-0001, Offsite Dose Calculation Manual".

[9] TSSD Services, "Fort Calhoun Nuclear Station Historical Site Assessment Report," 2016.

[10] TSSD Services, "Fort Calhoun Nuclear Station Limited Non-Radiological Characterization Survey," 2016.

[11] Omaha Public Power District, Fort Calhoun Station Unit 1, "Defueled Safety Analysis Report".

[12] Nuclear Energy Institute, "NEI 07-07, Industry Ground Water Protection Initiative - Final Guidance Document," 2007.

[13] Omaha Public Power District, "FC-18-002, Potential Radionuclides of Concern During the Decommissioning of Fort Calhoun Station".

[14] Radiation Safety and Control Services, "TSD 21-043, Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan," 2021.

[15] International Organization for Standardization, "ISO 7503-1, Part 1, Evaluation of Surface Contamination, Beta-Emitters (maximum beta energy greater than 0.15 MeV) and Alpha-Emitters," 2016.

[16] U.S. Nuclear Regulatory Commission, "NUREG-1576, Multi-Agency Radiological Laboratory analytical Protocols Manual (MARLAP)," 2004.

[17] U.S. Nuclear Regulatory Commission, "Regulatory Guide 4.15, Revision 2, Quality Assurance of Radiological Monitoring Programs (Inception through Normal Operations to License Termination) – Effluent Streams and the Environment," 2007.

[18] Omaha Public Power District, "NO-FC-10, Quality Assurance Topical Report".

[19] Omaha Public Power District, "FC-20-012, Fort Calhoun Station Decommissioning Project Radiological Characterization Report".

[20] "Nebraska Industrial Stormwater General Permit" No. NER910677.

[21] “National Pollutant Discharge Elimination System (NPDES) permit” No. NE0000418.

Figure 2-1 Structure Survey Units

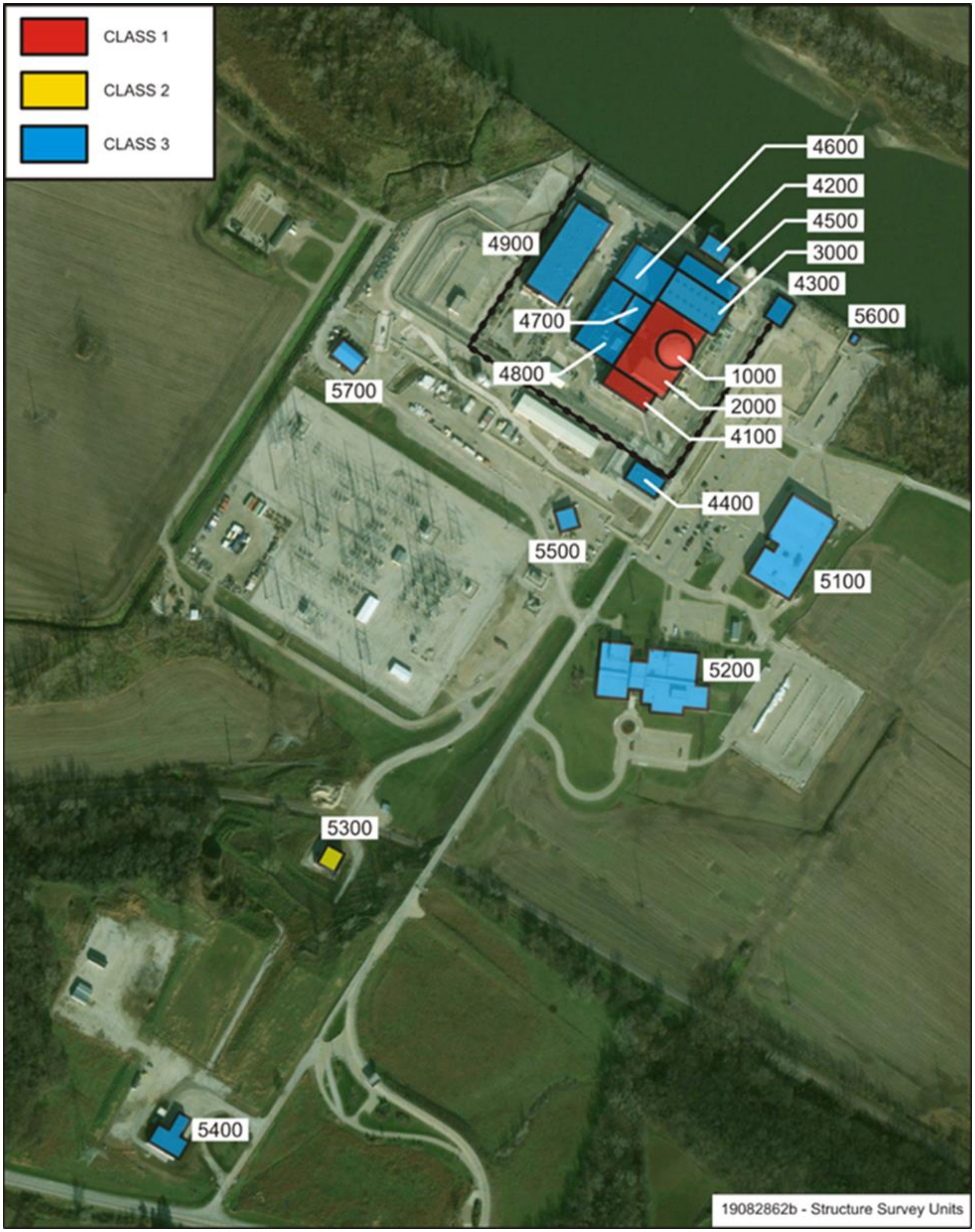
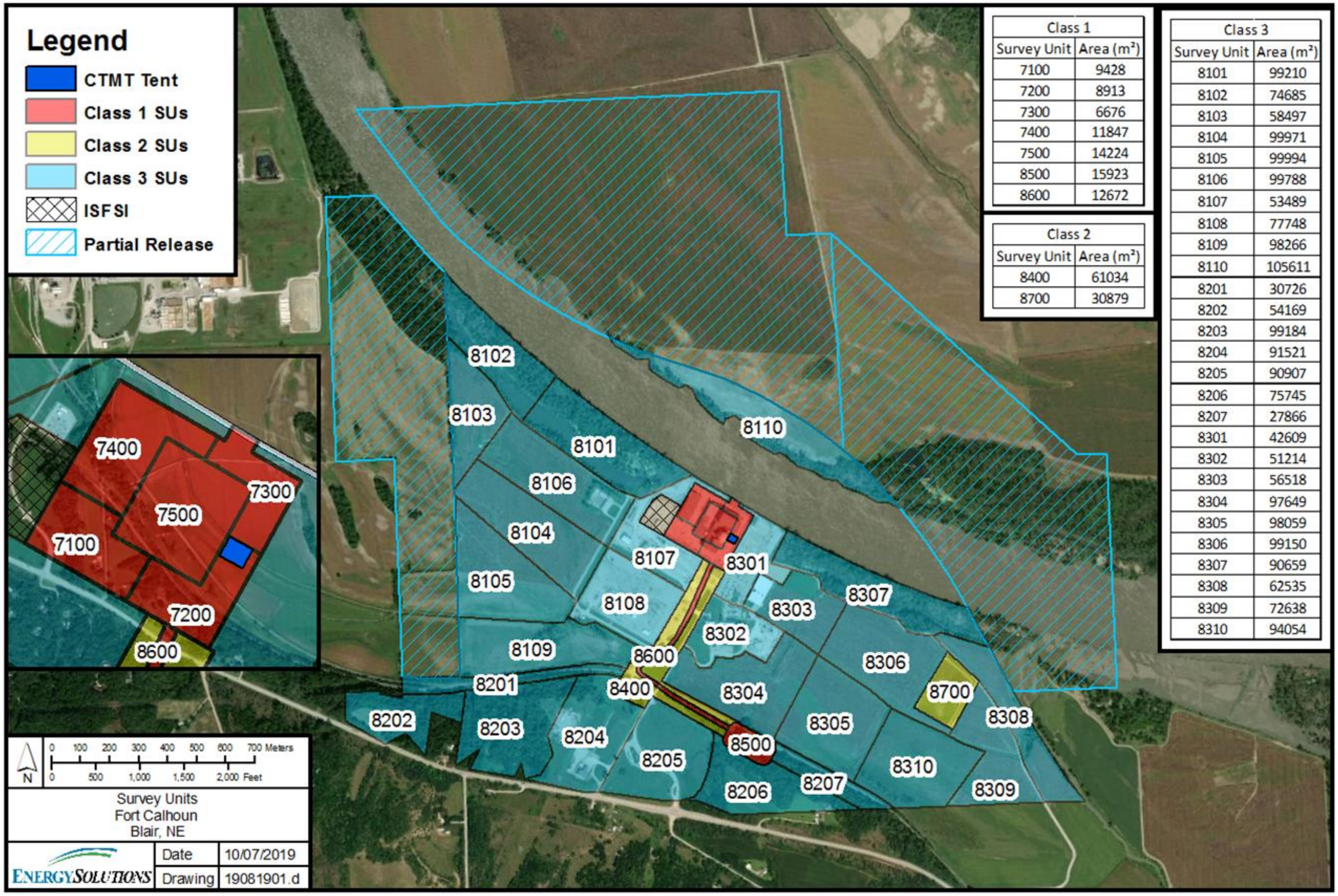


Figure 2-2 Class 3 Open Land Survey Units

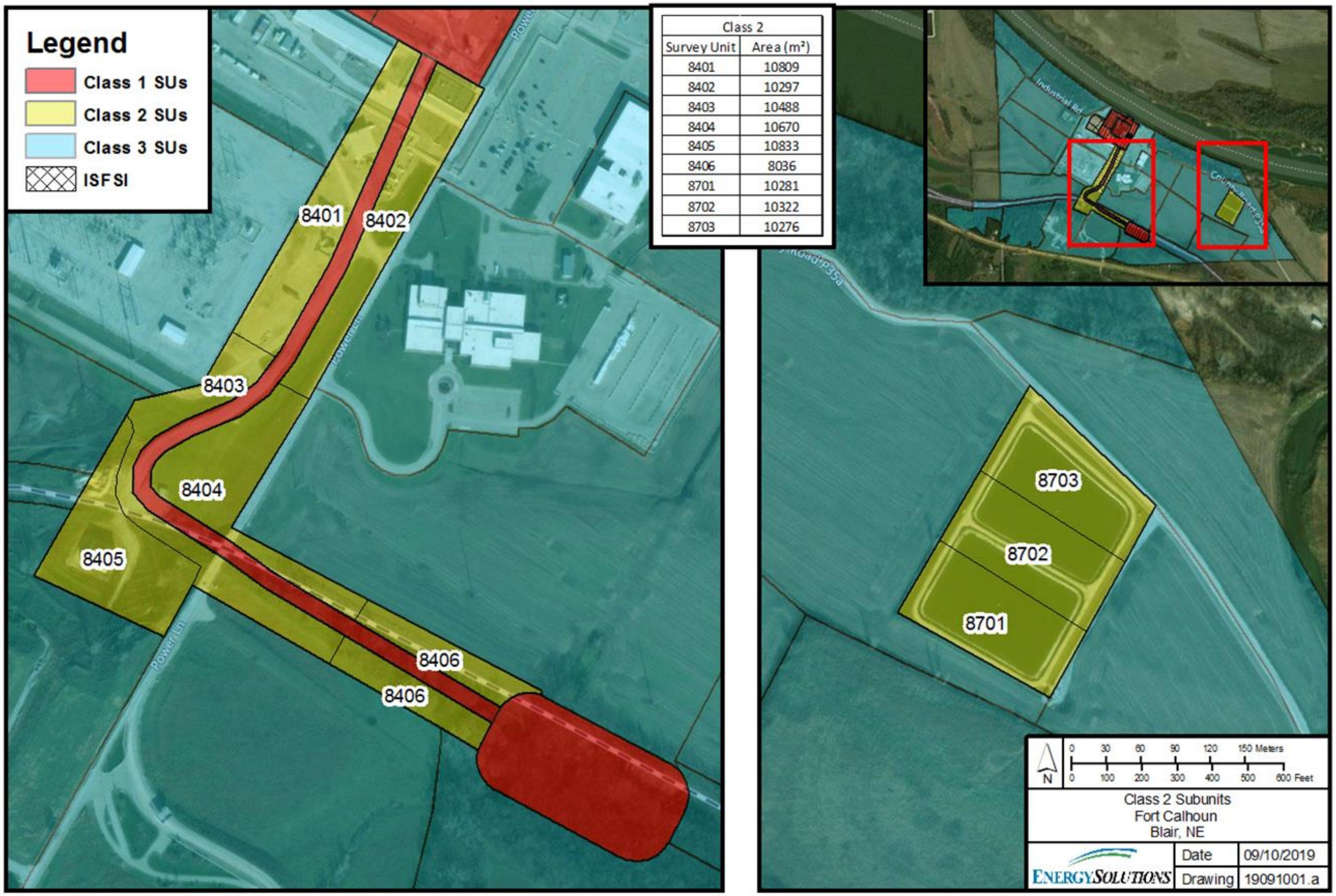


Class 1	
Survey Unit	Area (m²)
7100	9428
7200	8913
7300	6676
7400	11847
7500	14224
8500	15923
8600	12672

Class 2	
Survey Unit	Area (m²)
8400	61034
8700	30879

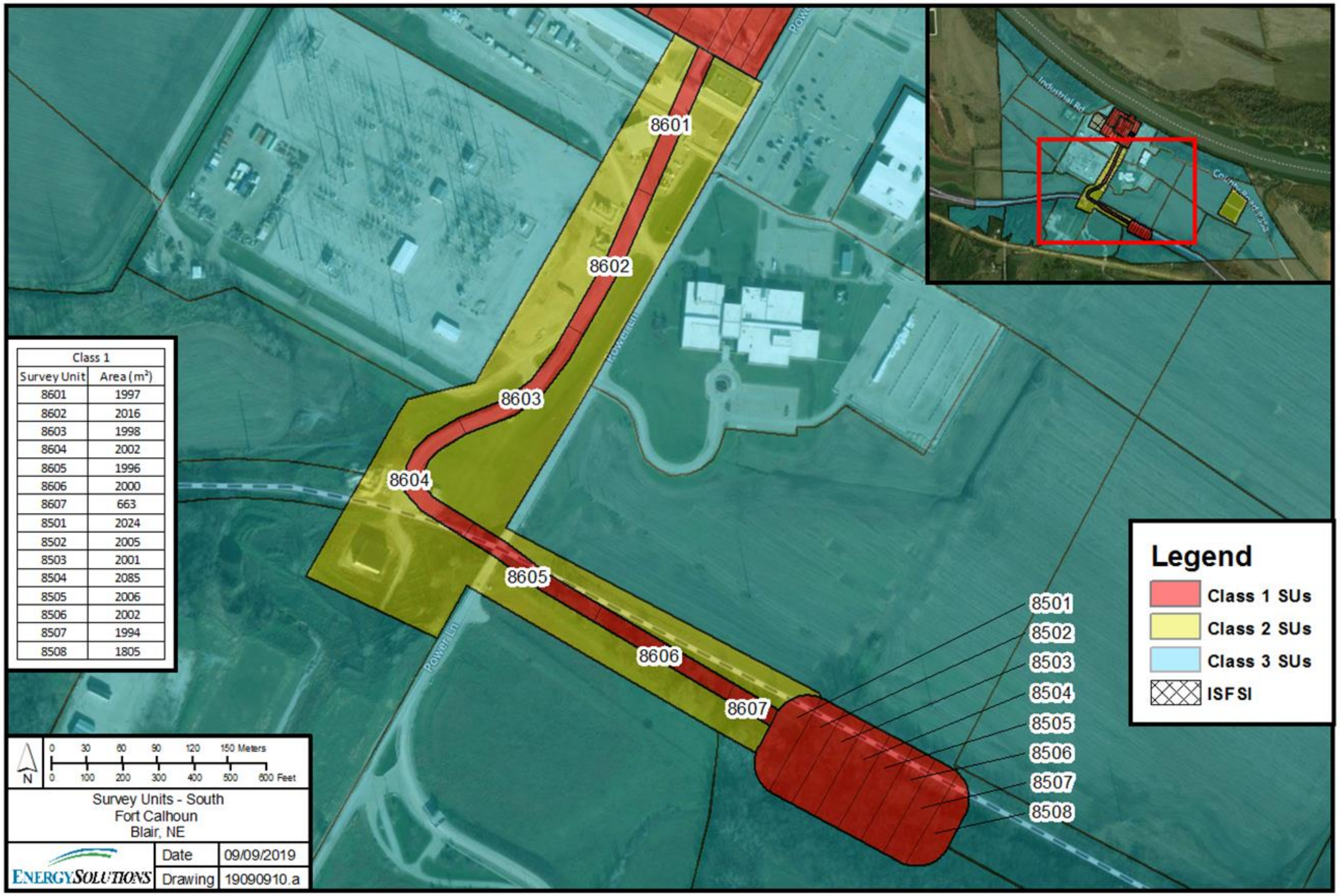
Class 3	
Survey Unit	Area (m²)
8101	99210
8102	74685
8103	58497
8104	99971
8105	99994
8106	99788
8107	53489
8108	77748
8109	98266
8110	105611
8201	30726
8202	54169
8203	99184
8204	91521
8205	90907
8206	75745
8207	27866
8301	42609
8302	51214
8303	56518
8304	97649
8305	98059
8306	99150
8307	90659
8308	62535
8309	72638
8310	94054

Figure 2-3 Class 2 Open Land Survey Units



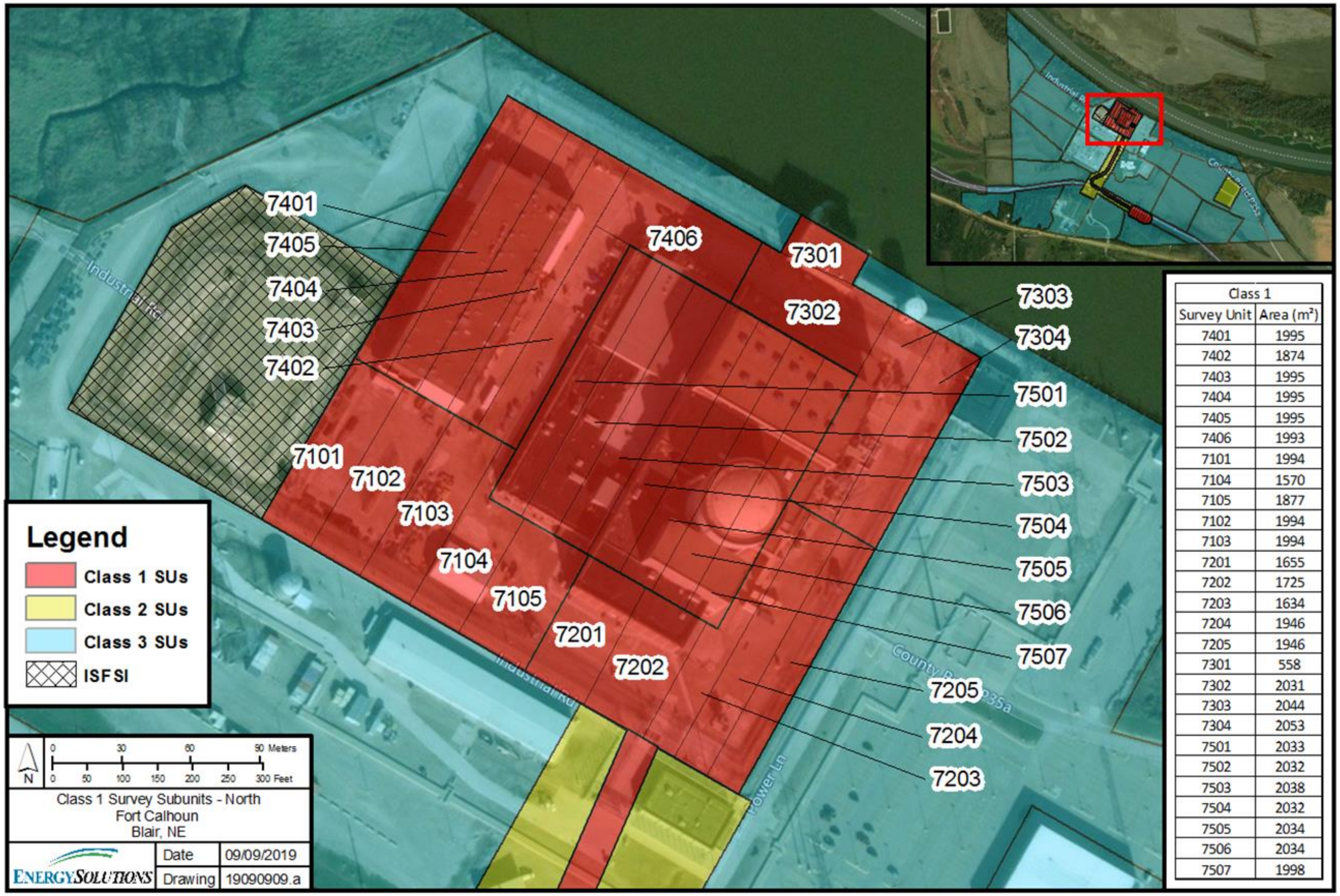
Class 2	
Survey Unit	Area (m^2)
8401	10809
8402	10297
8403	10488
8404	10670
8405	10833
8406	8036
8701	10281
8702	10322
8703	10276

Figure 2-4 Waste Haul Path and Waste Loadout Containment Structure Open Land Survey Units



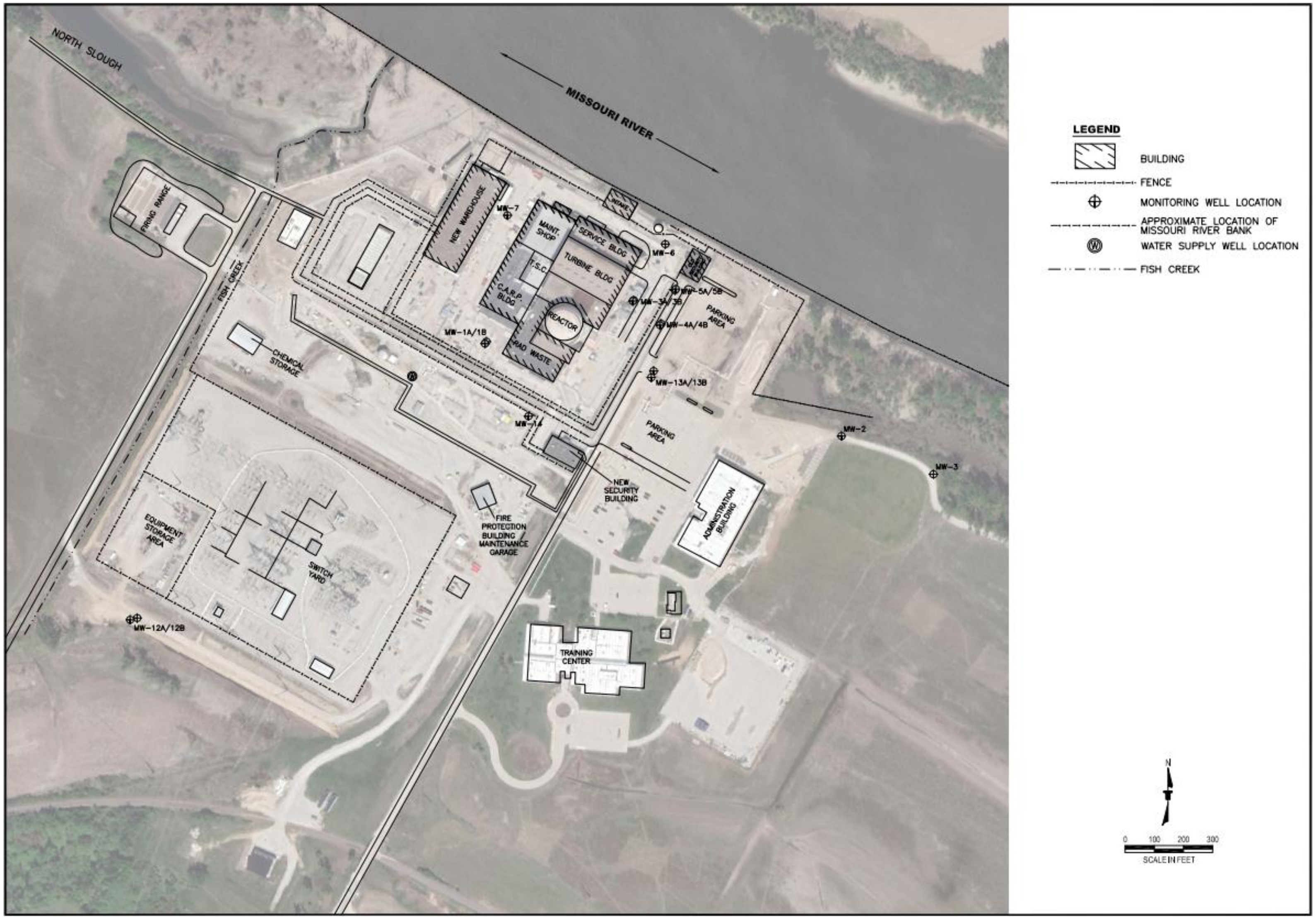
Class 1	
Survey Unit	Area (m^2)
8601	1997
8602	2016
8603	1998
8604	2002
8605	1996
8606	2000
8607	663
8501	2024
8502	2005
8503	2001
8504	2085
8505	2006
8506	2002
8507	1994
8508	1805

Figure 2-5 Class 1 Open Land Survey Units



Class 1	
Survey Unit	Area (m²)
7401	1995
7402	1874
7403	1995
7404	1995
7405	1995
7406	1993
7101	1994
7104	1570
7105	1877
7102	1994
7103	1994
7201	1655
7202	1725
7203	1634
7204	1946
7205	1946
7301	558
7302	2031
7303	2044
7304	2053
7501	2033
7502	2032
7503	2038
7504	2032
7505	2034
7506	2034
7507	1998

Figure 2-6 Monitoring Well Locations



Attachment 2-1 Results for Volumetric Concrete Samples

Table 2-12 Summary of Volumetric Concrete Sample Locations (Containment and Auxiliary Buildings)

Building	SU	Class	Description	Judgmental Measurement Locations		Total Judgmental Measurement Locations	# of Samples Collected	Number of Samples Analyzed
				3 increments (1.5")[a]	6 increments (6")[b]			
CB	1100	1	CB 977' Elevation – Under Vessel Area	0	5	71	429	429
	1200	1	CB 995'/996' Elevation G/A	0	29			
	1201	1	CB 995' Elevation – 'A' S/G Enclosure	0				
	1202	1	CB 995' Elevation – 'B' S/G Enclosure	0				
	1300	1	CB 1013' Elevation G/A	0	10			
	1400	1	CB 1045' Elevation G/A	0	27			
AB	2100	1	AB 971' Elevation G/A	7	7	67	261	261
	2200	1	AB 989' Elevation G/A	20	13[c]			
	2300	1	AB 1007' Elevation G/A	16	0			
	2400	1	AB 1011' Elevation G/A		0			
	2500	1	AB 1013' Elevation G/A		0			
	2600	1	AB 1025' Elevation G/A	4	0			
	2700	1	AB 1036' Elevation G/A		0			
	2800	1	AB 1039' Elevation G/A		0			

(a) 1.5" breakdown: three 0.5" increments
(b) 6" breakdown: four 0.5" increments, two 2" increments
(c) original number of locations was 20; 7 locations in the hold up tank cubicles were inaccessible during survey

Table 2-13 Summary of Volumetric Concrete Sample Locations (Turbine and BOP in DA Buildings)

Building	SU	Class	Description	Judgmental Measurement Locations		Total Judgmental Measurement Locations	# of Samples Collected	Number of Samples Analyzed
				3 increments (1.5")[a]	6 increments (6")[b]			
TB	3100	3	TB 990' Elevation G/A	18	0	18	54	18
BOP in DA	4100	1	Radwaste Processing Building	6	0	22	66	36
	4200	3	Intake Building	6	0			
	4300	3	Security Building	1	0			
	4400	3	Security Access Facility (SAF)	1	0			
	4500	3	Service Building	1	0			
	4600	3	Maintenance Shop	2	0			
	4700	3	Technical Support Center (TSC)	1	0			
	4800	3	Chemistry and Radiation Protection Facility	3	0			
	4900	3	New Warehouse	1	0			

(a) 1.5" breakdown: three 0.5" increments
(b) 6" breakdown: four 0.5" increments, two 2" increments

Figure 2-7 Volumetric Concrete Sample Locations in Containment Building 977 Foot Elevation

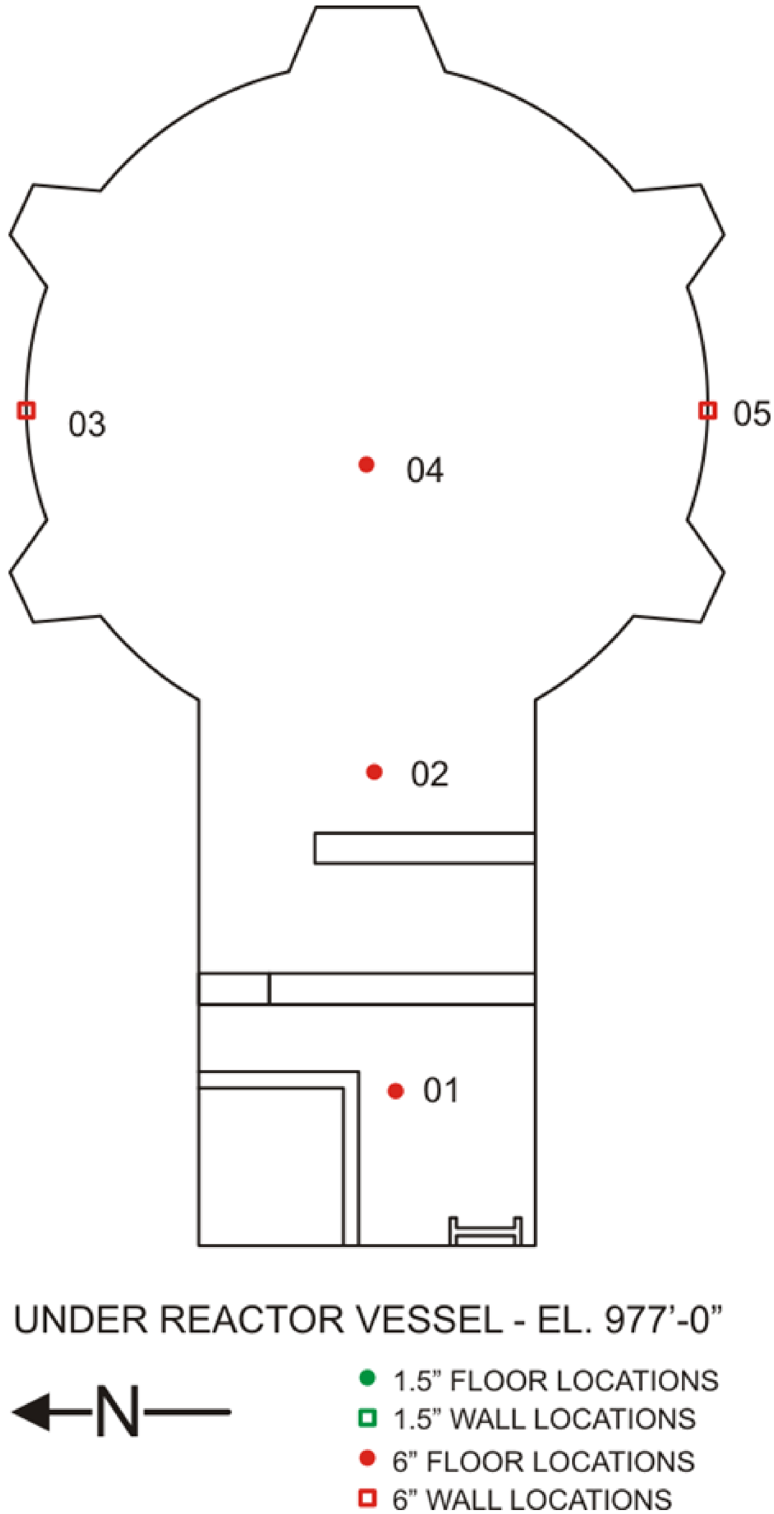


Figure 2-8 Volumetric Concrete Sample Locations in Containment Building 994 Foot Elevation

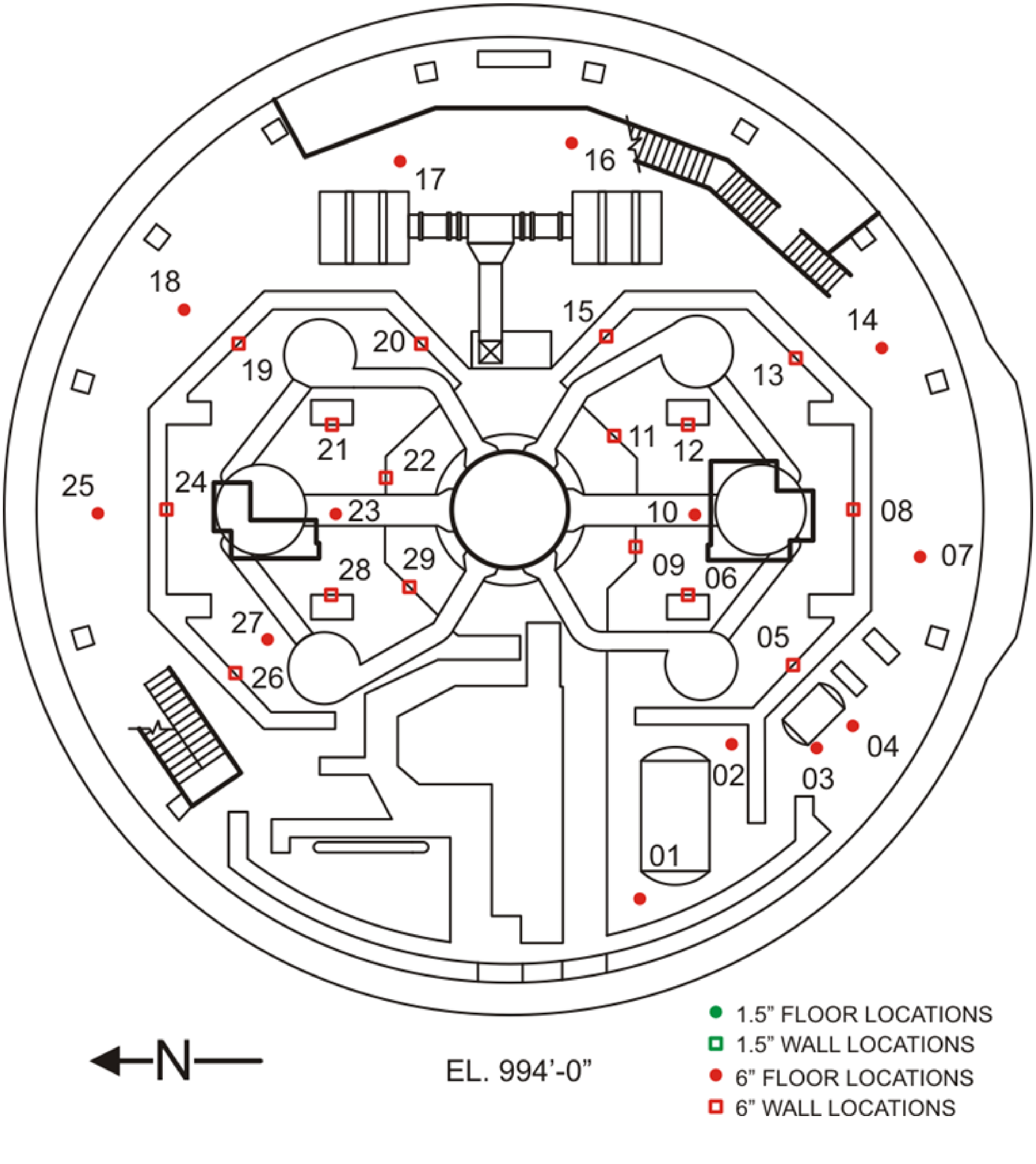


Figure 2-9 Volumetric Concrete Sample Locations in Containment Building 1013 Foot Elevation

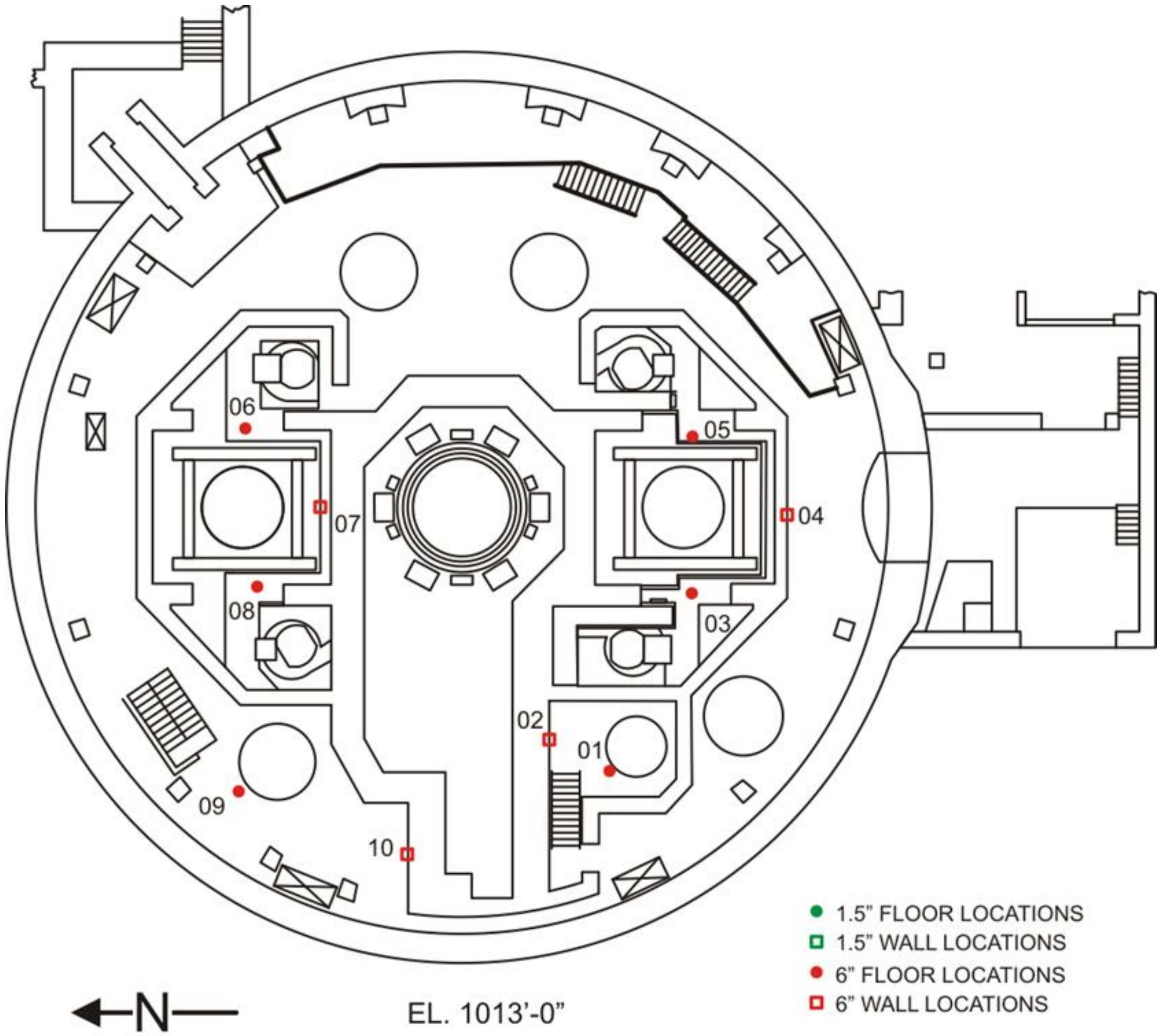


Figure 2-10 Volumetric Concrete Sample Locations in Containment Building 1045 Foot Elevation

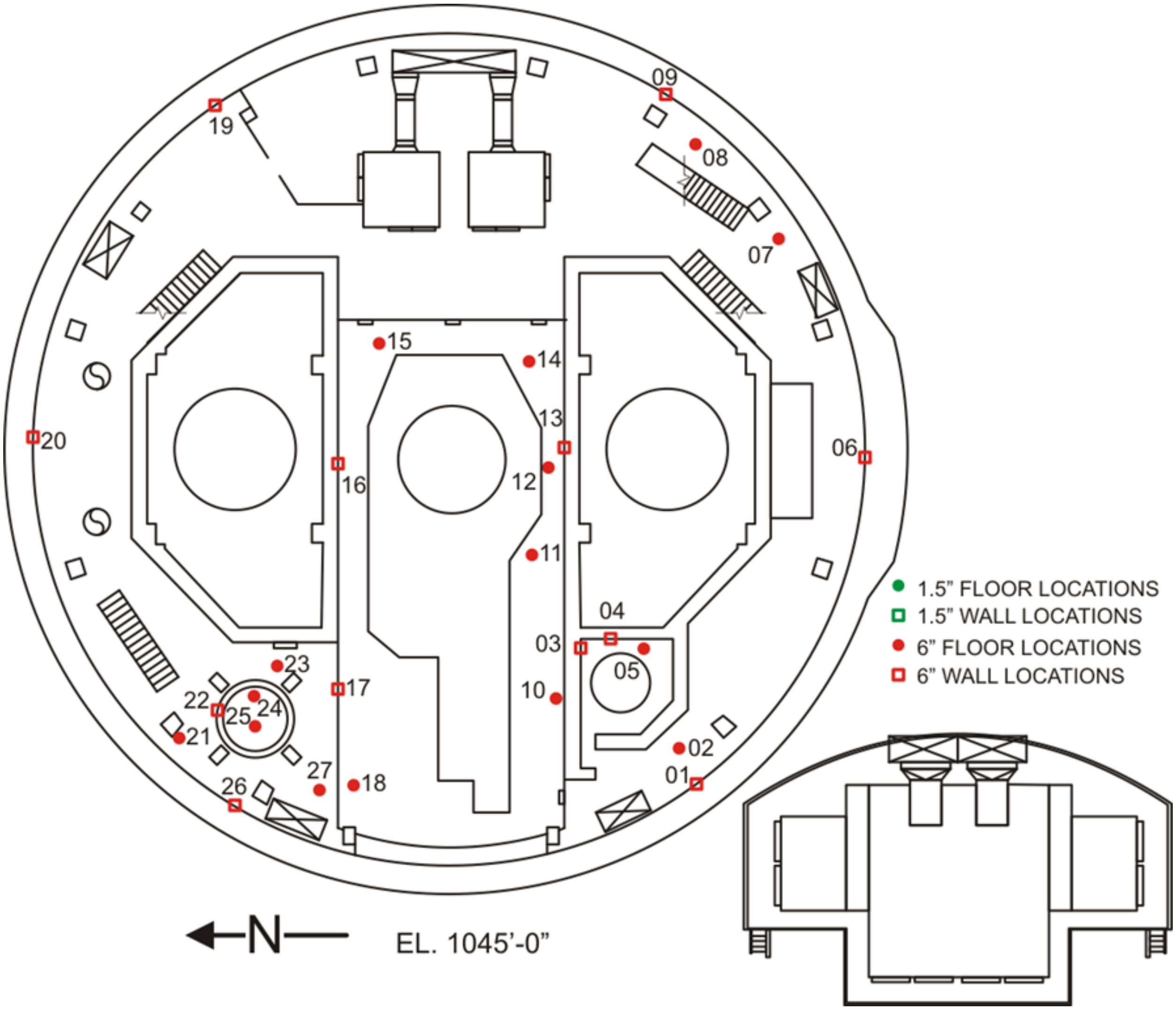


Figure 2-11 Volumetric Concrete Sample Locations in Auxiliary Building 971 Foot Elevation

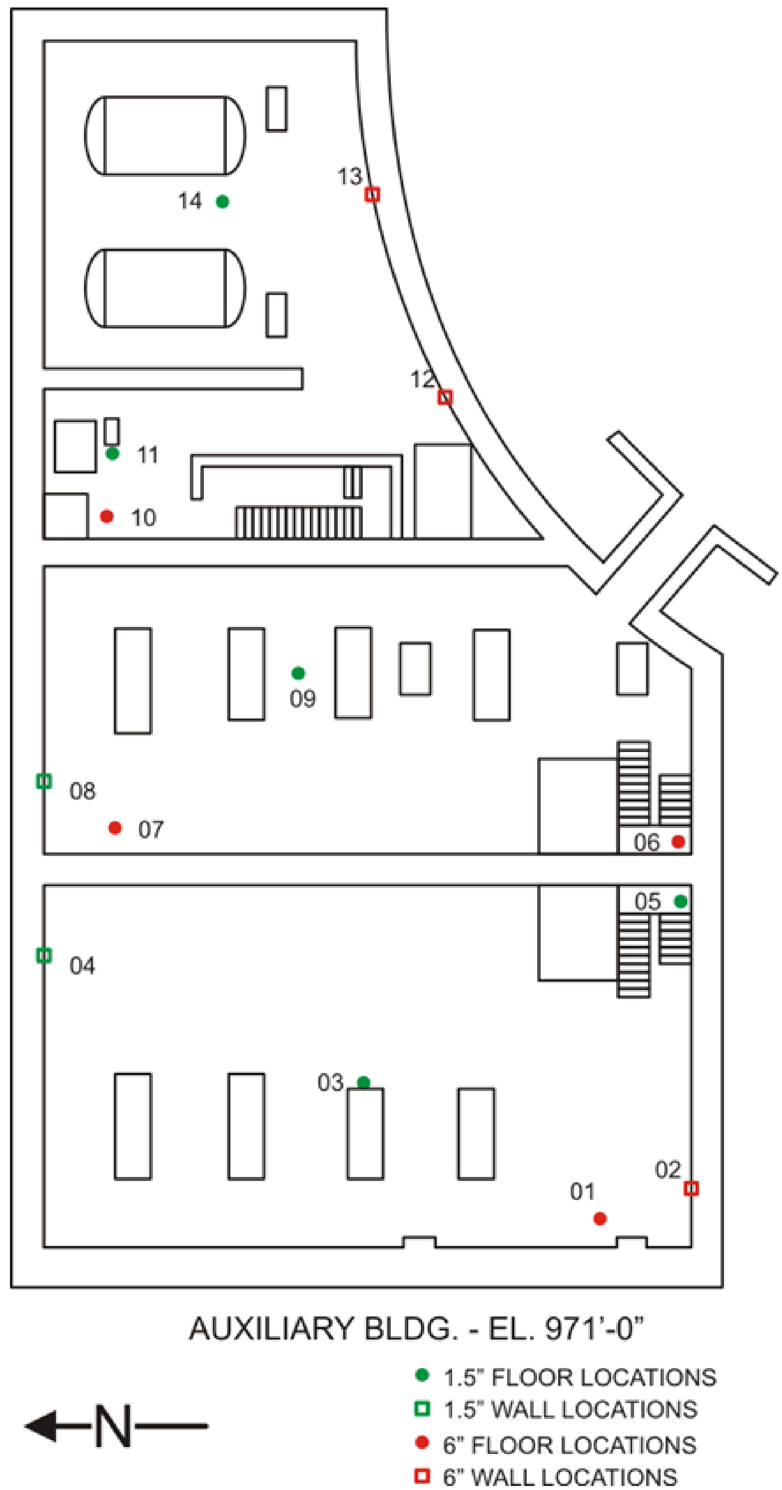


Figure 2-12 Volumetric Concrete Sample Locations in Auxiliary Building 989 Foot Elevation

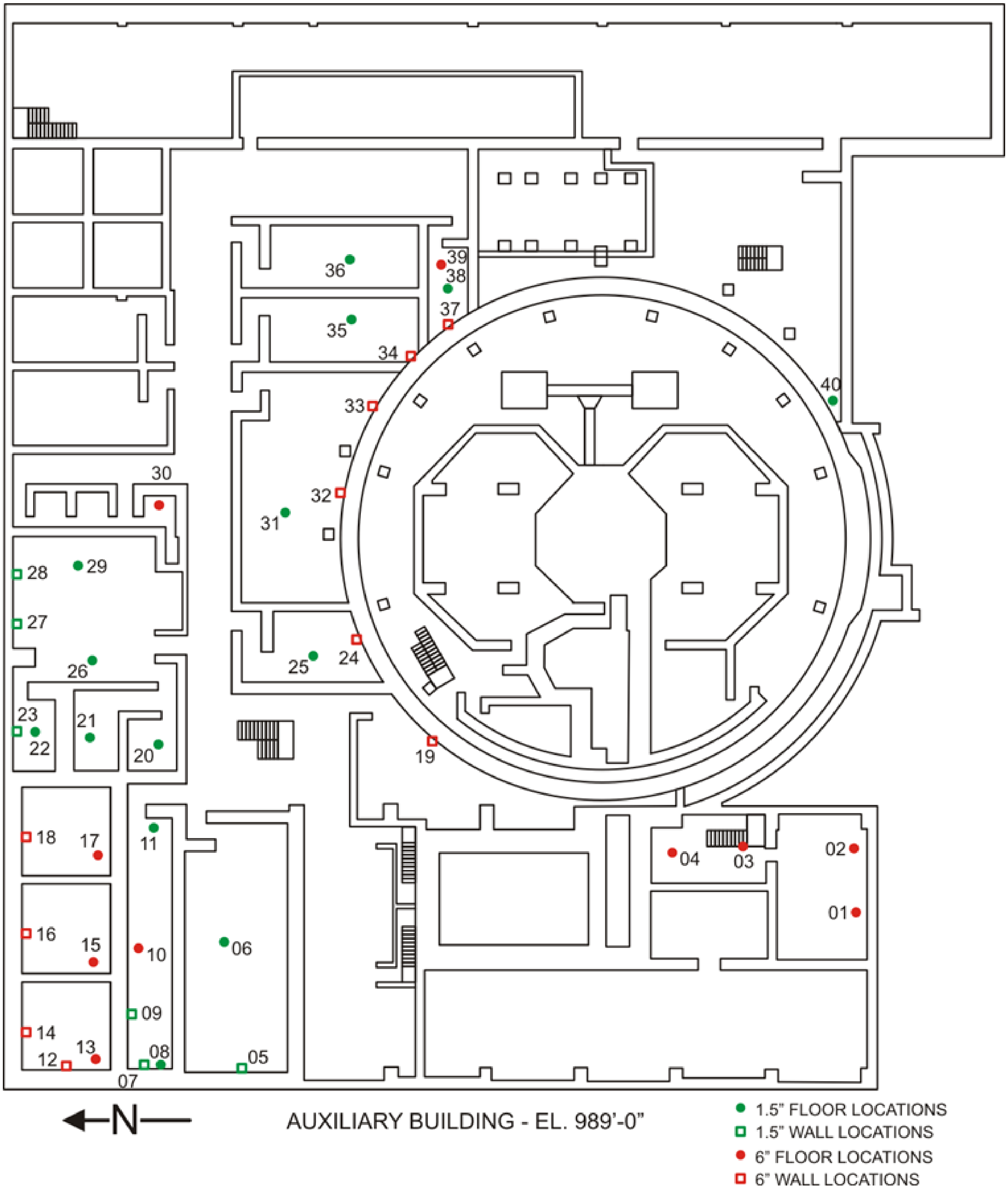


Figure 2-13 Volumetric Concrete Sample Locations in Auxiliary Building 1007 Foot Elevation

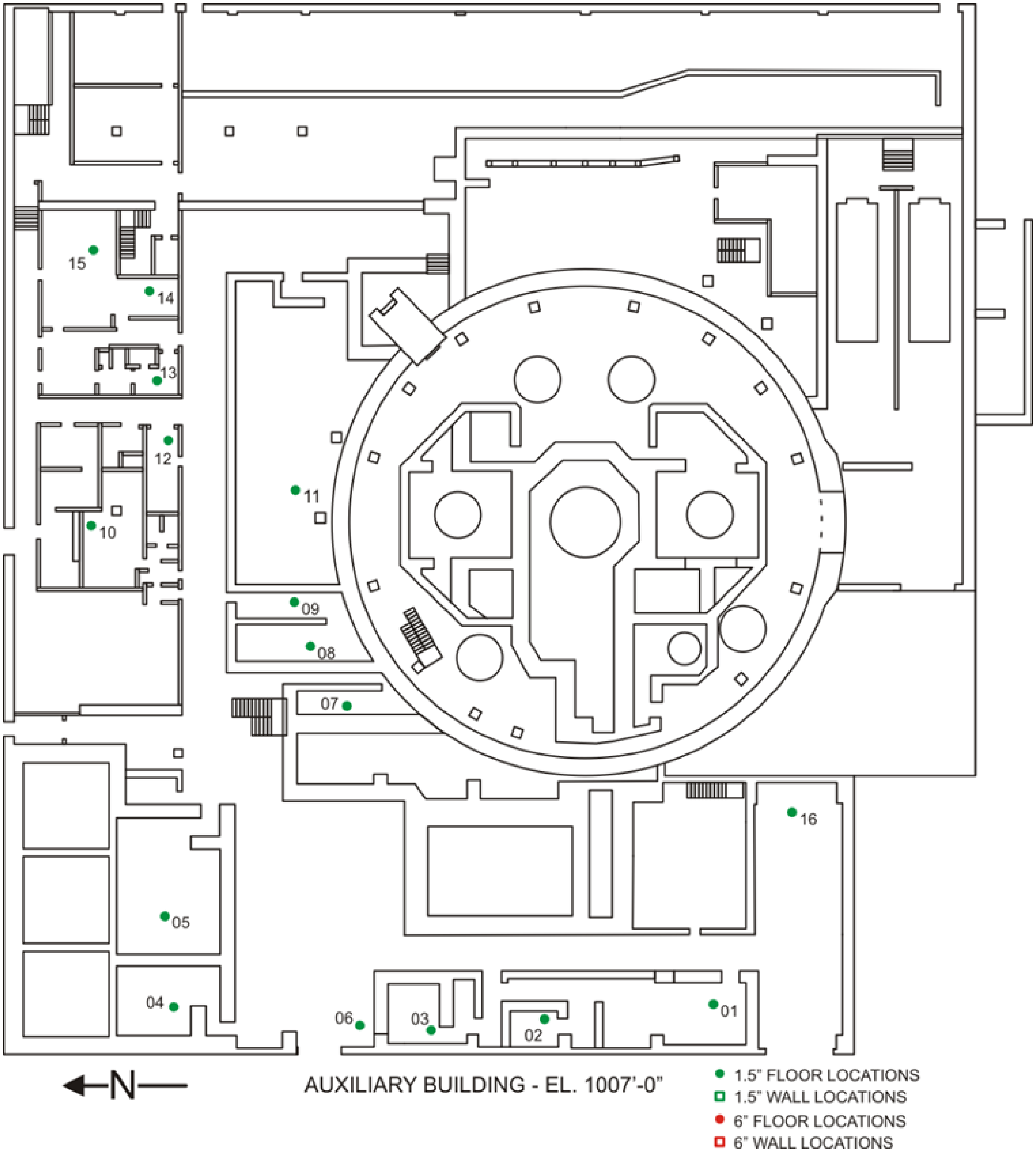


Figure 2-14 Volumetric Concrete Sample Locations in Auxiliary Building 1025 Foot Elevation

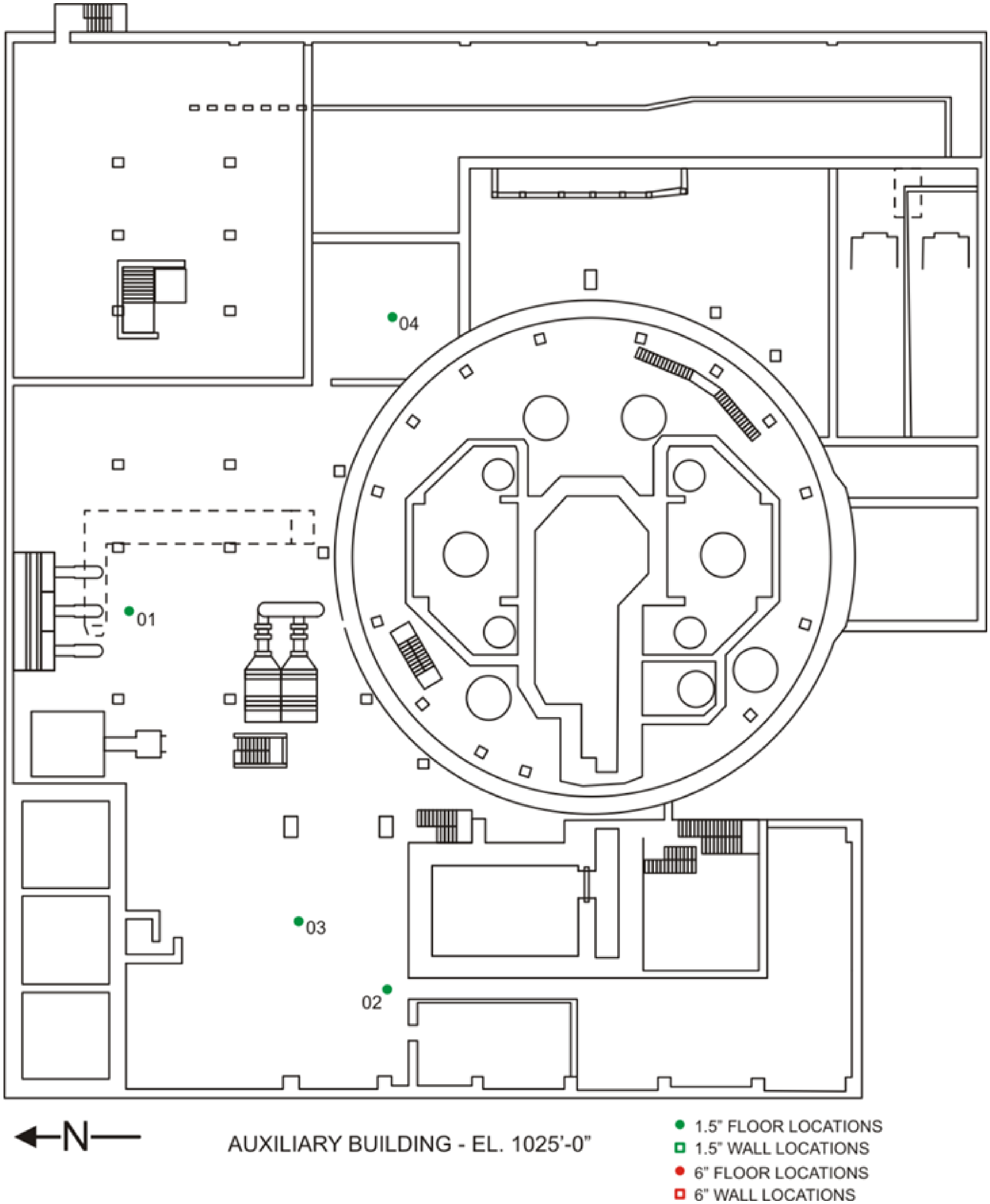


Figure 2-15 Volumetric Concrete Sample Locations in Turbine Building 990 Foot Elevation

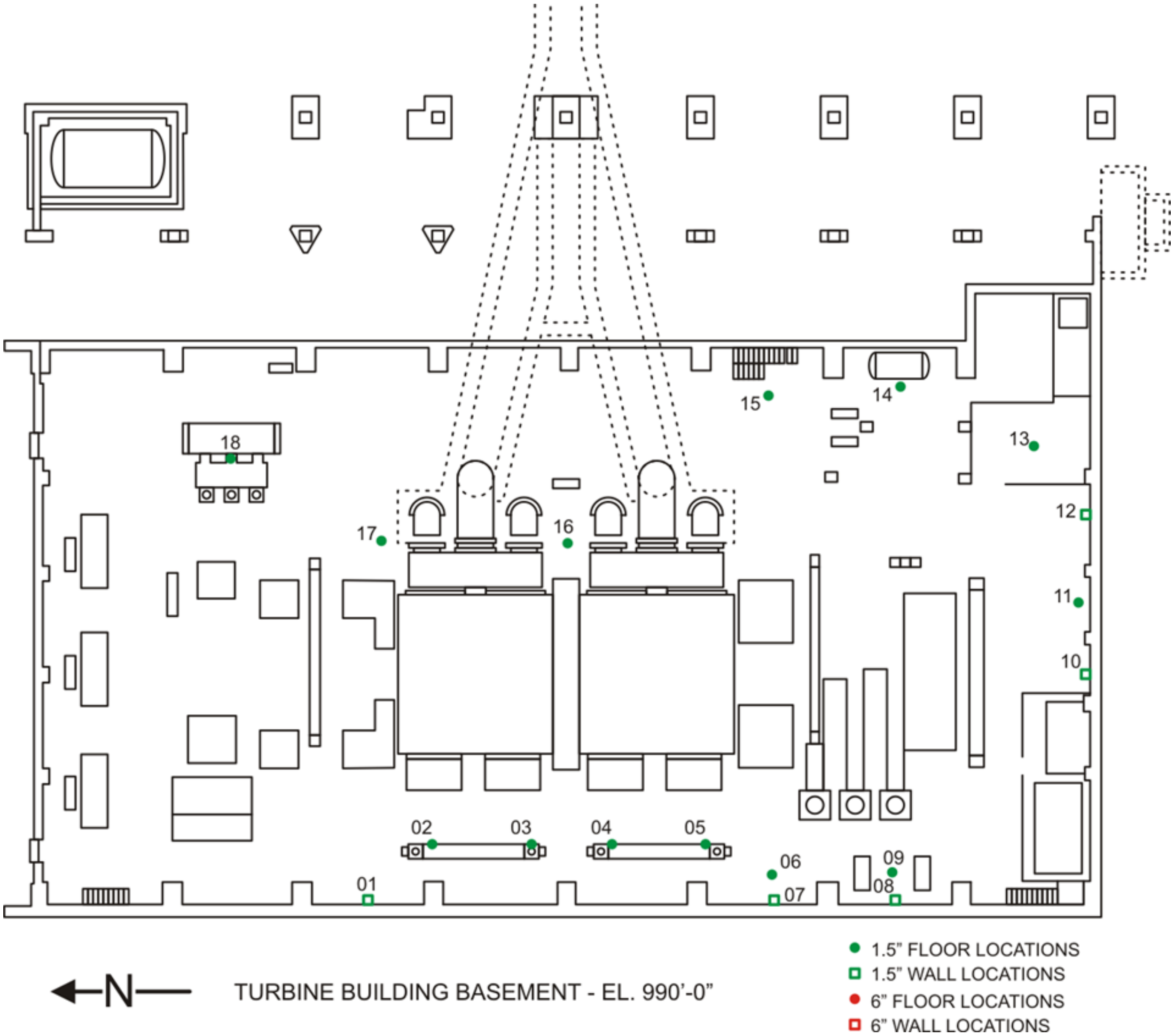


Figure 2-16 Volumetric Concrete Sample Location in New Warehouse

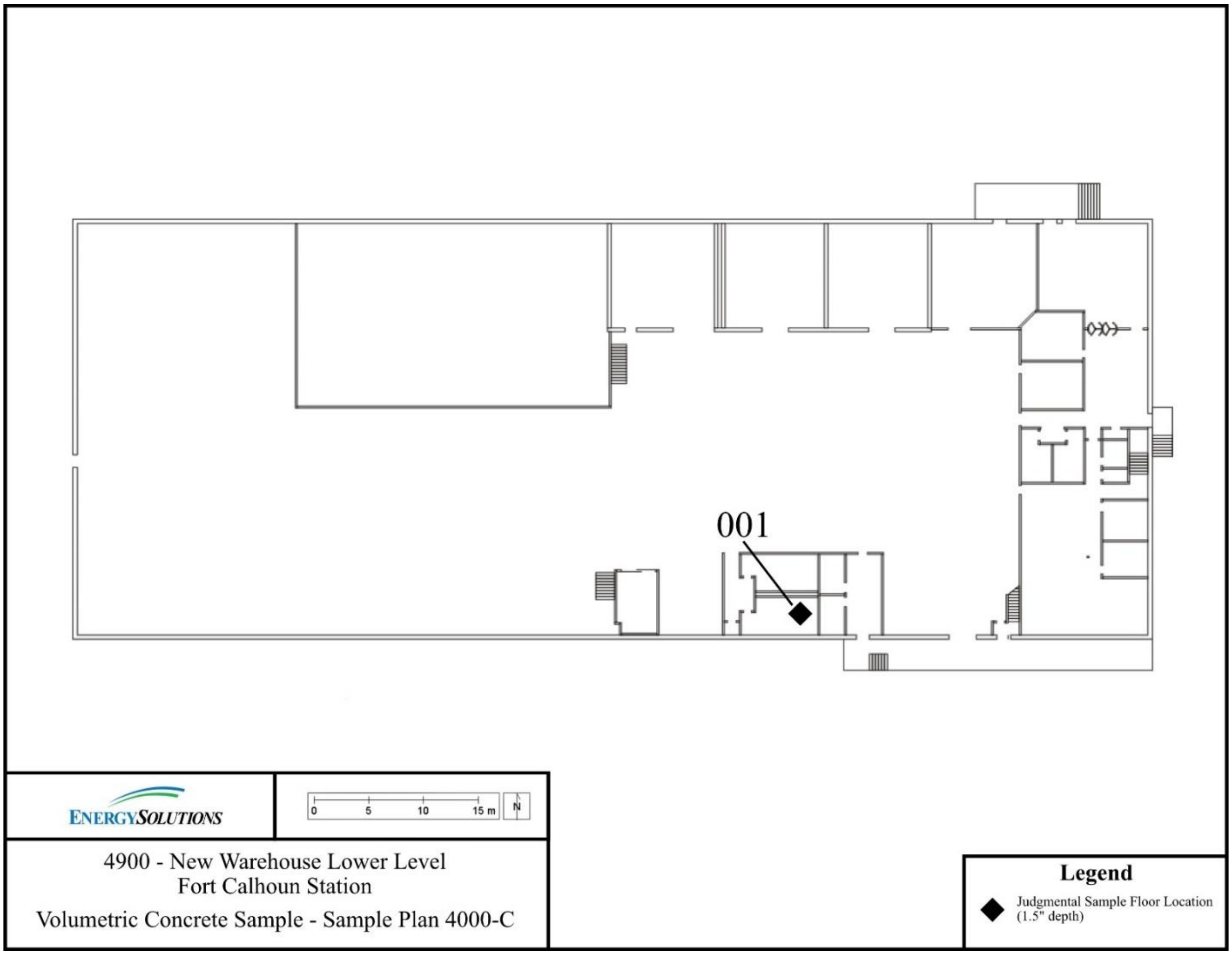


Figure 2-17 Volumetric Concrete Sample Location in Technical Support Center

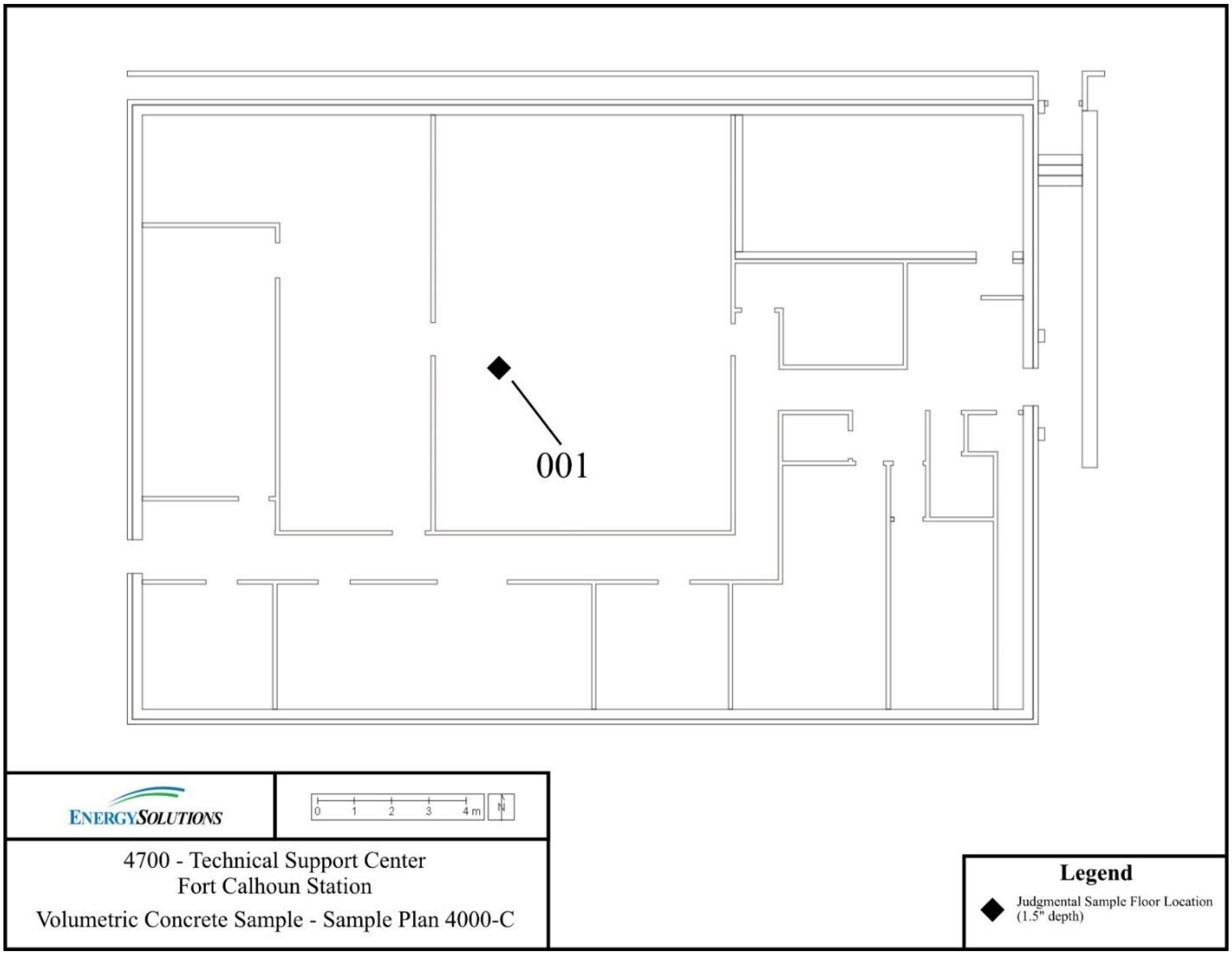


Figure 2-18 Volumetric Concrete Sample Location in Service Building

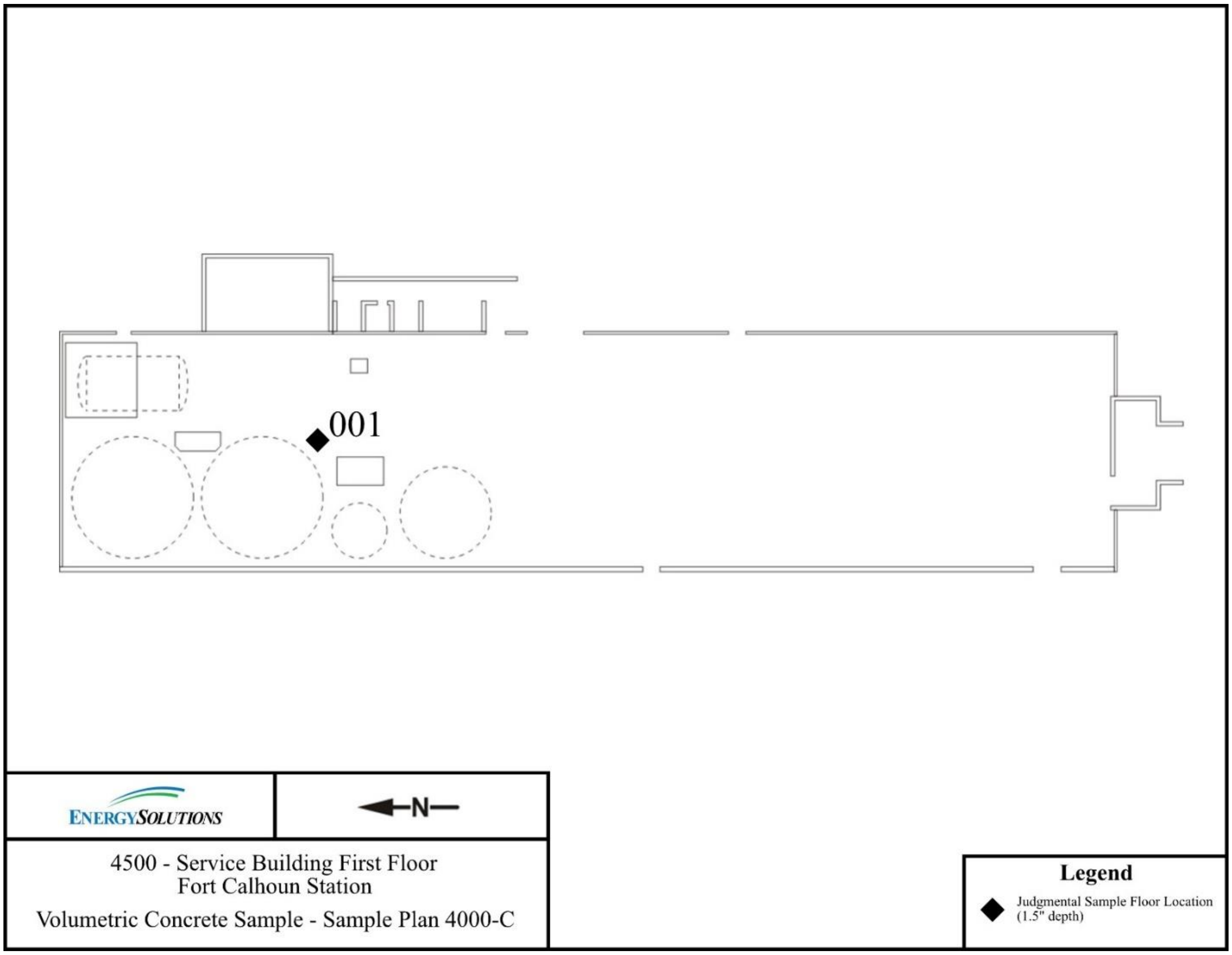


Figure 2-19 Volumetric Concrete Sample Location in Security Access Facility

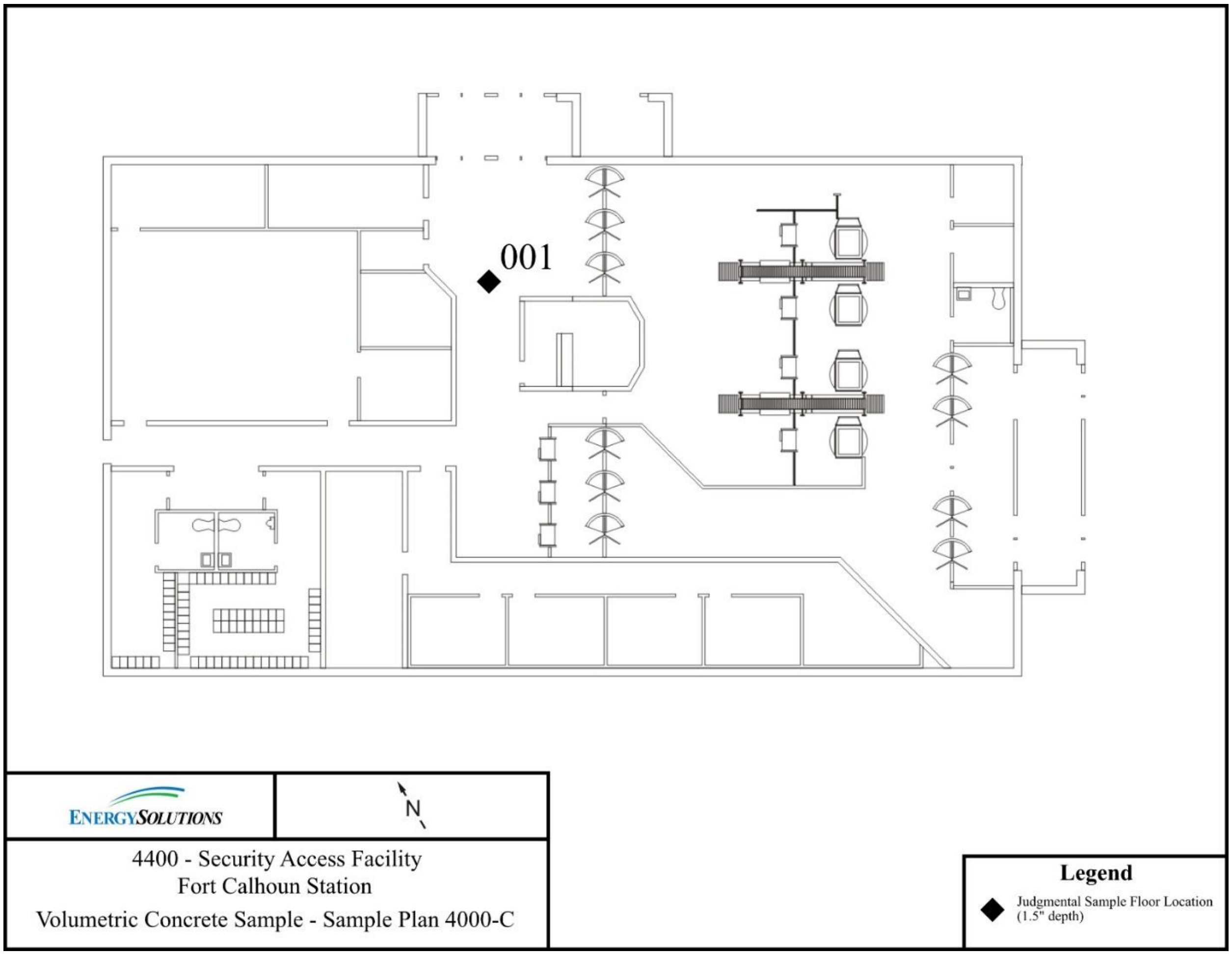


Figure 2-20 Volumetric Concrete Sample Location in Security Building

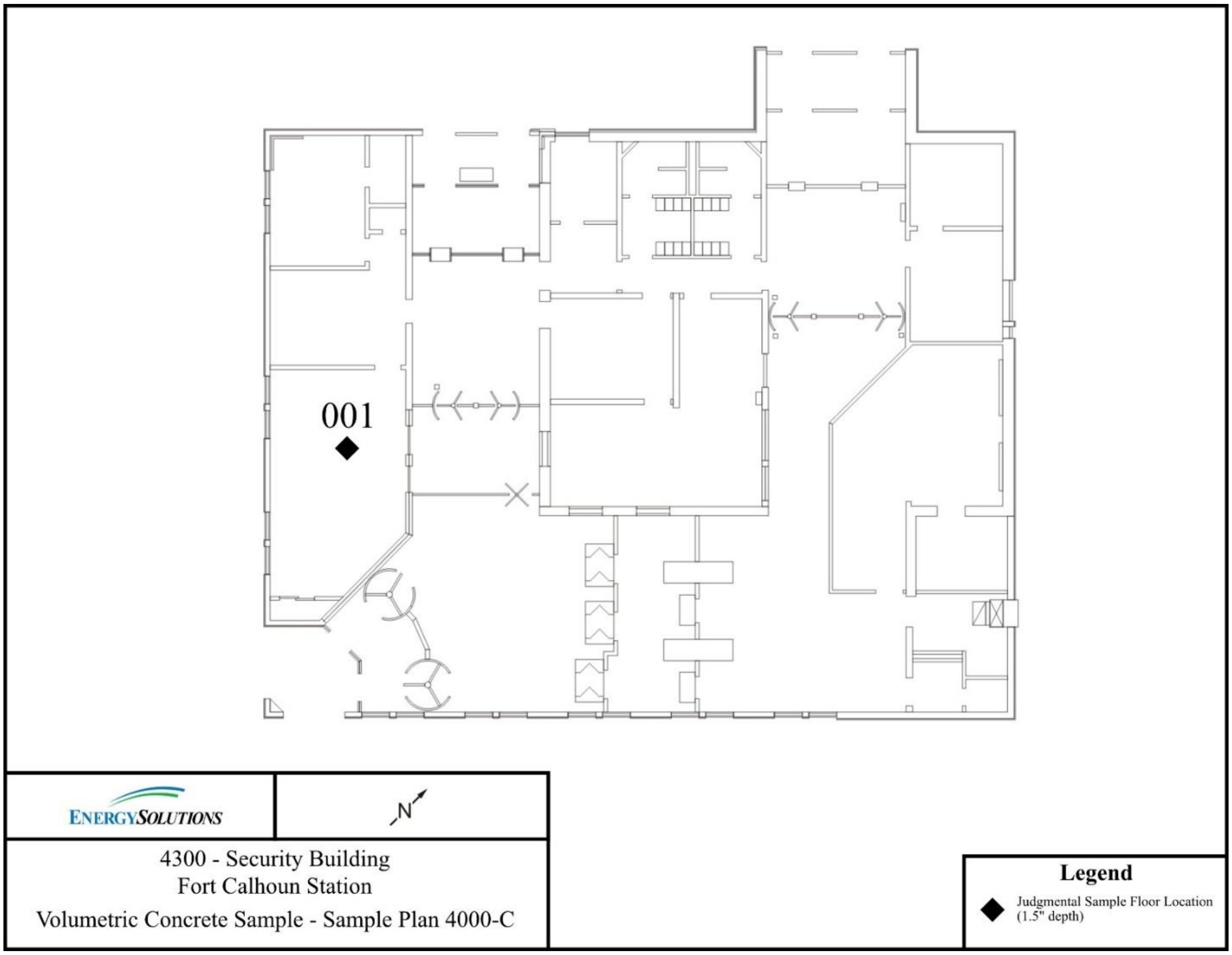


Figure 2-21 Volumetric Concrete Sample Location in Chemistry and Radiation Protection Building

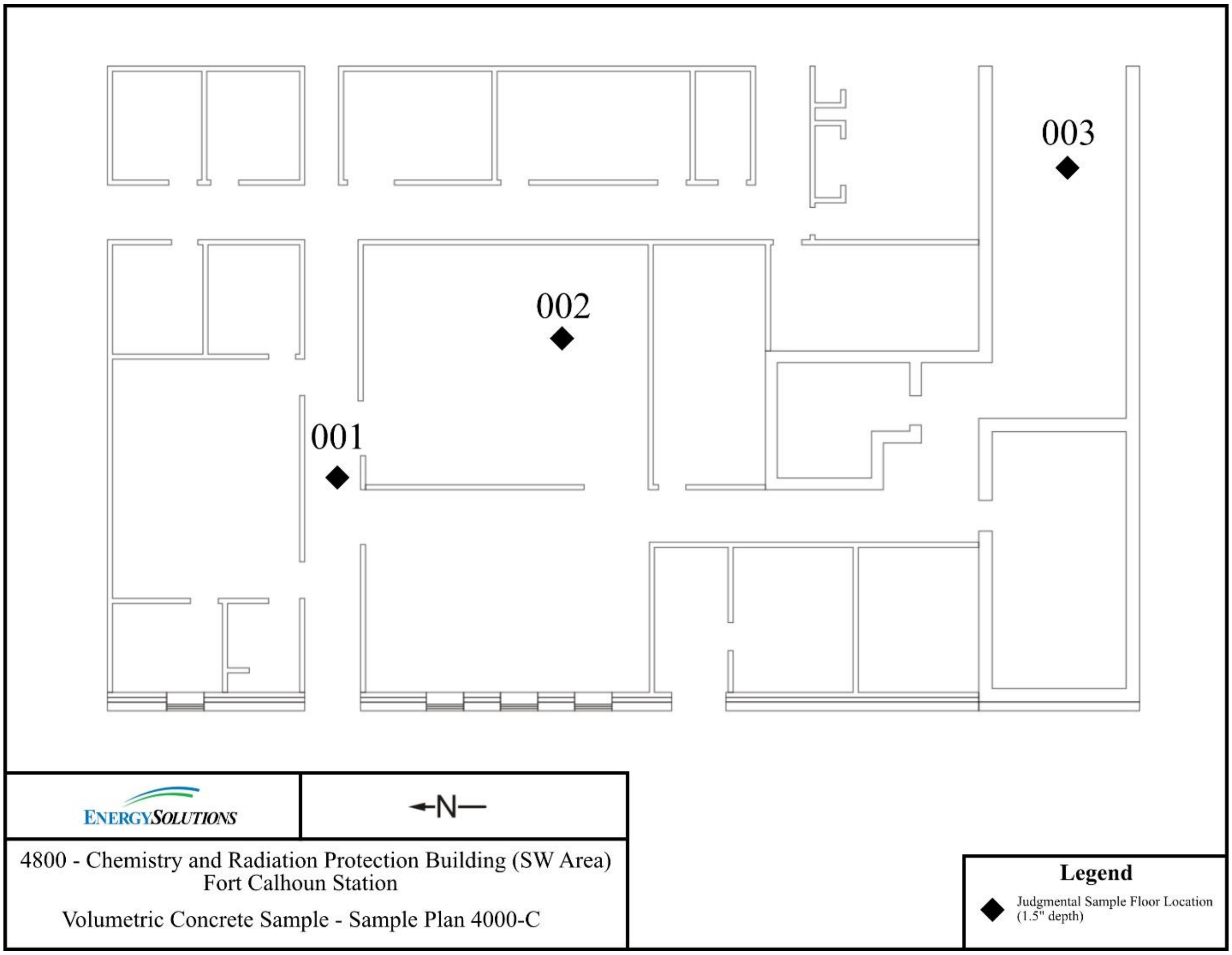


Figure 2-22 Volumetric Concrete Sample Location in Radwaste Building

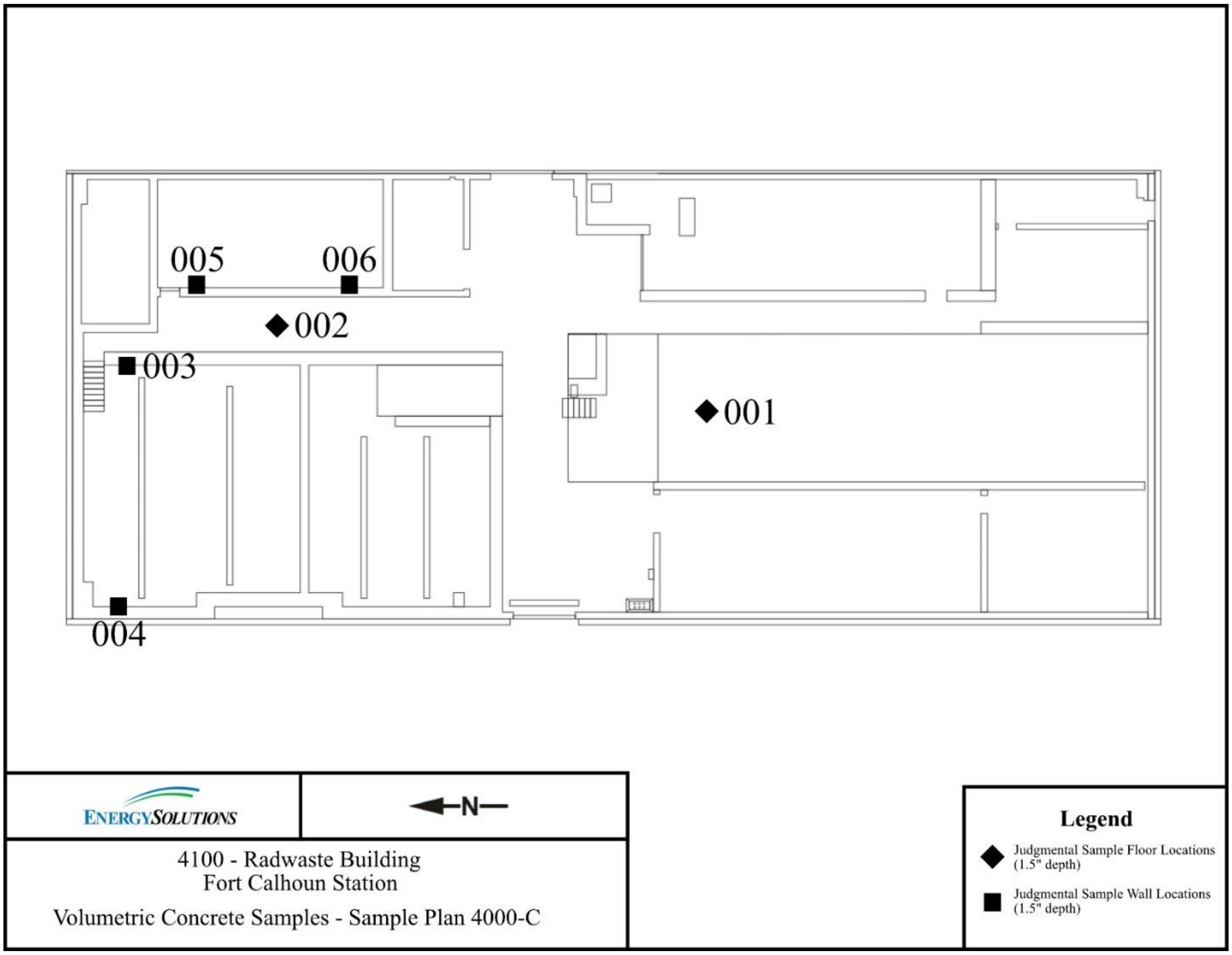


Figure 2-23 Volumetric Concrete Sample Location in Maintenance Shop

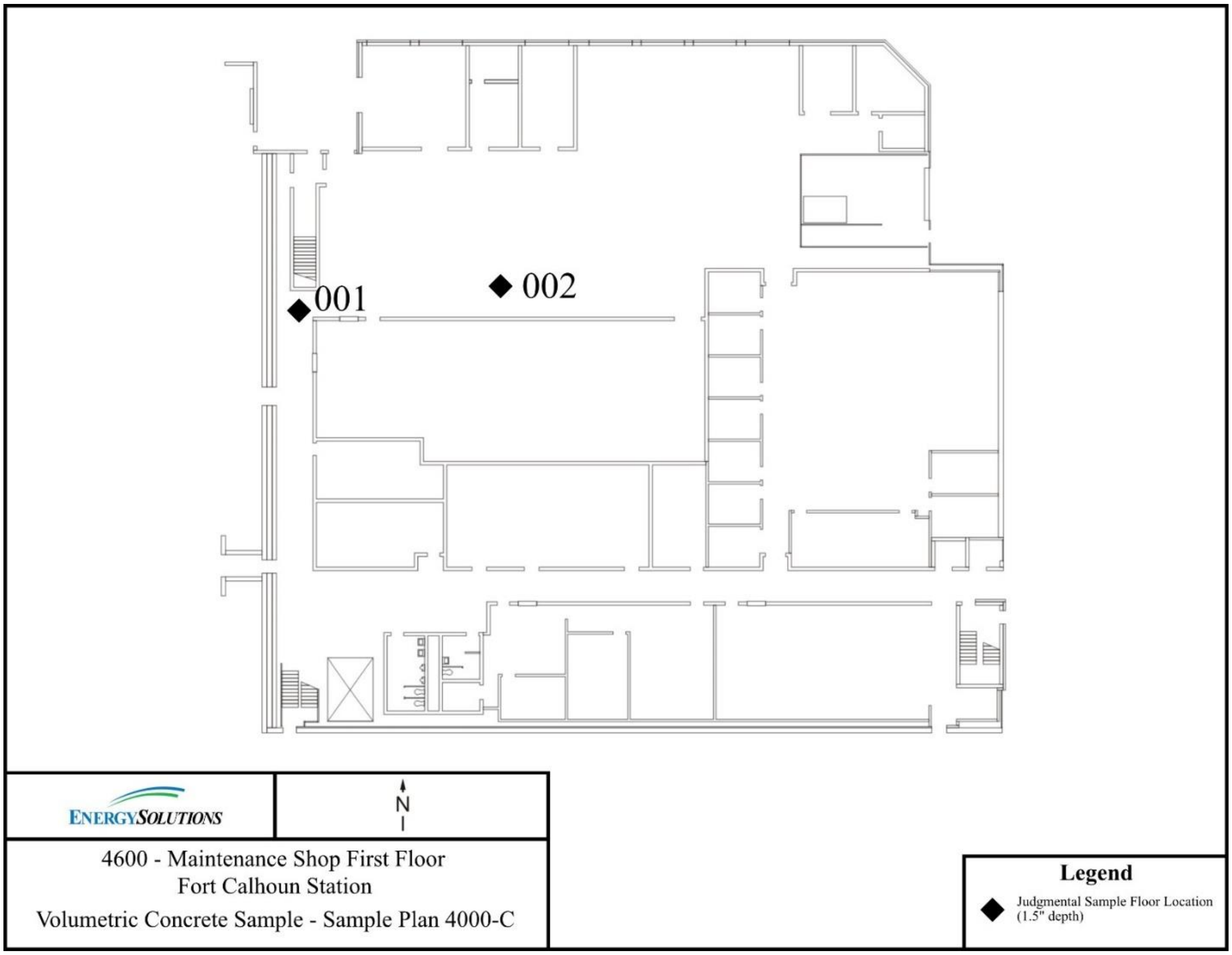


Figure 2-24 Volumetric Concrete Sample Location in Intake Building

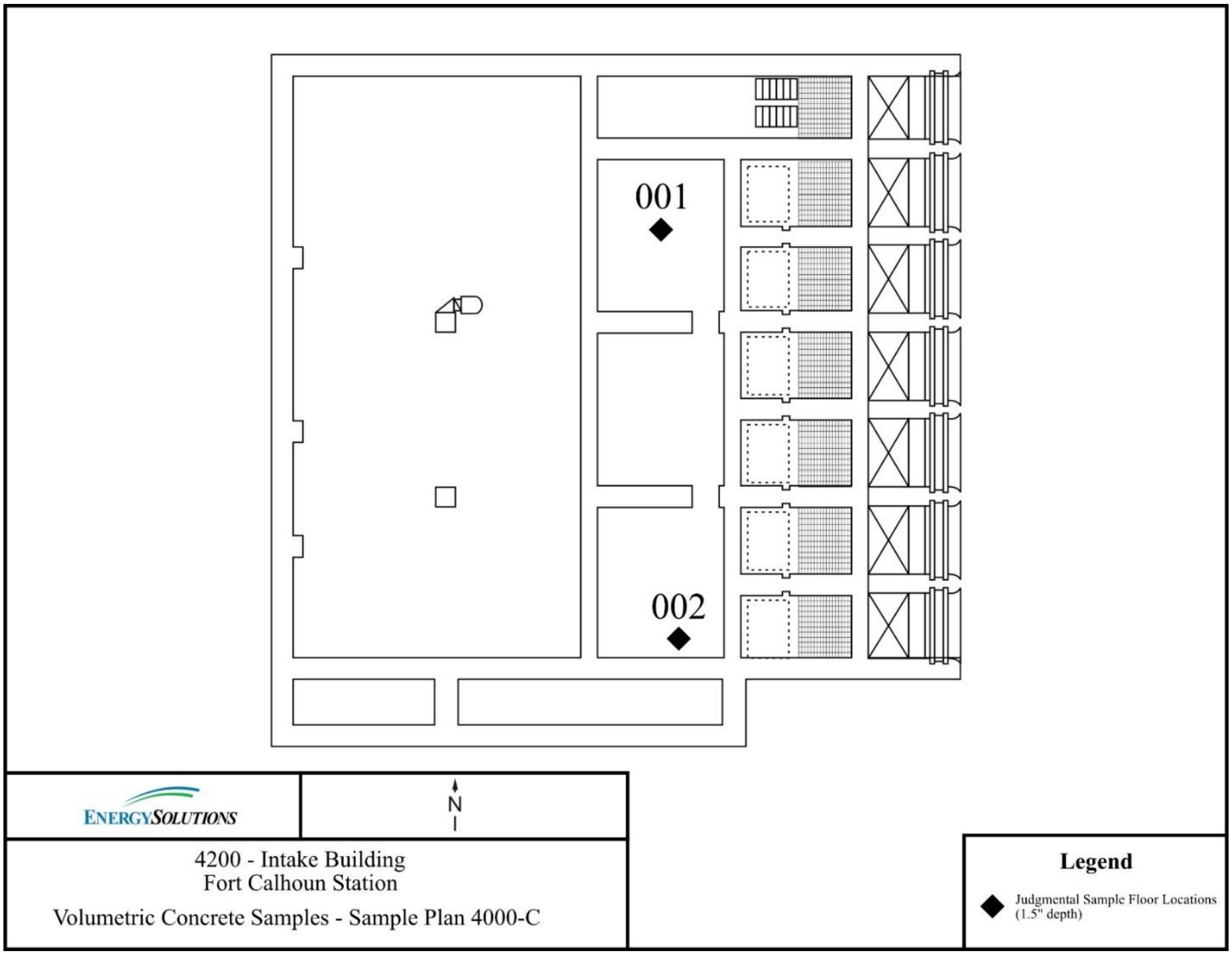


Figure 2-25 Volumetric Concrete Sample Location in Intake Building Basement

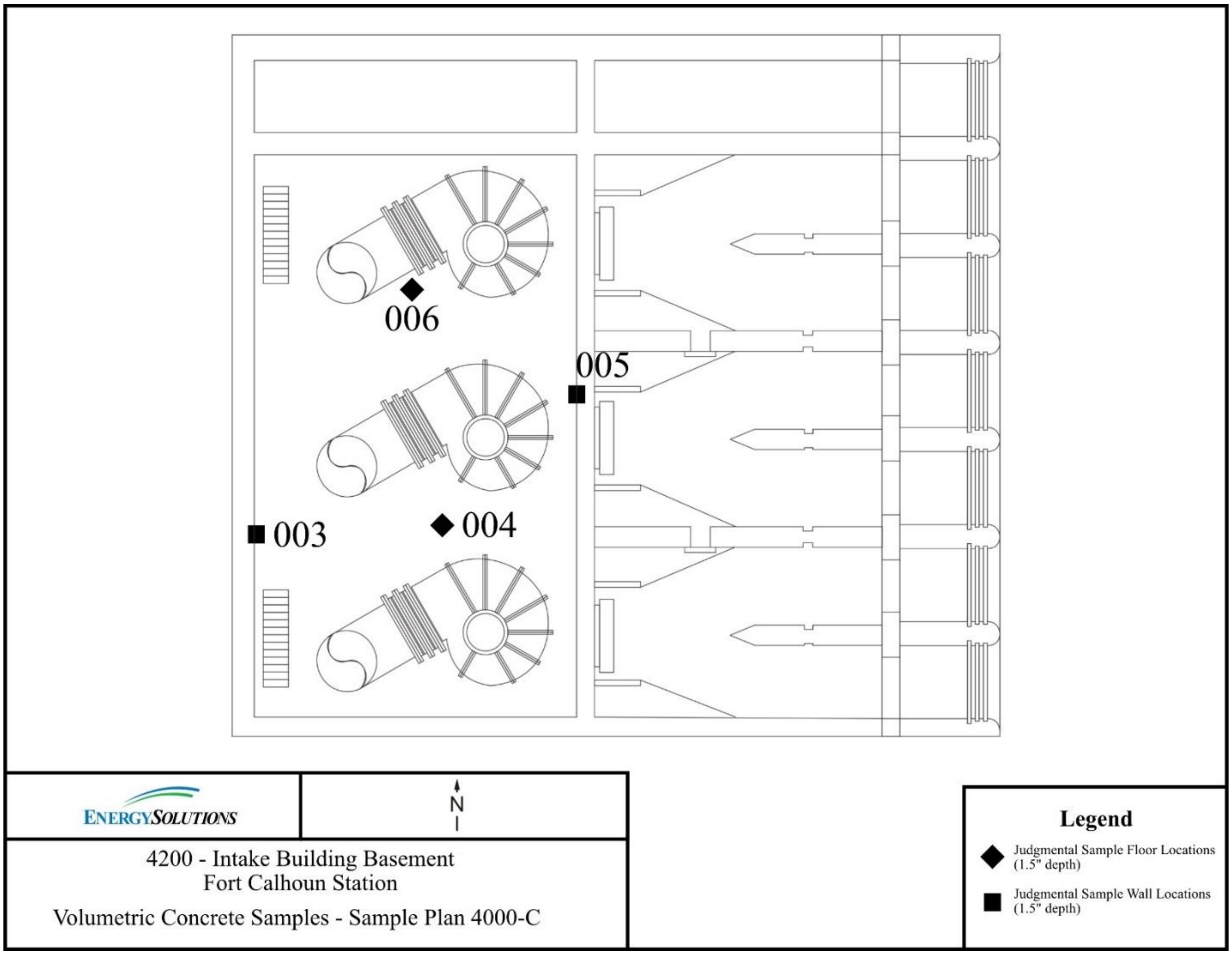


Table 2-14 1100 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1100X-1-CJ-FCV1-001	0.0 - 0.5	**9.09E+02**	**8.09E+00**	**1.49E+03**	**1.58E+02**	4.67E+01	9.27E-01	1.09E+01
1100X-1-CJ-FCV2-001	0.5 - 1.0	**4.09E+01**	**1.86E+00**	**5.61E+01**	**1.79E+02**	**1.40E+01**	2.46E+00	0.00E+00
1100X-1-CJ-FCV3-001	1.0 - 1.5	**2.56E+01**	0.00E+00	**2.09E+01**	**1.55E+02**	**7.36E+00**	2.24E+00	0.00E+00
1100X-1-CJ-FCV4-001	1.5 - 2.0	**3.64E+01**	1.39E-01	**7.42E+01**	**1.55E+02**	**8.67E+00**	7.70E-01	0.00E+00
1100X-1-CJ-FCV5-001	2.0 - 4.0	**1.68E+01**	**7.91E-01**	**9.21E+00**	**1.28E+02**	**6.94E+00**	6.21E-01	0.00E+00
1100X-1-CJ-FCV6-001	4.0 - 6.0	**9.54E+00**	**1.92E-01**	**8.30E+00**	**5.79E+01**	**3.36E+00**	7.33E-01	0.00E+00
Location 2								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1100X-1-CJ-FCV1-002	0.0 - 0.5	**8.39E+02**	**1.07E+02**	**1.32E+03**	**4.97E+03**	**5.03E+02**	**1.69E+01**	0.00E+00
1100X-1-CJ-FCV2-002	0.5 - 1.0	**6.58E+02**	**9.42E+01**	**1.19E+02**	**4.94E+03**	**4.67E+02**	**3.21E+01**	0.00E+00
1100X-1-CJ-FCV3-002	1.0 - 1.5	**8.18E+02**	**1.02E+02**	**1.46E+02**	**5.85E+03**	**5.36E+02**	**2.34E+01**	4.58E-01
1100X-1-CJ-FCV4-002	1.5 - 2.0	**8.06E+02**	**9.73E+01**	**1.32E+02**	**6.09E+03**	**5.30E+02**	0.00E+00	0.00E+00
1100X-1-CJ-FCV5-002	2.0 - 4.0	**1.04E+03**	**1.03E+02**	**1.60E+02**	**6.79E+03**	**5.64E+02**	6.48E+01	0.00E+00
1100X-1-CJ-FCV6-002	4.0 - 6.0	**3.21E+02**	**3.36E+01**	**3.39E+01**	**2.44E+03**	**1.90E+02**	1.98E+01	0.00E+00
Location 3								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1100X-1-CJ-WCV1-003	0.0 - 0.5	**5.18E+02**	**7.18E+01**	**9.45E+02**	**3.36E+03**	**3.18E+02**	2.91E+01	0.00E+00
1100X-1-CJ-WCV2-003	0.5 - 1.0	**5.15E+02**	**7.70E+01**	**7.64E+01**	**4.12E+03**	**3.91E+02**	**1.00E+02**	0.00E+00
1100X-1-CJ-WCV3-003	1.0 - 1.5	**6.30E+02**	**7.70E+01**	**7.94E+01**	**5.03E+03**	**4.30E+02**	**3.39E+01**	0.00E+00
1100X-1-CJ-WCV4-003	1.5 - 2.0	**6.06E+02**	**7.70E+01**	**6.73E+01**	**4.39E+03**	**3.82E+02**	3.79E+01	2.46E-01
1100X-1-CJ-WCV5-003	2.0 - 4.0	**6.85E+02**	**8.21E+01**	**3.45E+01**	**5.82E+03**	**4.45E+02**	5.61E+01	0.00E+00
1100X-1-CJ-WCV6-003	4.0 - 6.0	**5.21E+02**	**4.61E+01**	**4.85E+01**	**4.39E+03**	**3.03E+02**	**2.02E+01**	0.00E+00
Location 4								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1100X-1-CJ-WCV1-004	0.0 - 0.5	**1.42E+03**	**9.09E+01**	**3.82E+03**	**3.79E+03**	**3.82E+02**	3.01E+01	3.51E+00
1100X-1-CJ-WCV2-004	0.5 - 1.0	**7.03E+02**	**9.42E+01**	**1.23E+02**	**5.39E+03**	**4.91E+02**	**2.79E+01**	0.00E+00
1100X-1-CJ-WCV3-004	1.0 - 1.5	**7.09E+02**	**7.27E+01**	**7.91E+01**	**5.00E+03**	**4.12E+02**	3.88E+01	0.00E+00
1100X-1-CJ-WCV4-004	1.5 - 2.0	**8.27E+02**	**9.03E+01**	**4.03E+01**	**6.36E+03**	**4.88E+02**	5.15E+01	0.00E+00
1100X-1-CJ-WCV5-004	2.0 - 4.0	**7.94E+02**	**7.39E+01**	**1.96E+01**	**6.64E+03**	**4.67E+02**	**3.30E+01**	0.00E+00
1100X-1-CJ-WCV6-004	4.0 - 6.0	**4.88E+02**	**3.70E+01**	**4.42E+01**	**3.67E+03**	**2.11E+02**	0.00E+00	0.00E+00

Location 5								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1100X-1-CJ-FCV1-005	0.0 - 0.5	**7.54E+02**	**1.08E+02**	**3.61E+03**	**4.33E+03**	**4.27E+02**	1.63E+01	0.00E+00
1100X-1-CJ-FCV2-005	0.5 - 1.0	**6.24E+02**	**9.64E+01**	**2.08E+02**	**4.36E+03**	**3.88E+02**	**3.79E+01**	0.00E+00
1100X-1-CJ-FCV3-005	1.0 - 1.5	**5.88E+02**	**8.09E+01**	**8.48E+01**	**4.51E+03**	**3.64E+02**	**4.21E+01**	1.65E+00
1100X-1-CJ-FCV4-005	1.5 - 2.0	**5.39E+02**	**6.79E+01**	**1.74E+02**	**4.48E+03**	**3.48E+02**	**3.70E+01**	0.00E+00
1100X-1-CJ-FCV5-005	2.0 - 4.0	**5.70E+02**	**6.00E+01**	**5.39E+01**	**4.85E+03**	**3.67E+02**	**2.57E+01**	7.36E-01
1100X-1-CJ-FCV6-005	4.0 - 6.0	**2.69E+02**	**2.63E+01**	**2.63E+01**	**2.43E+03**	**1.74E+02**	2.10E+01	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-15 1100 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	9.54E+00	1.42E+03	5.76E+02	6.15E+02	3.27E+02
Cs-134	0.00E+00	1.08E+02	6.26E+01	7.54E+01	3.72E+01
Cs-137	8.30E+00	3.82E+03	4.37E+02	7.77E+01	9.64E+02
Eu-152	5.79E+01	6.79E+03	3.83E+03	4.39E+03	2.14E+03
Eu-154	3.36E+00	5.64E+02	3.22E+02	3.82E+02	1.83E+02
Eu-155	0.00E+00	1.00E+02	2.68E+01	2.68E+01	2.26E+01
Am-241	0.00E+00	1.09E+01	3.83E+03	0.00E+00	2.07E+00

Table 2-16 1200 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-001	0.0 - 0.5	**4.83E+01**	1.13E-01	**1.63E+02**	2.69E-01	1.51E-02	1.75E-01	0.00E+00
1200X-1-CJ-FCV2-001	0.5 - 1.0	**2.50E+00**	2.63E-02	**1.80E+01**	3.53E-01	2.38E-01	5.16E-02	0.00E+00
1200X-1-CJ-FCV3-001	1.0 - 1.5	**9.41E-01**	0.00E+00	**6.90E+00**	1.16E-01	0.00E+00	0.00E+00	1.19E-01
1200X-1-CJ-FCV4-001	1.5 - 2.0	**1.88E+00**	0.00E+00	**7.54E+00**	0.00E+00	1.07E-02	1.45E-01	0.00E+00
1200X-1-CJ-FCV5-001	2.0 - 4.0	**8.96E-01**	1.79E-01	**2.74E+00**	0.00E+00	1.59E-01	1.17E-01	0.00E+00
1200X-1-CJ-FCV6-001	4.0 - 6.0	**3.09E-01**	1.25E-01	**1.22E+00**	1.80E-01	1.91E-01	1.23E-01	4.21E-02
Location 2								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-002	0.0 - 0.5	**1.20E+02**	1.34E+00	**5.66E+02**	5.23E-01	9.33E-02	0.00E+00	1.96E+00
1200X-1-CJ-FCV2-002	0.5 - 1.0	**5.30E+00**	1.24E-01	**2.04E+01**	1.36E-01	0.00E+00	3.96E-01	0.00E+00
1200X-1-CJ-FCV3-002	1.0 - 1.5	**3.38E+00**	2.26E-01	**2.29E+01**	1.69E-01	6.46E-02	0.00E+00	3.21E-02
1200X-1-CJ-FCV4-002	1.5 - 2.0	**1.59E+00**	0.00E+00	**6.16E+00**	8.71E-02	0.00E+00	1.56E-03	0.00E+00
1200X-1-CJ-FCV5-002	2.0 - 4.0	**5.06E-01**	1.38E-01	**1.71E+00**	4.18E-02	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-FCV6-002	4.0 - 6.0	2.14E-01	0.00E+00	**1.07E+00**	0.00E+00	0.00E+00	5.56E-02	0.00E+00
Location 3								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-003	0.0 - 0.5	**3.81E+01**	**6.36E-01**	**1.16E+03**	0.00E+00	2.35E-01	0.00E+00	2.04E+00
1200X-1-CJ-FCV2-003	0.5 - 1.0	**6.85E+00**	2.13E-01	**1.78E+02**	5.49E-01	2.03E-01	1.73E-02	4.04E-01
1200X-1-CJ-FCV3-003	1.0 - 1.5	**1.68E+01**	**1.74E-01**	**2.79E+02**	6.41E-01	0.00E+00	2.71E-01	4.84E-01
1200X-1-CJ-FCV4-003	1.5 - 2.0	**6.39E+00**	2.40E-01	**1.20E+02**	0.00E+00	0.00E+00	3.43E-01	0.00E+00
1200X-1-CJ-FCV5-003	2.0 - 4.0	**9.08E-01**	9.11E-02	**6.39E+00**	2.31E-01	6.54E-02	2.46E-02	0.00E+00
1200X-1-CJ-FCV6-003	4.0 - 6.0	**1.94E+00**	2.14E-01	**1.14E+01**	0.00E+00	0.00E+00	4.55E-01	0.00E+00
Location 4								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-004	0.0 - 0.5	**1.73E+01**	0.00E+00	**9.70E+01**	3.80E-02	0.00E+00	4.26E-02	0.00E+00
1200X-1-CJ-FCV2-004	0.5 - 1.0	**5.96E-01**	0.00E+00	**2.95E+00**	5.14E-02	2.14E-01	4.20E-01	0.00E+00
1200X-1-CJ-FCV3-004	1.0 - 1.5	**6.19E-01**	2.04E-01	**2.86E+00**	0.00E+00	0.00E+00	1.02E-01	0.00E+00
1200X-1-CJ-FCV4-004	1.5 - 2.0	**1.06E+00**	4.63E-02	**5.76E+00**	0.00E+00	5.95E-02	3.21E-02	0.00E+00
1200X-1-CJ-FCV5-004	2.0 - 4.0	**2.18E-01**	0.00E+00	**9.64E-01**	0.00E+00	9.48E-02	0.00E+00	0.00E+00
1200X-1-CJ-FCV6-004	4.0 - 6.0	**3.96E-01**	2.05E-01	**2.93E+00**	0.00E+00	1.69E-01	1.64E-01	0.00E+00

Location 5								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-005	0.0 - 0.5	**1.90E+00**	0.00E+00	**1.05E+02**	**1.66E+00**	0.00E+00	3.03E-01	0.00E+00
1200X-1-CJ-WCV2-005	0.5 - 1.0	6.39E-01	0.00E+00	**9.36E+00**	**9.36E-01**	7.25E-01	1.76E-01	0.00E+00
1200X-1-CJ-WCV3-005	1.0 - 1.5	**3.05E-01**	0.00E+00	**1.55E+00**	**7.48E-01**	0.00E+00	1.69E-02	0.00E+00
1200X-1-CJ-WCV4-005	1.5 - 2.0	**3.85E-01**	0.00E+00	**2.84E+00**	**7.31E-01**	0.00E+00	0.00E+00	1.44E-01
1200X-1-CJ-WCV5-005	2.0 - 4.0	**9.80E-01**	0.00E+00	**1.29E+01**	**5.73E-01**	0.00E+00	6.01E-02	0.00E+00
1200X-1-CJ-WCV6-005	4.0 - 6.0	2.46E-01	0.00E+00	**2.18E+00**	1.81E-01	3.45E-01	1.07E-01	0.00E+00
Location 6								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-006	0.0 - 0.5	**9.61E-01**	0.00E+00	**4.56E+01**	**1.91E+00**	0.00E+00	0.00E+00	1.66E-01
1200X-1-CJ-WCV2-006	0.5 - 1.0	6.23E-01	0.00E+00	**8.23E+00**	**1.39E+00**	0.00E+00	4.93E-01	0.00E+00
1200X-1-CJ-WCV3-006	1.0 - 1.5	**4.84E-01**	0.00E+00	**2.33E+00**	**2.04E+00**	1.13E+00	7.98E-03	1.15E-02
1200X-1-CJ-WCV4-006	1.5 - 2.0	**3.94E-01**	0.00E+00	**1.93E+00**	**1.49E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV5-006	2.0 - 4.0	3.63E-01	1.88E-01	**1.46E+00**	**1.31E+00**	8.20E-01	2.14E-01	0.00E+00
1200X-1-CJ-WCV6-006	4.0 - 6.0	2.10E-01	1.15E-01	4.88E-01	**1.20E+00**	0.00E+00	6.70E-02	0.00E+00
Location 7								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-007	0.0 - 0.5	**8.16E+00**	4.71E-02	**1.06E+02**	7.53E-01	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-FCV2-007	0.5 - 1.0	**8.39E-01**	0.00E+00	**5.05E+00**	0.00E+00	0.00E+00	4.40E-02	1.98E-02
1200X-1-CJ-FCV3-007	1.0 - 1.5	**1.04E+00**	2.00E-01	**2.60E+00**	0.00E+00	8.06E-02	0.00E+00	0.00E+00
1200X-1-CJ-FCV4-007	1.5 - 2.0	**2.29E+00**	0.00E+00	**1.04E+01**	1.11E-01	0.00E+00	8.86E-02	0.00E+00
1200X-1-CJ-FCV5-007	2.0 - 4.0	2.98E-01	2.38E-02	**2.08E+00**	1.61E-01	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-FCV6-007	4.0 - 6.0	3.01E-01	3.20E-01	**1.45E+00**	0.00E+00	0.00E+00	2.04E-01	2.50E-02
Location 8								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-008	0.0 - 0.5	**2.55E+00**	1.28E-01	**1.56E+01**	**1.07E+00**	0.00E+00	6.24E-01	0.00E+00
1200X-1-CJ-WCV2-008	0.5 - 1.0	**4.65E-01**	0.00E+00	**1.39E+00**	**1.13E+00**	0.00E+00	2.18E-01	0.00E+00
1200X-1-CJ-WCV3-008	1.0 - 1.5	**4.50E-01**	0.00E+00	**1.41E+00**	**1.64E+00**	0.00E+00	2.96E-01	0.00E+00
1200X-1-CJ-WCV4-008	1.5 - 2.0	**3.39E-01**	0.00E+00	**1.90E+00**	**1.51E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV5-008	2.0 - 4.0	4.38E-01	0.00E+00	**1.90E+00**	**1.35E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV6-008	4.0 - 6.0	**3.33E-01**	0.00E+00	**1.12E+00**	**8.13E-01**	0.00E+00	7.05E-02	0.00E+00

Location 9								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-009	0.0 - 0.5	**1.05E+00**	0.00E+00	**3.00E+02**	**2.25E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV2-009	0.5 - 1.0	1.99E-01	0.00E+00	**3.04E+01**	**1.89E+00**	0.00E+00	9.16E-02	0.00E+00
1200X-1-CJ-WCV3-009	1.0 - 1.5	**3.69E-01**	0.00E+00	**4.04E+00**	**1.91E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV4-009	1.5 - 2.0	**4.25E-01**	0.00E+00	**1.68E+01**	**1.58E+00**	0.00E+00	3.38E-01	0.00E+00
1200X-1-CJ-WCV5-009	2.0 - 4.0	**5.08E-01**	0.00E+00	**5.88E+00**	**1.59E+00**	1.45E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV6-009	4.0 - 6.0	3.16E-01	1.38E-01	**4.03E+00**	**1.34E+00**	1.08E+00	2.00E-01	5.80E-02

Location 10								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-010	0.0 - 0.5	**1.31E+01**	8.21E-01	**3.74E+02**	1.71E+00	7.04E-02	0.00E+00	0.00E+00
1200X-1-CJ-FCV2-010	0.5 - 1.0	**1.10E+02**	**6.49E+00**	**6.81E+03**	0.00E+00	3.96E+00	5.61E+00	2.13E+00
1200X-1-CJ-FCV3-010	1.0 - 1.5	**2.38E+00**	0.00E+00	**2.04E+02**	1.14E+00	0.00E+00	0.00E+00	1.21E-01
1200X-1-CJ-FCV4-010	1.5 - 2.0	**9.13E-01**	3.73E-01	**5.41E+01**	**2.11E+00**	0.00E+00	4.74E-02	0.00E+00
1200X-1-CJ-FCV5-010	2.0 - 4.0	**1.53E+00**	4.66E-01	**6.00E+01**	**2.05E+00**	0.00E+00	6.89E-01	1.76E-01
1200X-1-CJ-FCV6-010	4.0 - 6.0	4.91E-01	0.00E+00	**2.65E+01**	**1.74E+00**	0.00E+00	0.00E+00	0.00E+00

Location 11								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-011	0.0 - 0.5	**5.54E-01**	0.00E+00	**1.10E+02**	**2.14E+00**	0.00E+00	3.84E-01	0.00E+00
1200X-1-CJ-WCV2-011	0.5 - 1.0	3.23E-01	0.00E+00	**1.70E+00**	**1.94E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV3-011	1.0 - 1.5	**3.63E-01**	0.00E+00	**1.76E+00**	**1.91E+00**	1.63E+00	2.33E-02	0.00E+00
1200X-1-CJ-WCV4-011	1.5 - 2.0	**4.50E-01**	0.00E+00	**2.29E+00**	**1.99E+00**	1.53E+00	2.19E-01	0.00E+00
1200X-1-CJ-WCV5-011	2.0 - 4.0	6.01E-01	1.17E-01	**1.50E+00**	**1.70E+00**	0.00E+00	2.44E-01	0.00E+00
1200X-1-CJ-WCV6-011	4.0 - 6.0	3.24E-01	0.00E+00	**1.20E+00**	**8.16E-01**	0.00E+00	0.00E+00	0.00E+00

Location 12								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-012	0.0 - 0.5	**8.69E-01**	0.00E+00	**6.28E+01**	**2.71E+00**	0.00E+00	4.43E-01	0.00E+00
1200X-1-CJ-WCV2-012	0.5 - 1.0	6.23E-01	0.00E+00	**2.33E+00**	**2.48E+00**	0.00E+00	2.35E-01	0.00E+00
1200X-1-CJ-WCV3-012	1.0 - 1.5	**4.39E-01**	0.00E+00	**1.75E+00**	**2.64E+00**	0.00E+00	3.14E-01	0.00E+00
1200X-1-CJ-WCV4-012	1.5 - 2.0	**6.44E-01**	0.00E+00	**3.91E+00**	**2.25E+00**	0.00E+00	1.11E-01	5.05E-02
1200X-1-CJ-WCV5-012	2.0 - 4.0	8.16E-01	4.99E-01	**5.34E+00**	**2.44E+00**	0.00E+00	2.19E-02	2.14E-01

Location 13								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-013	0.0 - 0.5	**1.33E+00**	0.00E+00	**1.99E+01**	**1.05E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV2-013	0.5 - 1.0	2.46E-01	0.00E+00	1.03E+00	1.25E+00	1.00E+00	2.21E-01	0.00E+00
1200X-1-CJ-WCV3-013	1.0 - 1.5	3.64E-01	3.89E-02	1.88E+00	8.91E-01	0.00E+00	1.28E-01	0.00E+00
1200X-1-CJ-WCV4-013	1.5 - 2.0	4.23E-01	3.83E-02	2.33E+00	9.04E-01	7.50E-01	1.38E-01	0.00E+00
1200X-1-CJ-WCV5-013	2.0 - 4.0	2.19E-01	1.31E-02	1.80E+00	9.93E-01	6.91E-01	2.74E-01	0.00E+00
1200X-1-CJ-WCV6-013	4.0 - 6.0	2.38E-01	0.00E+00	1.59E+00	8.06E-01	0.00E+00	1.14E-01	0.00E+00
Location 14								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-014	0.0 - 0.5	**2.68E+01**	4.08E-01	**9.31E+01**	7.78E-02	0.00E+00	6.43E-01	0.00E+00
1200X-1-CJ-FCV2-014	0.5 - 1.0	**7.96E-01**	2.01E-01	**3.03E+00**	0.00E+00	4.39E-02	0.00E+00	2.96E-02
1200X-1-CJ-FCV3-014	1.0 - 1.5	**5.79E-01**	9.35E-02	**1.20E+00**	1.25E-01	4.34E-02	1.14E-01	0.00E+00
1200X-1-CJ-FCV4-014	1.5 - 2.0	**5.43E-01**	9.56E-02	**2.28E+00**	0.00E+00	0.00E+00	3.09E-02	1.45E-01
1200X-1-CJ-FCV5-014	2.0 - 4.0	2.73E-01	0.00E+00	**8.45E-01**	0.00E+00	4.83E-02	0.00E+00	0.00E+00
1200X-1-CJ-FCV6-014	4.0 - 6.0	0.00E+00	7.85E-02	**5.98E-01**	1.96E-02	5.11E-02	0.00E+00	0.00E+00
Location 15								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-015	0.0 - 0.5	**5.66E-01**	0.00E+00	**2.71E+01**	**1.53E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV2-015	0.5 - 1.0	**3.38E-01**	0.00E+00	**1.18E+00**	**1.70E+00**	0.00E+00	2.58E-01	0.00E+00
1200X-1-CJ-WCV3-015	1.0 - 1.5	2.83E-01	0.00E+00	**9.66E-01**	**1.56E+00**	1.11E+00	6.66E-02	0.00E+00
1200X-1-CJ-WCV4-015	1.5 - 2.0	4.16E-01	0.00E+00	**1.30E+00**	**1.54E+00**	1.17E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV5-015	2.0 - 4.0	**3.34E-01**	0.00E+00	**1.39E+00**	**9.15E-01**	8.50E-01	7.29E-02	7.43E-02
1200X-1-CJ-WCV6-015	4.0 - 6.0	**3.91E-01**	0.00E+00	**2.50E+00**	**8.79E-01**	0.00E+00	2.16E-01	0.00E+00
Location 16								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-016	0.0 - 0.5	**1.31E+00**	6.85E-02	**1.10E+02**	1.39E-01	3.73E-01	0.00E+00	5.54E-02
1200X-1-CJ-FCV2-016	0.5 - 1.0	1.35E-01	0.00E+00	**2.14E+00**	3.06E-01	1.01E-01	0.00E+00	0.00E+00
1200X-1-CJ-FCV3-016	1.0 - 1.5	**2.04E-01**	1.90E-01	**2.45E+01**	1.51E-01	5.15E-03	1.39E-01	1.15E-01
1200X-1-CJ-FCV4-016	1.5 - 2.0	2.56E-01	0.00E+00	**1.29E+01**	2.55E-01	3.14E-01	0.00E+00	0.00E+00
1200X-1-CJ-FCV5-016	2.0 - 4.0	1.74E-01	0.00E+00	**5.53E-01**	1.74E-01	9.68E-02	1.46E-01	0.00E+00
1200X-1-CJ-FCV6-016	4.0 - 6.0	**2.58E-01**	0.00E+00	**1.69E+00**	0.00E+00	0.00E+00	6.59E-02	3.09E-02

Location 17								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-017	0.0 - 0.5	**8.28E-01**	3.01E-01	**2.99E+01**	8.31E-01	3.66E-01	5.56E-02	0.00E+00
1200X-1-CJ-FCV2-017	0.5 - 1.0	2.41E-01	7.28E-02	**7.09E+00**	2.36E-01	1.54E-01	1.04E-01	0.00E+00
1200X-1-CJ-FCV3-017	1.0 - 1.5	**1.88E+00**	0.00E+00	**1.79E+02**	2.68E-02	0.00E+00	3.46E-01	0.00E+00
1200X-1-CJ-FCV4-017	1.5 - 2.0	**3.85E-01**	0.00E+00	**1.07E+01**	3.54E-01	3.16E-02	0.00E+00	4.43E-02
1200X-1-CJ-FCV5-017	2.0 - 4.0	2.01E-01	1.79E-01	**5.19E+00**	1.41E-01	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-FCV6-017	4.0 - 6.0	3.66E-01	0.00E+00	**7.06E+00**	1.33E-01	0.00E+00	0.00E+00	0.00E+00
Location 18								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-018	0.0 - 0.5	**1.76E+00**	0.00E+00	**2.65E+02**	0.00E+00	6.36E-01	5.06E-01	0.00E+00
1200X-1-CJ-FCV2-018	0.5 - 1.0	**1.35E+00**	0.00E+00	**1.08E+01**	1.06E-01	1.69E-02	3.26E-02	0.00E+00
1200X-1-CJ-FCV3-018	1.0 - 1.5	**9.74E-01**	9.44E-02	**9.15E+00**	6.54E-03	0.00E+00	1.06E-01	2.73E-02
1200X-1-CJ-FCV4-018	1.5 - 2.0	2.16E-01	0.00E+00	**3.66E+00**	0.00E+00	0.00E+00	3.13E-01	0.00E+00
1200X-1-CJ-FCV5-018	2.0 - 4.0	9.74E-02	0.00E+00	**2.10E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-FCV6-018	4.0 - 6.0	1.35E-01	0.00E+00	**1.49E+00**	7.54E-02	3.64E-02	2.23E-01	0.00E+00
Location 19								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-019	0.0 - 0.5	**6.63E-01**	3.89E-01	**3.08E+01**	1.13E+00	6.66E-01	5.71E-01	0.00E+00
1200X-1-CJ-FCV2-019	0.5 - 1.0	2.20E-01	1.60E-01	**2.80E+00**	**1.02E+00**	5.03E-01	0.00E+00	0.00E+00
1200X-1-CJ-FCV3-019	1.0 - 1.5	3.20E-01	0.00E+00	**1.94E+00**	**9.23E-01**	6.10E-01	3.16E-01	9.83E-03
1200X-1-CJ-FCV4-019	1.5 - 2.0	3.73E-01	0.00E+00	**4.90E+00**	**9.56E-01**	8.48E-01	9.90E-02	0.00E+00
1200X-1-CJ-FCV5-019	2.0 - 4.0	**2.00E-01**	0.00E+00	**1.86E+00**	**7.93E-01**	0.00E+00	1.78E-02	8.36E-02
1200X-1-CJ-FCV6-019	4.0 - 6.0	3.40E-01	2.85E-01	**2.96E+00**	6.46E-01	4.68E-01	0.00E+00	0.00E+00
Location 20								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-020	0.0 - 0.5	**7.48E-01**	0.00E+00	**3.41E+01**	**2.39E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV2-020	0.5 - 1.0	**2.98E-01**	0.00E+00	**1.08E+00**	**2.48E+00**	0.00E+00	5.60E-01	0.00E+00
1200X-1-CJ-WCV3-020	1.0 - 1.5	**5.73E-01**	0.00E+00	**1.41E+00**	**2.60E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV4-020	1.5 - 2.0	5.29E-01	0.00E+00	**3.33E+00**	**2.28E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV5-020	2.0 - 4.0	**3.56E-01**	1.28E-02	**1.04E+00**	**1.95E+00**	0.00E+00	1.04E-01	0.00E+00
1200X-1-CJ-WCV6-020	4.0 - 6.0	1.89E-01	3.04E-02	**5.93E-01**	**1.15E+00**	0.00E+00	8.46E-02	0.00E+00

Location 21								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-021	0.0 - 0.5	5.38E-01	0.00E+00	**2.04E+01**	**1.07E+00**	0.00E+00	0.00E+00	1.58E-01
1200X-1-CJ-WCV2-021	0.5 - 1.0	4.31E-01	4.26E-01	**1.14E+00**	**1.70E+00**	0.00E+00	3.33E-01	0.00E+00
1200X-1-CJ-WCV3-021	1.0 - 1.5	3.51E-01	0.00E+00	**1.03E+00**	**1.49E+00**	1.07E+00	5.46E-01	4.38E-02
1200X-1-CJ-WCV4-021	1.5 - 2.0	**8.35E-01**	5.08E-01	**5.73E+01**	**1.51E+00**	0.00E+00	2.05E-01	0.00E+00
1200X-1-CJ-WCV5-021	2.0 - 4.0	4.11E-01	0.00E+00	**2.75E+00**	**1.64E+00**	1.04E+00	3.29E-01	0.00E+00
1200X-1-CJ-WCV6-021	4.0 - 6.0	2.58E-01	0.00E+00	**2.38E+00**	**9.96E-01**	0.00E+00	3.04E-01	0.00E+00
Location 22								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-022	0.0 - 0.5	4.03E-01	3.51E-01	**5.74E+01**	**1.64E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV2-022	0.5 - 1.0	**5.34E-01**	4.89E-01	**4.58E+00**	**1.44E+00**	0.00E+00	3.49E-01	0.00E+00
1200X-1-CJ-WCV3-022	1.0 - 1.5	1.44E-01	3.54E-01	**1.65E+00**	**1.56E+00**	1.19E+00	0.00E+00	4.35E-02
1200X-1-CJ-WCV4-022	1.5 - 2.0	4.61E-01	3.33E-01	**3.88E+00**	**1.69E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV5-022	2.0 - 4.0	3.54E-01	0.00E+00	**8.54E-01**	**1.24E+00**	0.00E+00	3.75E-01	0.00E+00
1200X-1-CJ-WCV6-022	4.0 - 6.0	1.07E-01	0.00E+00	**1.18E+00**	**9.19E-01**	0.00E+00	2.56E-01	0.00E+00
Location 23								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-023	0.0 - 0.5	**7.84E+01**	**3.00E+00**	**6.69E+03**	1.15E+00	2.39E+00	0.00E+00	3.06E-01
1200X-1-CJ-FCV2-023	0.5 - 1.0	**3.60E+00**	5.16E-01	**1.55E+02**	**1.39E+00**	0.00E+00	6.65E-04	0.00E+00
1200X-1-CJ-FCV3-023	1.0 - 1.5	**6.71E-01**	0.00E+00	**3.03E+01**	**1.35E+00**	1.00E+00	1.93E-01	0.00E+00
1200X-1-CJ-FCV4-023	1.5 - 2.0	**9.55E-01**	2.46E-01	**3.46E+01**	**1.78E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-FCV5-023	2.0 - 4.0	**4.83E-01**	1.25E-01	**2.56E+01**	**1.51E+00**	1.23E+00	0.00E+00	0.00E+00
1200X-1-CJ-FCV6-023	4.0 - 6.0	2.95E-01	0.00E+00	**1.30E+01**	**8.74E-01**	0.00E+00	2.59E-01	0.00E+00
Location 24								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-024	0.0 - 0.5	**4.96E+00**	6.56E-01	**3.64E+02**	0.00E+00	0.00E+00	1.10E+00	0.00E+00
1200X-1-CJ-WCV2-024	0.5 - 1.0	2.54E-01	1.04E-01	**2.78E+01**	1.53E+00	1.08E+00	3.21E-02	0.00E+00
1200X-1-CJ-WCV3-024	1.0 - 1.5	3.83E-01	0.00E+00	**1.73E+00**	**1.21E+00**	0.00E+00	2.40E-01	0.00E+00
1200X-1-CJ-WCV4-024	1.5 - 2.0	1.08E+00	2.33E-01	**4.71E+01**	1.11E+00	4.39E-01	3.38E-01	2.68E-01
1200X-1-CJ-WCV5-024	2.0 - 4.0	2.49E-01	9.71E-02	**1.38E+01**	**1.14E+00**	0.00E+00	3.63E-01	5.01E-02
1200X-1-CJ-WCV6-024	4.0 - 6.0	1.44E-01	2.21E-01	**1.78E+00**	9.74E-01	4.03E-01	1.35E-01	0.00E+00

Location 25								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-025	0.0 - 0.5	9.94E+00	0.00E+00	**9.56E+02**	2.93E-01	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-FCV2-025	0.5 - 1.0	**5.41E-01**	4.53E-02	**1.14E+01**	0.00E+00	1.30E-01	1.78E-02	5.61E-02
1200X-1-CJ-FCV3-025	1.0 - 1.5	**2.51E+00**	3.38E-01	**5.66E+01**	0.00E+00	1.81E-01	3.43E-01	0.00E+00
1200X-1-CJ-FCV4-025	1.5 - 2.0	**6.96E-01**	0.00E+00	**2.58E+01**	4.33E-01	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-FCV5-025	2.0 - 4.0	7.20E-02	0.00E+00	**1.94E+00**	4.34E-01	1.05E-01	6.31E-02	0.00E+00
1200X-1-CJ-FCV6-025	4.0 - 6.0	1.81E-01	0.00E+00	**1.88E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Location 26								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-026	0.0 - 0.5	**6.74E-01**	6.23E-01	**2.46E+02**	1.06E+00	7.24E-01	0.00E+00	0.00E+00
1200X-1-CJ-WCV2-026	0.5 - 1.0	3.04E-01	2.10E-01	**9.48E+00**	7.98E-01	4.88E-01	1.88E-01	0.00E+00
1200X-1-CJ-WCV3-026	1.0 - 1.5	**5.19E-01**	7.33E-02	**5.18E+00**	5.15E-01	2.39E-01	0.00E+00	0.00E+00
1200X-1-CJ-WCV4-026	1.5 - 2.0	3.83E-01	0.00E+00	**6.65E+00**	5.00E-01	4.25E-01	0.00E+00	2.64E-02
1200X-1-CJ-WCV5-026	2.0 - 4.0	3.18E-01	0.00E+00	**5.60E+00**	**3.85E-01**	0.00E+00	5.93E-02	0.00E+00
1200X-1-CJ-WCV6-026	4.0 - 6.0	2.99E-01	0.00E+00	**2.33E+00**	5.88E-01	2.33E-01	4.73E-02	0.00E+00
Location 27								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-FCV1-027	0.0 - 0.5	**2.04E+01**	**3.21E+00**	**5.13E+03**	0.00E+00	0.00E+00	1.25E+00	2.25E+00
1200X-1-CJ-FCV2-027	0.5 - 1.0	4.95E-01	1.43E-01	**2.49E+01**	1.01E+00	7.43E-01	0.00E+00	0.00E+00
1200X-1-CJ-FCV3-027	1.0 - 1.5	**1.51E+00**	7.64E-02	**1.44E+02**	1.35E+00	3.68E-01	1.31E-01	0.00E+00
1200X-1-CJ-FCV4-027	1.5 - 2.0	**1.50E+02**	**2.65E+01**	**1.71E+04**	0.00E+00	3.14E+00	2.70E+00	0.00E+00
1200X-1-CJ-FCV5-027	2.0 - 4.0	**7.98E+00**	**1.08E+00**	**7.63E+02**	0.00E+00	2.24E-01	1.03E+00	0.00E+00
1200X-1-CJ-FCV6-027	4.0 - 6.0	**5.41E-01**	0.00E+00	**1.78E+01**	3.43E-01	1.79E-01	0.00E+00	1.20E-02
Location 28								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1200X-1-CJ-WCV1-028	0.0 - 0.5	**7.76E+00**	2.56E-01	**1.35E+02**	1.68E+00	1.49E+00	1.18E+00	6.78E-02
1200X-1-CJ-WCV2-028	0.5 - 1.0	**9.89E-01**	4.94E-01	**1.09E+02**	**1.68E+00**	0.00E+00	5.35E-01	0.00E+00
1200X-1-CJ-WCV3-028	1.0 - 1.5	**2.80E+00**	4.28E-01	**1.06E+01**	**1.53E+00**	0.00E+00	4.50E-01	0.00E+00
1200X-1-CJ-WCV4-028	1.5 - 2.0	7.01E-01	1.24E-01	**1.10E+01**	**1.99E+00**	0.00E+00	4.94E-02	0.00E+00
1200X-1-CJ-WCV5-028	2.0 - 4.0	**2.29E-01**	0.00E+00	**3.36E+00**	**1.21E+00**	0.00E+00	1.02E-01	0.00E+00
1200X-1-CJ-WCV6-028	4.0 - 6.0	**4.09E-01**	0.00E+00	**4.45E+00**	**9.76E-01**	0.00E+00	1.07E-01	0.00E+00

Location 29								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1200X-1-CJ-WCV1-029	0.0 - 0.5	**5.49E-01**	0.00E+00	**1.59E+01**	**1.49E+00**	0.00E+00	8.11E-02	0.00E+00
1200X-1-CJ-WCV2-029	0.5 - 1.0	2.24E-01	0.00E+00	**1.31E+00**	**2.28E+00**	0.00E+00	6.88E-02	0.00E+00
1200X-1-CJ-WCV3-029	1.0 - 1.5	**2.98E-01**	2.51E-01	**1.54E+00**	**1.91E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV4-029	1.5 - 2.0	4.65E-01	1.11E-02	**3.79E+00**	**1.71E+00**	0.00E+00	8.70E-02	0.00E+00
1200X-1-CJ-WCV5-029	2.0 - 4.0	4.78E-01	0.00E+00	**2.99E+00**	**1.06E+00**	0.00E+00	0.00E+00	0.00E+00
1200X-1-CJ-WCV6-029	4.0 - 6.0	**7.88E-01**	0.00E+00	**5.71E+00**	3.66E-01	3.59E-01	0.00E+00	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-17 1200 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	1.20E+02	4.61E+00	4.91E-01	1.83E+01
Cs-134	0.00E+00	2.65E+01	3.38E-01	0.00E+00	2.09E+00
Cs-137	4.88E-01	1.71E+04	2.57E+02	5.19E+00	1.53E+03
Eu-152	0.00E+00	2.71E+00	9.27E-01	9.19E-01	7.66E-01
Eu-154	0.00E+00	3.96E+00	2.57E-01	0.00E+00	5.45E-01
Eu-155	0.00E+00	5.61E+00	2.01E-01	6.88E-02	5.08E-01
Am-241	0.00E+00	2.25E+00	9.27E-01	0.00E+00	3.19E-01

Table 2-18 1300 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1300X-1-CJ-FCV1-001	0.0 - 0.5	**1.18E+00**	4.90E-02	**7.78E+01**	0.00E+00	9.53E-02	0.00E+00	0.00E+00
1300X-1-CJ-FCV2-001	0.5 - 1.0	1.51E-01	1.64E-01	**1.68E+00**	0.00E+00	0.00E+00	1.24E-01	2.30E-01
1300X-1-CJ-FCV3-001	1.0 - 1.5	0.00E+00	0.00E+00	**2.25E+00**	5.14E-02	0.00E+00	1.68E-01	0.00E+00
1300X-1-CJ-FCV4-001	1.5 - 2.0	3.88E-01	0.00E+00	**4.16E+00**	2.80E-01	2.36E-01	1.22E-01	0.00E+00
1300X-1-CJ-FCV5-001	2.0 - 4.0	1.88E-01	0.00E+00	**3.26E+00**	8.41E-02	0.00E+00	1.71E-01	1.28E-02
1300X-1-CJ-FCV6-001	4.0 - 6.0	1.96E-01	0.00E+00	**5.26E-01**	2.30E-02	0.00E+00	2.05E-01	0.00E+00
Location 2								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1300X-1-CJ-WCV1-002	0.0 - 0.5	2.44E-01	1.93E-01	**4.14E+00**	1.34E-01	0.00E+00	0.00E+00	0.00E+00
1300X-1-CJ-WCV2-002	0.5 - 1.0	1.84E-01	0.00E+00	**4.11E-01**	0.00E+00	0.00E+00	6.25E-02	2.19E-02
1300X-1-CJ-WCV3-002	1.0 - 1.5	9.06E-02	0.00E+00	**2.71E+00**	9.05E-02	0.00E+00	1.35E-01	0.00E+00
1300X-1-CJ-WCV4-002	1.5 - 2.0	6.01E-02	7.63E-02	**2.40E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1300X-1-CJ-WCV5-002	2.0 - 4.0	2.54E-01	0.00E+00	**2.79E+00**	1.78E-01	2.30E-01	3.33E-01	0.00E+00
1300X-1-CJ-WCV6-002	4.0 - 6.0	1.53E-01	0.00E+00	**1.64E+00**	0.00E+00	8.24E-02	0.00E+00	0.00E+00
Location 3								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1300X-1-CJ-FCV1-003	0.0 - 0.5	**1.93E+01**	**4.10E+00**	**6.36E+03**	0.00E+00	0.00E+00	2.68E+00	2.30E-01
1300X-1-CJ-FCV2-003	0.5 - 1.0	**5.19E-01**	3.61E-02	**1.26E+02**	0.00E+00	5.00E-01	0.00E+00	2.84E-01
1300X-1-CJ-FCV3-003	1.0 - 1.5	5.71E-02	0.00E+00	**4.50E+01**	0.00E+00	5.34E-01	2.06E-01	0.00E+00
1300X-1-CJ-FCV4-003	1.5 - 2.0	**4.40E-01**	0.00E+00	**7.83E+01**	2.40E-02	6.18E-02	1.85E-01	0.00E+00
1300X-1-CJ-FCV5-003	2.0 - 4.0	5.04E-02	0.00E+00	**1.06E+00**	2.40E-01	1.09E-01	2.26E-01	2.89E-02
1300X-1-CJ-FCV6-003	4.0 - 6.0	9.48E-02	0.00E+00	**2.15E+00**	2.40E-01	3.44E-02	2.99E-01	0.00E+00
Location 4								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1300X-1-CJ-WCV1-004	0.0 - 0.5	2.74E-01	0.00E+00	**1.25E+01**	0.00E+00	0.00E+00	2.30E-01	0.00E+00
1300X-1-CJ-WCV2-004	0.5 - 1.0	2.54E-02	3.75E-03	**3.10E-01**	0.00E+00	1.07E-01	2.70E-01	0.00E+00
1300X-1-CJ-WCV3-004	1.0 - 1.5	1.38E-01	0.00E+00	4.46E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1300X-1-CJ-WCV4-004	1.5 - 2.0	2.25E-02	0.00E+00	**3.45E-01**	0.00E+00	0.00E+00	4.41E-02	0.00E+00
1300X-1-CJ-WCV5-004	2.0 - 4.0	1.09E-01	0.00E+00	**2.70E-01**	1.80E-01	7.39E-02	5.95E-02	0.00E+00
1300X-1-CJ-WCV6-004	4.0 - 6.0	1.33E-01	2.41E-01	6.69E-01	4.45E-01	1.50E-01	9.69E-02	0.00E+00

Location 5								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1300X-1-CJ-FCV1-005	0.0 - 0.5	**2.28E+00**	5.40E-02	**1.03E+02**	2.86E-01	2.34E-01	3.39E-01	0.00E+00
1300X-1-CJ-FCV2-005	0.5 - 1.0	2.03E-01	1.85E-01	**4.29E+00**	6.06E-01	7.51E-02	1.39E-01	0.00E+00
1300X-1-CJ-FCV3-005	1.0 - 1.5	3.18E-03	0.00E+00	**1.39E+00**	2.38E-01	2.28E-01	1.40E-01	0.00E+00
1300X-1-CJ-FCV4-005	1.5 - 2.0	1.21E-01	0.00E+00	**1.25E+00**	7.70E-02	1.21E-01	3.51E-02	0.00E+00
1300X-1-CJ-FCV5-005	2.0 - 4.0	1.03E-01	0.00E+00	**4.85E-01**	2.83E-01	1.75E-01	2.55E-01	0.00E+00
1300X-1-CJ-FCV6-005	4.0 - 6.0	1.75E-02	1.01E-01	**4.48E-01**	1.78E-01	0.00E+00	1.60E-01	0.00E+00
Location 6								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1300X-1-CJ-FCV1-006	0.0 - 0.5	**1.85E+00**	**6.46E-01**	**1.05E+03**	2.06E-01	3.08E-01	0.00E+00	0.00E+00
1300X-1-CJ-FCV2-006	0.5 - 1.0	**2.88E-01**	0.00E+00	**7.50E+01**	0.00E+00	2.81E-01	6.33E-01	4.68E-02
1300X-1-CJ-FCV3-006	1.0 - 1.5	1.59E-01	0.00E+00	**3.39E+00**	1.74E-01	1.50E-02	0.00E+00	0.00E+00
1300X-1-CJ-FCV4-006	1.5 - 2.0	6.51E-02	0.00E+00	**1.56E+01**	2.01E-01	1.21E-01	0.00E+00	0.00E+00
1300X-1-CJ-FCV5-006	2.0 - 4.0	2.50E-02	0.00E+00	**9.88E-01**	1.86E-02	0.00E+00	7.91E-02	1.71E-02
1300X-1-CJ-FCV6-006	4.0 - 6.0	1.96E-01	0.00E+00	**1.53E+00**	1.24E-01	6.31E-02	0.00E+00	0.00E+00
Location 7								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1300X-1-CJ-WCV1-007	0.0 - 0.5	**3.10E-01**	0.00E+00	**6.75E+01**	1.71E-02	4.76E-01	0.00E+00	0.00E+00
1300X-1-CJ-WCV2-007	0.5 - 1.0	7.26E-02	0.00E+00	**7.83E-01**	**6.34E-01**	0.00E+00	0.00E+00	0.00E+00
1300X-1-CJ-WCV3-007	1.0 - 1.5	2.39E-01	1.05E-01	**1.09E+00**	**7.38E-01**	3.31E-01	0.00E+00	0.00E+00
1300X-1-CJ-WCV4-007	1.5 - 2.0	2.76E-01	0.00E+00	**2.14E+00**	**7.80E-01**	0.00E+00	1.89E-01	0.00E+00
1300X-1-CJ-WCV5-007	2.0 - 4.0	2.15E-01	0.00E+00	**6.56E-01**	**6.60E-01**	0.00E+00	3.11E-01	1.12E-01
1300X-1-CJ-WCV6-007	4.0 - 6.0	1.03E-01	3.91E-02	**5.06E-01**	3.74E-01	3.94E-01	4.98E-02	0.00E+00
Location 8								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1300X-1-CJ-FCV1-008	0.0 - 0.5	**2.56E+00**	**8.00E-01**	**1.75E+03**	0.00E+00	0.00E+00	9.04E-01	0.00E+00
1300X-1-CJ-FCV2-008	0.5 - 1.0	2.31E-01	1.78E-01	**1.21E+01**	**4.96E-01**	0.00E+00	1.30E-01	0.00E+00
1300X-1-CJ-FCV3-008	1.0 - 1.5	**1.95E-01**	0.00E+00	**1.41E+01**	4.15E-01	3.19E-01	3.34E-01	0.00E+00
1300X-1-CJ-FCV4-008	1.5 - 2.0	1.91E-01	0.00E+00	**5.00E+00**	4.39E-01	1.73E-01	3.70E-01	0.00E+00
1300X-1-CJ-FCV5-008	2.0 - 4.0	2.48E-01	0.00E+00	**2.49E+00**	**3.44E-01**	3.89E-02	2.38E-01	0.00E+00
1300X-1-CJ-FCV6-008	4.0 - 6.0	2.04E-01	0.00E+00	**2.83E+00**	4.04E-01	1.63E-01	3.06E-02	0.00E+00

Location 9								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1300X-1-CJ-FCV1-009	0.0 - 0.5	**5.59E-01**	2.86E-01	**4.40E+01**	0.00E+00	2.69E-01	0.00E+00	0.00E+00
1300X-1-CJ-FCV2-009	0.5 - 1.0	2.13E-01	0.00E+00	**1.70E+00**	1.51E-01	2.89E-02	0.00E+00	0.00E+00
1300X-1-CJ-FCV3-009	1.0 - 1.5	7.44E-02	0.00E+00	**2.45E+00**	1.33E-01	6.13E-02	2.30E-01	9.23E-02
1300X-1-CJ-FCV4-009	1.5 - 2.0	8.43E-02	0.00E+00	**1.45E+00**	0.00E+00	2.93E-02	0.00E+00	0.00E+00
1300X-1-CJ-FCV5-009	2.0 - 4.0	1.22E-01	0.00E+00	**9.88E-01**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1300X-1-CJ-FCV6-009	4.0 - 6.0	1.04E-01	2.25E-01	3.46E-01	2.71E-01	1.30E-01	2.14E-01	0.00E+00
Location 10								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1300X-1-CJ-WCV1-010	0.0 - 0.5	2.68E-01	0.00E+00	**2.13E+00**	0.00E+00	1.22E-01	1.74E-02	0.00E+00
1300X-1-CJ-WCV2-010	0.5 - 1.0	3.09E-01	0.00E+00	1.16E-01	1.65E-01	4.61E-02	0.00E+00	0.00E+00
1300X-1-CJ-WCV3-010	1.0 - 1.5	1.22E-01	0.00E+00	2.90E-01	8.54E-02	4.69E-02	0.00E+00	0.00E+00
1300X-1-CJ-WCV4-010	1.5 - 2.0	4.00E-02	9.59E-02	**3.99E-01**	3.56E-02	0.00E+00	3.16E-02	0.00E+00
1300X-1-CJ-WCV5-010	2.0 - 4.0	1.50E-01	0.00E+00	**3.61E-01**	0.00E+00	0.00E+00	1.86E-01	0.00E+00
1300X-1-CJ-WCV6-010	4.0 - 6.0	2.40E-01	0.00E+00	**2.66E-01**	0.00E+00	0.00E+00	1.25E-01	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-19 1300 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	1.93E+01	6.11E-01	1.86E-01	2.49E+00
Cs-134	0.00E+00	4.10E+00	1.26E-01	0.00E+00	5.41E-01
Cs-137	1.16E-01	6.36E+03	1.65E+02	2.13E+00	8.54E+02
Eu-152	0.00E+00	7.80E-01	1.75E-01	1.07E-01	2.09E-01
Eu-154	0.00E+00	5.34E-01	1.08E-01	5.41E-02	1.39E-01
Eu-155	0.00E+00	2.68E+00	1.79E-01	1.23E-01	3.67E-01
Am-241	0.00E+00	2.84E-01	1.75E-01	0.00E+00	5.71E-02

Table 2-20 1400 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-WMTX-001	0	**5.99E-01**	6.34E-02	**3.16E+00**	0.00E+00	1.41E-02	7.41E-02	0.00E+00
1400X-1-CJ-WCV1-001	0.0 - 0.5	1.94E-01	7.88E-02	**7.69E-01**	2.23E-01	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-WCV2-001	0.5 - 1.0	0.00E+00	2.69E-01	**9.45E-01**	6.01E-01	4.68E-01	0.00E+00	0.00E+00
1400X-1-CJ-WCV3-001	1.0 - 1.5	2.10E-01	0.00E+00	**6.10E-01**	0.00E+00	4.96E-02	0.00E+00	0.00E+00
1400X-1-CJ-WCV4-001	1.5 - 2.0	8.51E-02	0.00E+00	**6.19E-01**	2.48E-01	1.65E-01	0.00E+00	0.00E+00
1400X-1-CJ-WCV5-001	2.0 - 4.0	8.28E-02	0.00E+00	**5.80E-01**	4.98E-02	0.00E+00	2.54E-01	0.00E+00
1400X-1-CJ-WCV6-001	4.0 - 6.0	0.00E+00	0.00E+00	**5.91E-01**	3.29E-01	1.93E-02	2.38E-01	2.45E-02
Location 2								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-FCV1-002	0.0 - 0.5	**3.33E+00**	**7.19E-01**	**3.05E+03**	0.00E+00	6.30E-01	0.00E+00	0.00E+00
1400X-1-CJ-FCV2-002	0.5 - 1.0	1.75E-01	0.00E+00	**2.20E+01**	7.20E-01	4.60E-01	5.10E-01	0.00E+00
1400X-1-CJ-FCV3-002	1.0 - 1.5	0.00E+00	1.15E-01	**8.45E+01**	2.74E-01	2.46E-01	1.40E-01	0.00E+00
1400X-1-CJ-FCV4-002	1.5 - 2.0	2.30E-01	0.00E+00	**4.48E+01**	1.05E-01	2.19E-01	1.79E-01	0.00E+00
1400X-1-CJ-FCV5-002	2.0 - 4.0	1.64E-01	1.89E-02	**3.11E+01**	0.00E+00	1.51E-01	0.00E+00	0.00E+00
Location 3								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-WCV1-003	0.0 - 0.5	1.66E-01	0.00E+00	**5.24E+00**	8.64E-02	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-WCV2-003	0.5 - 1.0	2.34E-01	0.00E+00	2.09E-01	1.04E-01	2.14E-02	6.65E-02	2.04E-03
1400X-1-CJ-WCV3-003	1.0 - 1.5	5.95E-03	0.00E+00	1.59E-01	1.16E-01	3.69E-02	0.00E+00	1.14E-01
1400X-1-CJ-WCV4-003	1.5 - 2.0	3.08E-01	0.00E+00	**1.94E+00**	1.91E-02	8.55E-02	0.00E+00	0.00E+00
1400X-1-CJ-WCV5-003	2.0 - 4.0	3.73E-01	5.63E-02	**6.58E-01**	1.76E-01	0.00E+00	3.88E-01	0.00E+00
1400X-1-CJ-WCV6-003	4.0 - 6.0	2.29E-01	0.00E+00	2.29E-01	2.40E-01	0.00E+00	2.01E-01	2.19E-02
Location 4								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-WCV1-004	0.0 - 0.5	1.54E+00	5.39E-02	**3.86E+00**	3.23E-01	0.00E+00	3.63E-02	0.00E+00
1400X-1-CJ-WCV2-004	0.5 - 1.0	**8.33E-01**	0.00E+00	**4.80E-01**	3.00E-01	2.11E-01	2.23E-01	0.00E+00
1400X-1-CJ-WCV3-004	1.0 - 1.5	2.55E-02	0.00E+00	1.59E-01	3.21E-01	0.00E+00	9.81E-02	0.00E+00
1400X-1-CJ-WCV4-004	1.5 - 2.0	1.61E-01	0.00E+00	**5.29E-01**	5.40E-02	0.00E+00	4.08E-01	6.23E-04
1400X-1-CJ-WCV5-004	2.0 - 4.0	**1.79E+01**	0.00E+00	3.49E-01	1.94E-01	0.00E+00	6.00E-02	0.00E+00
1400X-1-CJ-WCV6-004	4.0 - 6.0	1.64E-01	3.60E-02	3.89E-01	9.14E-02	0.00E+00	0.00E+00	0.00E+00

Location 5								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-FCV1-005	0.0 - 0.5	**9.99E+00**	0.00E+00	**1.31E+02**	8.54E-01	5.71E-01	1.44E-02	0.00E+00
1400X-1-CJ-FCV2-005	0.5 - 1.0	3.06E-01	1.71E-01	**1.34E+00**	3.55E-01	0.00E+00	0.00E+00	1.18E-01
1400X-1-CJ-FCV3-005	1.0 - 1.5	**2.83E-01**	2.13E-01	**5.40E+00**	0.00E+00	6.40E-02	2.11E-01	0.00E+00
1400X-1-CJ-FCV4-005	1.5 - 2.0	**5.03E-01**	0.00E+00	**3.70E+00**	1.63E-01	1.48E-01	3.44E-01	0.00E+00
1400X-1-CJ-FCV5-005	2.0 - 4.0	**2.33E-01**	0.00E+00	**5.50E+00**	3.34E-01	1.39E-01	1.48E-01	0.00E+00
1400X-1-CJ-FCV6-005	4.0 - 6.0	1.63E-01	4.26E-02	**7.81E-01**	1.05E-01	1.95E-01	0.00E+00	0.00E+00
Location 6								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-WMTX-006	0	**5.54E-01**	1.93E-02	**1.29E+00**	0.00E+00	0.00E+00	0.00E+00	1.03E-02
1400X-1-CJ-WCV1-006	0.0 - 0.5	3.76E-01	2.83E-01	**2.73E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-WCV2-006	0.5 - 1.0	2.95E-01	0.00E+00	**6.29E-01**	3.19E-01	5.86E-02	5.08E-01	0.00E+00
1400X-1-CJ-WCV3-006	1.0 - 1.5	2.76E-01	1.85E-01	**7.85E-01**	1.55E-01	2.53E-01	7.80E-02	0.00E+00
1400X-1-CJ-WCV4-006	1.5 - 2.0	1.20E-01	0.00E+00	**8.34E-01**	1.38E-01	1.58E-01	0.00E+00	0.00E+00
1400X-1-CJ-WCV5-006	2.0 - 4.0	1.81E-01	0.00E+00	**6.10E-01**	4.29E-02	4.88E-03	9.53E-02	0.00E+00
1400X-1-CJ-WCV6-006	4.0 - 6.0	8.83E-02	1.94E-01	**2.99E-01**	1.46E-01	7.83E-02	7.29E-02	0.00E+00
Location 7								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-FCV1-007	0.0 - 0.5	3.58E-01	0.00E+00	**5.55E+01**	4.56E-01	6.13E-02	2.46E-01	0.00E+00
1400X-1-CJ-FCV2-007	0.5 - 1.0	1.43E-01	0.00E+00	**1.26E+00**	6.39E-02	1.81E-01	1.02E-01	0.00E+00
1400X-1-CJ-FCV3-007	1.0 - 1.5	0.00E+00	0.00E+00	**1.15E+00**	2.63E-01	5.74E-02	4.15E-01	0.00E+00
1400X-1-CJ-FCV4-007	1.5 - 2.0	5.50E-02	0.00E+00	**1.60E+00**	2.23E-01	1.59E-01	1.80E-01	0.00E+00
1400X-1-CJ-FCV5-007	2.0 - 4.0	7.14E-02	0.00E+00	**4.16E-01**	0.00E+00	5.61E-02	1.93E-01	0.00E+00
1400X-1-CJ-FCV6-007	4.0 - 6.0	6.15E-02	0.00E+00	**2.10E-01**	2.05E-01	0.00E+00	2.23E-01	0.00E+00
Location 8								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-FCV1-008	0.0 - 0.5	2.40E-01	1.19E-01	**8.23E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-FCV2-008	0.5 - 1.0	1.31E-01	3.21E-01	2.18E-01	9.63E-02	1.35E-01	0.00E+00	0.00E+00
1400X-1-CJ-FCV3-008	1.0 - 1.5	1.30E-01	0.00E+00	1.98E-01	1.66E-01	1.69E-01	0.00E+00	0.00E+00
1400X-1-CJ-FCV4-008	1.5 - 2.0	7.69E-02	0.00E+00	**5.28E-01**	5.03E-02	3.24E-02	0.00E+00	1.75E-01
1400X-1-CJ-FCV5-008	2.0 - 4.0	1.16E-01	2.03E-01	**8.84E-02**	9.65E-02	1.25E-01	0.00E+00	2.44E-02
1400X-1-CJ-FCV6-008	4.0 - 6.0	1.11E-01	0.00E+00	2.26E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Location 9								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-WMTX-009	0	**1.39E-01**	9.73E-02	**1.30E+00**	0.00E+00	0.00E+00	3.80E-02	0.00E+00
1400X-1-CJ-WCV1-009	0.0 - 0.5	2.81E-01	2.75E-01	**3.28E+00**	0.00E+00	0.00E+00	1.35E-01	0.00E+00
1400X-1-CJ-WCV2-009	0.5 - 1.0	2.43E-01	0.00E+00	3.54E-01	8.21E-02	1.81E-01	0.00E+00	0.00E+00
1400X-1-CJ-WCV3-009	1.0 - 1.5	3.64E-01	4.09E-01	4.50E-01	0.00E+00	3.20E-02	0.00E+00	0.00E+00
1400X-1-CJ-WCV4-009	1.5 - 2.0	**6.40E-01**	0.00E+00	**1.21E+00**	1.94E-01	3.53E-02	1.23E-02	9.31E-02
1400X-1-CJ-WCV5-009	2.0 - 4.0	1.06E-01	6.79E-02	3.10E-01	1.30E-01	6.80E-02	0.00E+00	0.00E+00
1400X-1-CJ-WCV6-009	4.0 - 6.0	1.07E-01	2.38E-01	5.81E-02	1.18E-01	2.54E-01	1.35E-01	0.00E+00
Location 10								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-FCV1-010	0.0 - 0.5	6.91E-01	0.00E+00	**2.15E+01**	**1.71E+00**	0.00E+00	0.00E+00	3.19E-01
1400X-1-CJ-FCV2-010	0.5 - 1.0	3.34E-01	0.00E+00	**4.19E-01**	**1.85E+00**	0.00E+00	2.10E-01	5.48E-02
1400X-1-CJ-FCV3-010	1.0 - 1.5	**2.84E-01**	1.26E-01	**3.49E-01**	**1.71E+00**	0.00E+00	4.40E-01	0.00E+00
1400X-1-CJ-FCV4-010	1.5 - 2.0	**1.89E-01**	0.00E+00	**1.09E+00**	**1.25E+00**	1.03E+00	2.73E-01	7.23E-02
1400X-1-CJ-FCV5-010	2.0 - 4.0	2.70E-01	4.90E-02	**2.58E-01**	**9.70E-01**	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-FCV6-010	4.0 - 6.0	2.63E-01	0.00E+00	**3.01E-01**	**7.94E-01**	0.00E+00	2.56E-01	0.00E+00
Location 11								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-FCV1-011	0.0 - 0.5	**2.30E+00**	0.00E+00	**4.18E+01**	**3.33E+00**	0.00E+00	7.39E-01	5.98E-01
1400X-1-CJ-FCV2-011	0.5 - 1.0	**7.25E-01**	0.00E+00	**3.04E+00**	**3.79E+00**	0.00E+00	4.84E-01	0.00E+00
1400X-1-CJ-FCV3-011	1.0 - 1.5	**5.79E-01**	3.49E-02	**1.55E+00**	**3.65E+00**	2.49E+00	2.31E-01	0.00E+00
1400X-1-CJ-FCV4-011	1.5 - 2.0	**5.04E-01**	4.71E-01	**1.19E+01**	**2.93E+00**	0.00E+00	3.20E-01	0.00E+00
1400X-1-CJ-FCV5-011	2.0 - 4.0	**3.81E-01**	0.00E+00	**6.80E-01**	**2.00E+00**	0.00E+00	2.28E-01	0.00E+00
1400X-1-CJ-FCV6-011	4.0 - 6.0	**4.78E-01**	0.00E+00	**4.35E-01**	**2.50E+00**	1.90E+00	1.85E-01	0.00E+00
Location 12								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-FCV1-012	0.0 - 0.5	**1.73E+01**	2.09E-01	**7.91E+02**	2.20E+00	1.81E+00	2.65E-01	3.84E-01
1400X-1-CJ-FCV2-012	0.5 - 1.0	**2.01E+00**	2.38E-01	**9.94E+01**	**3.08E+00**	0.00E+00	5.98E-01	0.00E+00
1400X-1-CJ-FCV3-012	1.0 - 1.5	**1.75E+00**	0.00E+00	**4.05E+01**	**3.19E+00**	0.00E+00	0.00E+00	1.64E-01
1400X-1-CJ-FCV4-012	1.5 - 2.0	**5.98E-01**	0.00E+00	**1.08E+01**	**2.46E+00**	1.56E+00	0.00E+00	3.29E-02
1400X-1-CJ-FCV5-012	2.0 - 4.0	**6.85E-01**	1.94E-01	**2.44E+00**	**2.24E+00**	1.61E+00	1.29E-01	0.00E+00
1400X-1-CJ-FCV6-012	4.0 - 6.0	3.84E-01	2.55E-01	**7.04E-01**	**1.76E+00**	0.00E+00	3.09E-01	0.00E+00

Location 13								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-WCV1-013	0.0 - 0.5	**1.03E+00**	2.09E-01	**2.00E+01**	1.30E+00	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-WCV2-013	0.5 - 1.0	3.29E-01	7.53E-02	**2.29E-01**	**1.01E+00**	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-WCV3-013	1.0 - 1.5	3.46E-01	0.00E+00	2.99E-01	**1.02E+00**	1.18E-03	1.51E-01	8.29E-02
1400X-1-CJ-WCV4-013	1.5 - 2.0	**1.65E-01**	0.00E+00	**5.76E-01**	**1.01E+00**	6.56E-01	0.00E+00	0.00E+00
1400X-1-CJ-WCV5-013	2.0 - 4.0	3.56E-01	0.00E+00	2.04E-01	1.53E+00	0.00E+00	6.35E-02	0.00E+00
1400X-1-CJ-WCV6-013	4.0 - 6.0	**2.45E-01**	0.00E+00	**2.26E-01**	1.04E+00	0.00E+00	1.83E-01	1.03E-01
Location 14								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-FCV1-014	0.0 - 0.5	**6.39E+00**	4.76E-02	**6.60E+01**	**2.59E+00**	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-FCV2-014	0.5 - 1.0	**7.96E-01**	2.35E-01	**2.80E+00**	**2.44E+00**	0.00E+00	2.58E-01	0.00E+00
1400X-1-CJ-FCV3-014	1.0 - 1.5	**4.36E-01**	4.01E-02	**1.64E+00**	**2.78E+00**	0.00E+00	0.00E+00	4.69E-02
1400X-1-CJ-FCV4-014	1.5 - 2.0	**5.88E-01**	0.00E+00	**4.44E+00**	**2.05E+00**	0.00E+00	5.38E-02	0.00E+00
1400X-1-CJ-FCV5-014	2.0 - 4.0	**3.59E-01**	0.00E+00	**1.15E+00**	**1.56E+00**	0.00E+00	1.25E-01	0.00E+00
1400X-1-CJ-FCV6-014	4.0 - 6.0	1.95E-01	0.00E+00	**1.88E+00**	**8.63E-01**	0.00E+00	4.20E-01	0.00E+00
Location 15								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-FCV1-015	0.0 - 0.5	**2.05E+00**	0.00E+00	**5.78E+01**	**2.08E+00**	0.00E+00	0.00E+00	2.71E-01
1400X-1-CJ-FCV2-015	0.5 - 1.0	**4.88E-01**	0.00E+00	**2.39E+00**	**2.70E+00**	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-FCV3-015	1.0 - 1.5	4.03E-01	0.00E+00	**8.24E-01**	**2.76E+00**	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-FCV4-015	1.5 - 2.0	**3.85E-01**	0.00E+00	**2.03E+00**	**2.05E+00**	1.79E+00	1.89E-01	6.15E-02
1400X-1-CJ-FCV5-015	2.0 - 4.0	**6.25E-01**	2.75E-01	4.40E-01	**1.93E+00**	0.00E+00	3.58E-01	1.56E-02
1400X-1-CJ-FCV6-015	4.0 - 6.0	**1.90E+00**	1.31E-02	3.89E-01	**1.38E+00**	0.00E+00	0.00E+00	0.00E+00
Location 16								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-WCV1-016	0.0 - 0.5	**1.22E+00**	0.00E+00	**1.30E+01**	**1.90E+00**	0.00E+00	2.84E-01	0.00E+00
1400X-1-CJ-WCV2-016	0.5 - 1.0	3.33E-01	2.60E-01	**3.00E-01**	**1.61E+00**	0.00E+00	5.66E-01	0.00E+00
1400X-1-CJ-WCV3-016	1.0 - 1.5	**4.00E-01**	3.40E-01	**6.69E-01**	**2.13E+00**	0.00E+00	6.43E-02	0.00E+00
1400X-1-CJ-WCV4-016	1.5 - 2.0	**7.04E-01**	0.00E+00	**8.18E-01**	**1.85E+00**	0.00E+00	2.30E-02	2.35E-03
1400X-1-CJ-WCV5-016	2.0 - 4.0	**3.58E-01**	0.00E+00	**9.95E-02**	**1.51E+00**	0.00E+00	3.76E-01	0.00E+00
1400X-1-CJ-WCV6-016	4.0 - 6.0	2.43E-01	2.45E-01	4.81E-01	**1.14E+00**	8.03E-01	3.86E-01	0.00E+00

Location 17								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-WCV1-017	0.0 - 0.5	**1.71E+00**	5.18E-02	**7.84E+00**	**1.20E+00**	0.00E+00	0.00E+00	6.18E-02
1400X-1-CJ-WCV2-017	0.5 - 1.0	**1.25E+00**	0.00E+00	**1.21E+00**	**1.28E+00**	0.00E+00	9.66E-02	0.00E+00
1400X-1-CJ-WCV3-017	1.0 - 1.5	**1.58E+00**	1.49E-01	**4.14E-01**	**1.34E+00**	0.00E+00	3.05E-01	0.00E+00
1400X-1-CJ-WCV4-017	1.5 - 2.0	**4.63E-01**	2.48E-01	**3.09E-01**	**1.20E+00**	7.30E-01	5.28E-01	0.00E+00
1400X-1-CJ-WCV5-017	2.0 - 4.0	**7.01E-01**	1.48E-01	**2.99E-01**	**1.53E+00**	1.04E+00	1.66E-01	1.63E-01
1400X-1-CJ-WCV6-017	4.0 - 6.0	**9.86E-01**	0.00E+00	**8.96E-01**	**1.15E+00**	0.00E+00	0.00E+00	0.00E+00

Location 18								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-FCV1-018	0.0 - 0.5	**1.54E+01**	0.00E+00	**2.65E+01**	**8.06E-01**	9.61E-01	4.53E-01	4.48E-01
1400X-1-CJ-FCV2-018	0.5 - 1.0	**9.56E-01**	0.00E+00	**1.08E+00**	**1.31E+00**	0.00E+00	3.05E-01	0.00E+00
1400X-1-CJ-FCV3-018	1.0 - 1.5	**3.05E-01**	0.00E+00	**5.21E-01**	**1.33E+00**	0.00E+00	1.26E-01	0.00E+00
1400X-1-CJ-FCV4-018	1.5 - 2.0	**9.39E-01**	2.20E-01	**8.39E-01**	**9.36E-01**	0.00E+00	1.34E-01	0.00E+00
1400X-1-CJ-FCV5-018	2.0 - 4.0	**8.79E-01**	0.00E+00	**2.53E-01**	**9.04E-01**	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-FCV6-018	4.0 - 6.0	**7.25E-01**	7.08E-02	3.33E-01	4.95E-01	4.88E-01	0.00E+00	0.00E+00

Location 19								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-WMTX-019	0	**4.10E-01**	7.64E-02	**1.80E+00**	0.00E+00	0.00E+00	3.24E-02	0.00E+00
1400X-1-CJ-WCV1-019	0.0 - 0.5	2.96E-01	2.26E-01	**3.78E+00**	5.71E-01	4.08E-01	0.00E+00	1.02E-01
1400X-1-CJ-WCV2-019	0.5 - 1.0	3.75E-01	3.53E-01	5.71E-01	2.81E-01	2.41E-01	0.00E+00	3.86E-02
1400X-1-CJ-WCV3-019	1.0 - 1.5	2.35E-01	3.01E-01	**5.14E-01**	2.46E-01	1.94E-01	4.61E-01	0.00E+00
1400X-1-CJ-WCV4-019	1.5 - 2.0	0.00E+00	0.00E+00	**6.45E-01**	9.80E-02	6.50E-02	4.89E-02	2.06E-02
1400X-1-CJ-WCV5-019	2.0 - 4.0	3.20E-01	0.00E+00	**2.95E-01**	3.89E-01	1.00E-01	0.00E+00	0.00E+00
1400X-1-CJ-WCV6-019	4.0 - 6.0	4.75E-01	0.00E+00	**3.23E-01**	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Location 20								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
1400X-1-CJ-WMTX-020	0	**7.31E-01**	4.10E-02	**2.70E-01**	5.44E-02	3.24E-02	0.00E+00	1.43E-01
1400X-1-CJ-WCV1-020	0.0 - 0.5	**2.08E-01**	0.00E+00	**1.10E+01**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-WCV2-020	0.5 - 1.0	2.59E-01	0.00E+00	2.23E-01	2.21E-01	5.01E-02	2.31E-01	0.00E+00
1400X-1-CJ-WCV3-020	1.0 - 1.5	2.09E-01	0.00E+00	4.01E-01	1.04E-01	4.16E-02	6.08E-02	0.00E+00
1400X-1-CJ-WCV4-020	1.5 - 2.0	2.19E-01	0.00E+00	**1.83E+00**	2.93E-02	1.68E-01	2.11E-01	0.00E+00
1400X-1-CJ-WCV5-020	2.0 - 4.0	2.54E-01	0.00E+00	0.00E+00	1.43E-01	4.75E-02	7.34E-02	1.35E-02
1400X-1-CJ-WCV6-020	4.0 - 6.0	2.26E-02	1.38E-02	1.29E-01	1.00E-01	8.83E-02	1.63E-01	0.00E+00

Location 21								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-FCV1-021	0.0 - 0.5	**4.44E+00**	0.00E+00	**1.05E+02**	0.00E+00	0.00E+00	1.34E-01	2.41E-01
1400X-1-CJ-FCV2-021	0.5 - 1.0	1.61E-01	2.55E-01	**2.03E+00**	3.50E-01	2.36E-01	6.76E-01	0.00E+00
1400X-1-CJ-FCV3-021	1.0 - 1.5	**4.98E-01**	0.00E+00	**1.00E+00**	3.75E-01	0.00E+00	0.00E+00	1.66E-01
1400X-1-CJ-FCV4-021	1.5 - 2.0	3.56E-01	0.00E+00	**7.46E+00**	4.21E-01	0.00E+00	3.01E-01	0.00E+00
1400X-1-CJ-FCV5-021	2.0 - 4.0	3.09E-01	1.05E-01	**2.19E+00**	2.30E-01	1.38E-02	0.00E+00	0.00E+00
Location 22								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-WCV1-022	0.0 - 0.5	**1.48E+00**	5.39E-02	**1.48E+01**	4.30E-02	6.30E-02	6.58E-02	9.23E-03
1400X-1-CJ-WCV2-022	0.5 - 1.0	**1.68E-01**	0.00E+00	**3.10E-01**	4.00E-01	2.09E-01	1.93E-01	0.00E+00
1400X-1-CJ-WCV3-022	1.0 - 1.5	1.75E-01	1.19E-01	**1.89E-01**	6.80E-02	0.00E+00	7.48E-03	0.00E+00
1400X-1-CJ-WCV4-022	1.5 - 2.0	**4.85E-01**	0.00E+00	**9.44E-01**	2.66E-01	1.31E-01	8.66E-02	0.00E+00
1400X-1-CJ-WCV5-022	2.0 - 4.0	**2.40E-01**	0.00E+00	4.41E-01	5.41E-01	1.53E-01	0.00E+00	0.00E+00
1400X-1-CJ-WCV6-022	4.0 - 6.0	**4.83E-01**	0.00E+00	1.07E-01	3.00E-01	3.31E-01	1.14E-01	0.00E+00
Location 23								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-FCV1-023	0.0 - 0.5	**1.39E+01**	**5.90E-01**	**4.95E+02**	1.35E+00	6.95E-01	3.25E-02	1.03E-01
1400X-1-CJ-FCV2-023	0.5 - 1.0	**1.86E+00**	3.09E-01	**7.76E+01**	1.22E+00	6.48E-01	0.00E+00	4.61E-02
1400X-1-CJ-FCV3-023	1.0 - 1.5	**1.02E+00**	2.78E-01	**4.35E+01**	6.73E-01	4.46E-01	2.91E-01	0.00E+00
1400X-1-CJ-FCV4-023	1.5 - 2.0	**8.03E-01**	0.00E+00	**2.21E+01**	4.76E-01	3.53E-01	2.45E-01	0.00E+00
1400X-1-CJ-FCV5-023	2.0 - 4.0	**1.58E+02**	4.58E-02	**1.15E+00**	2.91E-01	4.21E-01	1.19E-01	3.04E-01
1400X-1-CJ-FCV6-023	4.0 - 6.0	**9.24E-01**	0.00E+00	**2.14E+00**	3.04E-01	3.01E-01	0.00E+00	0.00E+00
Location 24								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-FCV1-024	0.0 - 0.5	**6.58E-01**	0.00E+00	**3.26E+01**	0.00E+00	1.89E-01	0.00E+00	0.00E+00
1400X-1-CJ-FCV2-024	0.5 - 1.0	4.51E-02	1.84E-02	**9.81E-01**	0.00E+00	1.06E-01	1.48E-01	0.00E+00
1400X-1-CJ-FCV3-024	1.0 - 1.5	1.73E-01	1.61E-01	2.53E-01	3.89E-01	1.65E-01	1.24E-01	0.00E+00
1400X-1-CJ-FCV4-024	1.5 - 2.0	**2.45E-01**	0.00E+00	**2.49E+00**	1.39E-01	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-FCV5-024	2.0 - 4.0	1.91E-01	0.00E+00	**2.41E-01**	2.10E-02	0.00E+00	0.00E+00	1.49E-02
1400X-1-CJ-FCV6-024	4.0 - 6.0	5.18E-01	1.00E-01	2.84E-01	0.00E+00	0.00E+00	2.38E-01	0.00E+00

Location 25								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-FCV1-025	0.0 - 0.5	**1.39E+00**	2.64E-01	**6.18E+02**	0.00E+00	1.51E+00	1.08E+00	0.00E+00
1400X-1-CJ-FCV2-025	0.5 - 1.0	2.23E-01	0.00E+00	**4.61E+00**	3.11E-01	1.85E-01	2.23E-01	0.00E+00
1400X-1-CJ-FCV3-025	1.0 - 1.5	**3.51E-01**	0.00E+00	**4.06E+00**	4.45E-01	3.00E-01	0.00E+00	1.85E-02
1400X-1-CJ-FCV4-025	1.5 - 2.0	2.53E-01	0.00E+00	**1.49E+01**	2.73E-01	5.16E-02	0.00E+00	0.00E+00
1400X-1-CJ-FCV5-025	2.0 - 4.0	4.00E-01	0.00E+00	**9.13E-01**	1.34E-01	0.00E+00	0.00E+00	0.00E+00
1400X-1-CJ-FCV6-025	4.0 - 6.0	3.34E-01	2.15E-01	**1.09E+00**	0.00E+00	0.00E+00	1.24E-01	0.00E+00
Location 26								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-WMTX-026	0	**2.28E+00**	2.71E-02	**4.70E+00**	0.00E+00	1.16E-01	1.00E-01	0.00E+00
1400X-1-CJ-WCV1-026	0.0 - 0.5	**5.11E-01**	2.86E-01	**2.49E+00**	1.28E-01	2.36E-02	0.00E+00	0.00E+00
1400X-1-CJ-WCV2-026	0.5 - 1.0	**2.23E-01**	0.00E+00	3.26E-01	6.11E-01	4.14E-01	4.68E-01	0.00E+00
1400X-1-CJ-WCV3-026	1.0 - 1.5	2.43E-01	0.00E+00	1.66E-01	3.71E-01	2.81E-01	3.81E-02	0.00E+00
1400X-1-CJ-WCV4-026	1.5 - 2.0	4.08E-01	0.00E+00	**4.38E-01**	7.19E-01	3.91E-01	4.34E-01	0.00E+00
1400X-1-CJ-WCV5-026	2.0 - 4.0	**4.64E-01**	0.00E+00	3.81E-01	4.78E-01	2.41E-01	8.58E-02	7.08E-02
1400X-1-CJ-WCV6-026	4.0 - 6.0	**7.18E-01**	5.35E-02	1.70E-01	5.68E-02	2.16E-01	0.00E+00	0.00E+00
Location 27								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CJ-FCV1-027	0.0 - 0.5	**1.09E+01**	3.53E-01	**4.81E+02**	2.59E-01	6.33E-01	6.53E-01	2.54E-01
1400X-1-CJ-FCV2-027	0.5 - 1.0	**8.16E-01**	0.00E+00	**1.46E+01**	3.55E-01	4.46E-01	0.00E+00	5.60E-02
1400X-1-CJ-FCV3-027	1.0 - 1.5	2.51E-01	9.45E-04	**4.74E+00**	6.61E-01	1.98E-01	2.88E-02	0.00E+00
1400X-1-CJ-FCV4-027	1.5 - 2.0	**2.24E+00**	2.99E-01	**2.68E+01**	2.05E-01	3.56E-01	0.00E+00	0.00E+00
1400X-1-CJ-FCV5-027	2.0 - 4.0	**5.88E-01**	7.05E-02	**6.20E-01**	4.90E-01	2.16E-01	1.08E-01	0.00E+00
1400X-1-CJ-FCV6-027	4.0 - 6.0	**3.36E-01**	1.24E-01	**8.00E-01**	1.30E-01	1.68E-01	2.09E-01	9.80E-03

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-21 1400 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	1.79E+01	2.01E+00	3.54E-01	1.24E+01
Cs-134	0.00E+00	7.19E-01	7.94E-02	0.00E+00	1.27E-01
Cs-137	0.00E+00	3.05E+03	4.06E+01	9.04E-01	2.53E+02
Eu-152	0.00E+00	3.79E+00	7.36E-01	3.20E-01	8.86E-01
Eu-154	0.00E+00	2.49E+00	2.08E-01	4.46E-02	4.03E-01
Eu-155	0.00E+00	1.08E+00	1.44E-01	7.61E-02	1.83E-01
Am-241	0.00E+00	5.98E-01	7.36E-01	0.00E+00	8.45E-02

Table 2-22 Gamma Spectroscopy Results for Containment Building Exterior Concrete Cores

Sample ID	Activity (pCi/g)						
	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
1400X-1-CV-WCVX-006 0-0.5" Top	5.56E-02	4.48E-01	1.98E-01	3.95E-01	0.00E+00	0.00E+00	9.55E-01
1400X-1-CV-WCVX-006 0-0.5" Bottom	3.47E-01	3.63E-01	1.46E-01	0.00E+00	0.00E+00	0.00E+00	3.58E-02
1400X-1-CV-WCVX-009 0-0.5" Top	1.16E-01	9.08E-01	3.05E-02	1.55E-02	0.00E+00	0.00E+00	3.05E-01
1400X-1-CV-WCVX-009 0-0.5" Bottom	1.72E-01	7.79E-01	0.00E+00	2.12E-01	1.88E-01	0.00E+00	6.46E-01
1400X-1-CV-WCVX-020 0-0.5" Top	2.37E-01	7.83E-02	4.72E-02	5.76E-01	7.29E-01	3.06E+00	4.78E+00
1400X-1-CV-WCVX-020 0-0.5" Bottom	0.00E+00	0.00E+00	1.02E-01	8.11E-01	3.80E-01	4.82E-01	1.02E+01

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-23 Containment Building Exterior Concrete Cores Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	3.47E-01	1.55E-01	1.44E-01	1.26E-01
Cs-134	0.00E+00	9.08E-01	4.29E-01	4.06E-01	3.64E-01
Cs-137	0.00E+00	1.98E-01	8.73E-02	7.46E-02	7.53E-02
Eu-152	0.00E+00	8.11E-01	3.35E-01	3.04E-01	3.22E-01
Eu-154	0.00E+00	7.29E-01	2.16E-01	9.40E-02	2.94E-01
Eu-155	0.00E+00	3.06E+00	5.90E-01	0.00E+00	1.23E+00
Am-241	3.58E-02	1.02E+01	2.82E+00	8.01E-01	4.01E+00

Table 2-24 2100 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2100X-1-CJ-FCV1-001	0.0 - 0.5	**1.41E+00**	1.22E+00	**5.30E+02**	0.00E+00	0.00E+00	2.09E-01	8.48E-02
2100X-1-CJ-FCV2-001	0.5 - 1.0	9.78E-03	0.00E+00	**7.80E+00**	0.00E+00	0.00E+00	0.00E+00	1.61E-02
2100X-1-CJ-FCV3-001	1.0 - 1.5	0.00E+00	1.26E-01	**3.55E+00**	0.00E+00	0.00E+00	1.35E-01	0.00E+00
2100X-1-CJ-FCV4-001	1.5 - 2.0	1.02E-01	5.96E-02	**6.63E+00**	0.00E+00	0.00E+00	2.16E-01	4.91E-02
2100X-1-CJ-FCV5-001	2.0 - 4.0	0.00E+00	1.85E-01	**7.85E-01**	0.00E+00	4.29E-02	0.00E+00	0.00E+00
2100X-1-CJ-FCV6-001	4.0 - 6.0	5.41E-02	0.00E+00	**8.76E-01**	0.00E+00	0.00E+00	3.05E-01	1.54E-01
Location 2								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2100X-1-CJ-WCV1-002	0.0 - 0.5	3.09E-02	0.00E+00	**8.21E-01**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2100X-1-CJ-WCV2-002	0.5 - 1.0	3.05E-02	0.00E+00	5.33E-02	2.29E-03	0.00E+00	2.29E-01	0.00E+00
2100X-1-CJ-WCV3-002	1.0 - 1.5	3.91E-02	0.00E+00	1.98E-01	3.31E-03	0.00E+00	4.31E-03	0.00E+00
2100X-1-CJ-WCV4-002	1.5 - 2.0	5.06E-02	0.00E+00	**7.33E-01**	0.00E+00	0.00E+00	4.86E-02	0.00E+00
2100X-1-CJ-WCV5-002	2.0 - 4.0	1.90E-01	0.00E+00	1.89E-01	2.26E-01	3.18E-02	1.10E-01	0.00E+00
2100X-1-CJ-WCV6-002	4.0 - 6.0	7.99E-02	0.00E+00	3.11E-03	0.00E+00	0.00E+00	7.81E-03	0.00E+00
Location 3								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2100X-1-CJ-FCV1-003	0.0 - 0.5	**1.66E+00**	1.41E-01	**3.20E+02**	0.00E+00	0.00E+00	0.00E+00	3.89E-01
2100X-1-CJ-FCV2-003	0.5 - 1.0	**2.71E-01**	1.28E-01	**3.44E+01**	0.00E+00	1.89E-01	0.00E+00	0.00E+00
2100X-1-CJ-FCV3-003	1.0 - 1.5	1.35E-01	2.38E-01	**2.08E+01**	0.00E+00	0.00E+00	7.24E-02	0.00E+00
Location 4								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2100X-1-CJ-WCV1-004	0.0 - 0.5	4.35E-02	0.00E+00	**4.18E+00**	7.34E-02	0.00E+00	6.69E-02	0.00E+00
2100X-1-CJ-WCV2-004	0.5 - 1.0	1.49E-01	8.96E-02	**3.70E-01**	0.00E+00	2.89E-02	2.85E-01	0.00E+00
2100X-1-CJ-WCV3-004	1.0 - 1.5	1.53E-01	0.00E+00	**2.58E-01**	3.71E-02	1.61E-02	1.45E-01	0.00E+00
Location 5								
Sample ID	**Depth**	**Activity (pCi/g)**						
	(inches)	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2100X-1-CJ-FCV1-005	0.0 - 0.5	**8.21E-01**	**1.43E+00**	**7.93E+02**	7.41E-01	5.70E-01	0.00E+00	1.28E-01
2100X-1-CJ-FCV2-005	0.5 - 1.0	5.45E-02	4.50E-02	**2.55E+01**	2.53E-01	0.00E+00	0.00E+00	0.00E+00
2100X-1-CJ-FCV3-005	1.0 - 1.5	7.95E-02	2.43E-01	**5.93E+01**	0.00E+00	0.00E+00	8.99E-01	2.65E-01

Location 6								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2100X-1-CJ-FCV1-006	0.0 - 0.5	**2.09E+00**	8.23E-02	**1.33E+02**	0.00E+00	0.00E+00	7.11E-02	0.00E+00
2100X-1-CJ-FCV2-006	0.5 - 1.0	3.90E-02	0.00E+00	**1.93E+00**	0.00E+00	0.00E+00	4.41E-02	0.00E+00
2100X-1-CJ-FCV3-006	1.0 - 1.5	0.00E+00	0.00E+00	**2.26E+00**	0.00E+00	1.98E-04	1.86E-01	0.00E+00
2100X-1-CJ-FCV4-006	1.5 - 2.0	1.50E-01	0.00E+00	**9.00E-01**	4.28E-02	0.00E+00	1.91E-01	0.00E+00
2100X-1-CJ-FCV5-006	2.0 - 4.0	2.84E-01	0.00E+00	**5.64E-01**	0.00E+00	1.14E-01	1.16E-01	0.00E+00
2100X-1-CJ-FCV6-006	4.0 - 6.0	1.65E-01	0.00E+00	**1.23E+00**	0.00E+00	6.19E-02	0.00E+00	0.00E+00
Location 7								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2100X-1-CJ-FCV1-007	0.0 - 0.5	**5.81E-01**	0.00E+00	**5.81E+01**	7.95E-01	3.00E-01	4.28E-03	1.10E-01
2100X-1-CJ-FCV2-007	0.5 - 1.0	9.88E-02	0.00E+00	**1.04E+00**	2.94E-01	0.00E+00	1.20E-01	0.00E+00
2100X-1-CJ-FCV3-007	1.0 - 1.5	8.84E-02	6.26E-02	**3.05E-01**	0.00E+00	0.00E+00	7.34E-02	0.00E+00
2100X-1-CJ-FCV4-007	1.5 - 2.0	1.53E-01	0.00E+00	**2.88E+00**	0.00E+00	8.60E-02	0.00E+00	0.00E+00
2100X-1-CJ-FCV5-007	2.0 - 4.0	1.11E-01	0.00E+00	3.49E-01	2.51E-01	6.70E-02	4.05E-01	7.18E-02
2100X-1-CJ-FCV6-007	4.0 - 6.0	0.00E+00	0.00E+00	4.16E-01	0.00E+00	0.00E+00	1.39E-02	5.38E-02
Location 8								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2100X-1-CJ-WCV1-008	0.0 - 0.5	0.00E+00	0.00E+00	1.80E-01	0.00E+00	9.70E-03	2.58E-01	8.76E-02
2100X-1-CJ-WCV2-008	0.5 - 1.0	9.99E-02	3.45E-02	1.61E-01	0.00E+00	1.75E-02	8.41E-02	0.00E+00
2100X-1-CJ-WCV3-008	1.0 - 1.5	7.11E-02	1.70E-01	5.13E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Location 9								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2100X-1-CJ-FCV1-009	0.0 - 0.5	2.73E-01	2.99E-02	**3.31E+01**	1.84E-01	1.17E-01	0.00E+00	0.00E+00
2100X-1-CJ-FCV2-009	0.5 - 1.0	8.78E-02	0.00E+00	1.01E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2100X-1-CJ-FCV3-009	1.0 - 1.5	1.19E-01	6.94E-02	4.48E-02	5.94E-02	9.88E-02	0.00E+00	0.00E+00
Location 10								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2100X-1-CJ-FCV1-010	0.0 - 0.5	**1.73E+00**	1.14E-01	**8.26E+01**	0.00E+00	0.00E+00	0.00E+00	3.55E-01
2100X-1-CJ-FCV2-010	0.5 - 1.0	1.93E-02	0.00E+00	**1.55E+01**	0.00E+00	0.00E+00	2.01E-01	1.08E-01
2100X-1-CJ-FCV3-010	1.0 - 1.5	1.50E-01	0.00E+00	**3.43E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2100X-1-CJ-FCV4-010	1.5 - 2.0	8.91E-02	0.00E+00	**2.43E+00**	1.60E-01	0.00E+00	1.90E-01	2.88E-02
2100X-1-CJ-FCV5-010	2.0 - 4.0	2.43E-02	0.00E+00	**1.69E+00**	1.95E-01	1.10E-01	3.09E-01	1.76E-01
2100X-1-CJ-FCV6-010	4.0 - 6.0	1.93E-02	0.00E+00	**3.96E-01**	0.00E+00	0.00E+00	2.13E-01	0.00E+00

Location 11								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2100X-1-CJ-FCV1-011	0.0 - 0.5	**3.06E+00**	2.78E-01	**5.73E+01**	0.00E+00	6.69E-02	9.48E-02	7.49E-02
2100X-1-CJ-FCV2-011	0.5 - 1.0	1.34E-01	1.44E-01	**6.41E+00**	1.54E+00	2.15E-01	8.34E-02	0.00E+00
2100X-1-CJ-FCV3-011	1.0 - 1.5	7.14E-02	0.00E+00	**1.20E+00**	0.00E+00	0.00E+00	1.93E-01	0.00E+00
Location 12								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2100X-1-CJ-WCV1-012	0.0 - 0.5	9.44E-02	0.00E+00	**9.86E-01**	8.01E-02	4.35E-02	3.24E-01	0.00E+00
2100X-1-CJ-WCV2-012	0.5 - 1.0	7.05E-01	9.78E-02	0.00E+00	0.00E+00	7.80E-02	2.79E-02	0.00E+00
2100X-1-CJ-WCV3-012	1.0 - 1.5	1.85E-01	2.71E-01	3.03E-02	0.00E+00	3.88E-02	7.44E-02	0.00E+00
2100X-1-CJ-WCV4-012	1.5 - 2.0	1.71E-02	5.05E-02	3.23E-01	0.00E+00	0.00E+00	1.63E-01	8.40E-02
2100X-1-CJ-WCV5-012	2.0 - 4.0	2.74E-02	0.00E+00	0.00E+00	8.64E-03	1.39E-01	0.00E+00	0.00E+00
2100X-1-CJ-WCV6-012	4.0 - 6.0	4.61E-02	0.00E+00	8.45E-02	0.00E+00	0.00E+00	3.66E-01	0.00E+00
Location 13								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2100X-1-CJ-WCV1-013	0.0 - 0.5	9.96E-02	9.83E-02	**1.53E+00**	1.50E-01	2.25E-01	3.06E-01	0.00E+00
2100X-1-CJ-WCV2-013	0.5 - 1.0	7.13E-02	0.00E+00	3.91E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2100X-1-CJ-WCV3-013	1.0 - 1.5	4.69E-02	0.00E+00	1.84E-01	1.15E-01	0.00E+00	0.00E+00	0.00E+00
2100X-1-CJ-WCV4-013	1.5 - 2.0	0.00E+00	2.96E-01	1.78E-01	0.00E+00	0.00E+00	3.23E-01	0.00E+00
2100X-1-CJ-WCV5-013	2.0 - 4.0	1.16E-01	0.00E+00	1.99E-01	0.00E+00	0.00E+00	7.09E-02	0.00E+00
2100X-1-CJ-WCV6-013	4.0 - 6.0	6.04E-02	0.00E+00	1.06E-01	1.86E-02	1.66E-01	0.00E+00	1.17E-01
Location 14								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2100X-1-CJ-FCV1-014	0.0 - 0.5	**7.81E-01**	3.44E-02	**1.30E+02**	1.25E-01	1.13E-02	2.66E-02	1.83E-01
2100X-1-CJ-FCV2-014	0.5 - 1.0	8.46E-02	0.00E+00	**2.05E+01**	6.28E-02	0.00E+00	1.96E-01	1.39E-01
2100X-1-CJ-FCV3-014	1.0 - 1.5	0.00E+00	0.00E+00	**6.79E+00**	2.15E-01	1.79E-01	3.28E-01	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-25 2100 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	2.09E+00	2.76E-01	8.84E-02	5.60E-01
Cs-134	0.00E+00	1.43E+00	9.09E-02	0.00E+00	2.39E-01
Cs-137	0.00E+00	7.93E+02	3.77E+01	9.86E-01	1.26E+02
Eu-152	0.00E+00	1.54E+00	8.94E-02	0.00E+00	2.38E-01
Eu-154	0.00E+00	5.70E-01	4.80E-02	0.00E+00	9.54E-02
Eu-155	0.00E+00	8.99E-01	1.24E-01	7.34E-02	1.54E-01
Am-241	0.00E+00	3.89E-01	8.94E-02	0.00E+00	8.37E-02

Table 2-26 2200 Gamma Spectroscopy Results for Concrete Samples

Sample ID	Depth (inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
Location 1								
		Activity (pCi/g)						
2200X-1-CJ-FCV1-001	0.0 - 0.5	**3.66E-01**	7.39E-02	**2.23E+01**	1.86E-01	2.24E-01	1.22E-01	1.40E-02
2200X-1-CJ-FCV2-001	0.5 - 1.0	1.78E-01	0.00E+00	**8.99E-01**	0.00E+00	0.00E+00	3.24E-01	0.00E+00
2200X-1-CJ-FCV3-001	1.0 - 1.5	1.18E-01	0.00E+00	**3.06E-01**	8.23E-02	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-FCV4-001	1.5 - 2.0	3.54E-02	0.00E+00	**9.19E-01**	0.00E+00	0.00E+00	2.14E-01	0.00E+00
2200X-1-CJ-FCV5-001	2.0 - 4.0	9.08E-02	3.99E-02	6.30E-02	6.99E-02	0.00E+00	4.06E-02	0.00E+00
2200X-1-CJ-FCV6-001	4.0 - 6.0	1.16E-01	0.00E+00	5.54E-02	0.00E+00	6.89E-02	2.23E-01	9.04E-03
Location 2								
Sample ID	Depth (inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2200X-1-CJ-FCV1-002	0.0 - 0.5	1.34E-01	0.00E+00	**4.40E+01**	2.89E-02	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-FCV2-002	0.5 - 1.0	1.01E-01	0.00E+00	**2.20E+00**	0.00E+00	0.00E+00	0.00E+00	3.90E-02
2200X-1-CJ-FCV3-002	1.0 - 1.5	7.66E-02	0.00E+00	**8.05E-01**	0.00E+00	0.00E+00	2.24E-01	0.00E+00
2200X-1-CJ-FCV4-002	1.5 - 2.0	1.64E-01	3.21E-02	**4.50E+00**	1.76E-02	0.00E+00	6.04E-02	3.44E-02
2200X-1-CJ-FCV5-002	2.0 - 4.0	5.95E-02	2.84E-02	2.84E-01	0.00E+00	1.68E-01	2.04E-02	0.00E+00
2200X-1-CJ-FCV6-002	4.0 - 6.0	1.45E-01	0.00E+00	2.66E-01	0.00E+00	0.00E+00	3.98E-02	0.00E+00
Location 3								
Sample ID	Depth (inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2200X-1-CJ-FCV1-003	0.0 - 0.5	2.59E-01	1.86E-01	**2.35E+01**	0.00E+00	8.79E-03	0.00E+00	0.00E+00
2200X-1-CJ-FCV2-003	0.5 - 1.0	2.03E-01	0.00E+00	**1.12E+00**	1.41E-01	9.56E-02	1.79E-02	6.29E-02
2200X-1-CJ-FCV3-003	1.0 - 1.5	2.74E-01	1.17E-01	**1.23E+00**	0.00E+00	0.00E+00	0.00E+00	6.79E-02
2200X-1-CJ-FCV4-003	1.5 - 2.0	2.70E-02	9.23E-02	**5.39E-01**	6.94E-02	0.00E+00	1.26E-01	1.38E-03
2200X-1-CJ-FCV5-003	2.0 - 4.0	7.21E-02	0.00E+00	2.66E-01	0.00E+00	0.00E+00	3.16E-01	7.11E-02
2200X-1-CJ-FCV6-003	4.0 - 6.0	0.00E+00	0.00E+00	1.33E-01	0.00E+00	0.00E+00	3.08E-01	0.00E+00
Location 4								
Sample ID	Depth (inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2200X-1-CJ-FCV1-004	0.0 - 0.5	**1.83E+00**	**6.26E-01**	**2.68E+02**	0.00E+00	0.00E+00	1.66E-01	0.00E+00
2200X-1-CJ-FCV2-004	0.5 - 1.0	8.04E-02	0.00E+00	**3.08E+00**	0.00E+00	6.18E-02	1.61E-01	2.89E-01
2200X-1-CJ-FCV3-004	1.0 - 1.5	1.33E-01	1.66E-01	**2.21E+00**	1.56E-01	1.34E-01	3.30E-01	0.00E+00
2200X-1-CJ-FCV4-004	1.5 - 2.0	1.45E-01	0.00E+00	**1.63E+00**	1.18E-01	1.74E-02	1.51E-01	0.00E+00
2200X-1-CJ-FCV5-004	2.0 - 4.0	4.71E-02	2.74E-02	1.94E-01	1.38E-02	1.48E-01	0.00E+00	0.00E+00
2200X-1-CJ-FCV6-004	4.0 - 6.0	1.89E-01	1.21E-01	**3.14E-01**	8.40E-02	0.00E+00	2.18E-01	0.00E+00

Location 5								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-005	0.0 - 0.5	3.95E-01	1.75E-01	**5.11E+00**	0.00E+00	0.00E+00	1.09E-02	0.00E+00
2200X-1-CJ-WCV2-005	0.5 - 1.0	1.36E-01	0.00E+00	**3.96E-01**	1.09E-01	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-WCV3-005	1.0 - 1.5	0.00E+00	0.00E+00	**3.56E-01**	0.00E+00	0.00E+00	4.18E-01	0.00E+00
Location 6								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-006	0.0 - 0.5	**7.76E-01**	1.71E-01	**7.54E+02**	0.00E+00	6.36E-01	0.00E+00	0.00E+00
2200X-1-CJ-FCV2-006	0.5 - 1.0	0.00E+00	0.00E+00	**3.21E+01**	2.93E-01	7.35E-02	6.84E-01	0.00E+00
2200X-1-CJ-FCV3-006	1.0 - 1.5	5.69E-02	0.00E+00	**1.24E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Location 7								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-007	0.0 - 0.5	1.18E-01	0.00E+00	**9.30E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-WCV2-007	0.5 - 1.0	1.18E-01	2.48E-01	**2.25E+00**	1.35E-01	6.55E-02	1.19E-01	0.00E+00
2200X-1-CJ-WCV3-007	1.0 - 1.5	4.25E-02	0.00E+00	**1.80E+00**	1.75E-01	0.00E+00	0.00E+00	0.00E+00
Location 8								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-008	0.0 - 0.5	**4.94E+01**	**1.21E+00**	**1.33E+03**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-FCV2-008	0.5 - 1.0	**9.20E+00**	4.35E-01	**2.06E+02**	5.84E-01	4.38E-01	0.00E+00	0.00E+00
2200X-1-CJ-FCV3-008	1.0 - 1.5	**4.04E+00**	9.53E-01	**2.68E+02**	0.00E+00	0.00E+00	0.00E+00	1.12E+00
Location 9								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-009	0.0 - 0.5	**3.81E-01**	4.79E-01	**5.09E+01**	9.51E-01	0.00E+00	3.75E-01	0.00E+00
2200X-1-CJ-WCV2-009	0.5 - 1.0	2.53E-01	2.83E-01	**2.81E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-WCV3-009	1.0 - 1.5	6.94E-02	0.00E+00	**6.28E+00**	2.86E-02	1.64E-03	5.06E-03	0.00E+00
Location 10								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-010	0.0 - 0.5	**9.13E+00**	6.39E-01	**4.51E+02**	0.00E+00	0.00E+00	1.99E-01	4.53E-01
2200X-1-CJ-FCV2-010	0.5 - 1.0	2.46E-01	3.59E-02	**2.15E+01**	0.00E+00	4.55E-02	2.49E-01	4.88E-02
2200X-1-CJ-FCV3-010	1.0 - 1.5	4.10E-01	1.26E-02	**1.08E+01**	8.51E-02	1.98E-01	0.00E+00	0.00E+00
2200X-1-CJ-FCV4-010	1.5 - 2.0	**1.60E+00**	0.00E+00	**1.81E+01**	2.73E-01	0.00E+00	5.26E-01	2.75E-01
2200X-1-CJ-FCV5-010	2.0 - 4.0	2.03E-02	7.71E-02	**2.65E+00**	1.79E-01	1.20E-02	0.00E+00	0.00E+00
2200X-1-CJ-FCV6-010	4.0 - 6.0	0.00E+00	8.41E-02	**2.45E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Location 11								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-011	0.0 - 0.5	**4.85E-01**	0.00E+00	**7.23E+01**	2.53E-01	1.04E-01	1.16E+00	0.00E+00
2200X-1-CJ-FCV2-011	0.5 - 1.0	2.58E-01	0.00E+00	**2.75E+00**	2.48E-01	8.21E-02	0.00E+00	0.00E+00
2200X-1-CJ-FCV3-011	1.0 - 1.5	1.22E-01	0.00E+00	**5.41E+00**	1.41E-01	7.65E-03	2.31E-01	0.00E+00
Location 19								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-019	0.0 - 0.5	1.17E-01	8.18E-03	**4.83E+00**	0.00E+00	1.64E-02	0.00E+00	0.00E+00
2200X-1-CJ-WCV2-019	0.5 - 1.0	0.00E+00	0.00E+00	**8.09E-01**	1.07E-01	1.84E-02	0.00E+00	0.00E+00
2200X-1-CJ-WCV3-019	1.0 - 1.5	8.14E-02	1.13E-01	1.07E-01	0.00E+00	8.75E-02	0.00E+00	0.00E+00
2200X-1-CJ-WCV4-019	1.5 - 2.0	1.28E-01	0.00E+00	1.74E-01	0.00E+00	4.66E-02	0.00E+00	0.00E+00
2200X-1-CJ-WCV5-019	2.0 - 4.0	1.18E-01	0.00E+00	2.43E-01	0.00E+00	3.56E-03	0.00E+00	3.01E-02
2200X-1-CJ-WCV6-019	4.0 - 6.0	8.50E-02	0.00E+00	**1.08E-01**	0.00E+00	0.00E+00	1.06E-01	0.00E+00
Location 20								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-020	0.0 - 0.5	**2.06E+01**	**2.60E+01**	**1.25E+04**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-FCV2-020	0.5 - 1.0	**2.30E+00**	**1.58E+00**	**1.16E+03**	7.84E-01	2.81E-01	0.00E+00	0.00E+00
2200X-1-CJ-FCV3-020	1.0 - 1.5	**3.68E-01**	1.81E-01	**1.43E+02**	0.00E+00	0.00E+00	0.00E+00	1.17E-02
Location 21								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-021	0.0 - 0.5	**5.18E-01**	7.65E-01	**4.73E+02**	0.00E+00	0.00E+00	0.00E+00	7.38E-01
2200X-1-CJ-FCV2-021	0.5 - 1.0	1.17E-01	2.94E-01	**1.54E+01**	1.15E-01	1.74E-01	0.00E+00	0.00E+00
2200X-1-CJ-FCV3-021	1.0 - 1.5	5.16E-02	2.88E-01	**4.29E+00**	0.00E+00	6.04E-02	0.00E+00	0.00E+00
Location 22								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-022	0.0 - 0.5	**1.04E+00**	1.09E+00	**1.81E+04**	0.00E+00	0.00E+00	3.09E+00	0.00E+00
2200X-1-CJ-FCV2-022	0.5 - 1.0	4.49E-01	2.09E-01	**2.63E+03**	3.50E+00	1.04E+00	1.20E+00	0.00E+00
2200X-1-CJ-FCV3-022	1.0 - 1.5	**1.24E-01**	2.84E-01	**1.18E+03**	0.00E+00	0.00E+00	7.08E-01	0.00E+00
Location 23								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-023	0.0 - 0.5	1.33E-01	8.85E-02	**5.28E+01**	0.00E+00	7.95E-02	6.74E-01	0.00E+00
2200X-1-CJ-WCV2-023	0.5 - 1.0	2.54E-02	0.00E+00	**1.64E+01**	0.00E+00	2.11E-01	3.01E-01	0.00E+00
2200X-1-CJ-WCV3-023	1.0 - 1.5	3.19E-02	0.00E+00	**2.45E+00**	0.00E+00	8.85E-02	1.90E-01	0.00E+00

Location 24								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-024	0.0 - 0.5	4.25E-01	0.00E+00	**1.33E+00**	1.09E-01	0.00E+00	0.00E+00	2.10E-01
2200X-1-CJ-WCV2-024	0.5 - 1.0	1.11E-01	0.00E+00	1.55E-01	2.45E-01	6.50E-02	0.00E+00	1.29E-01
2200X-1-CJ-WCV3-024	1.0 - 1.5	1.51E-01	0.00E+00	**1.19E-01**	0.00E+00	0.00E+00	1.01E-01	0.00E+00
2200X-1-CJ-WCV4-024	1.5 - 2.0	2.28E-01	0.00E+00	2.28E-01	1.08E-01	2.04E-01	0.00E+00	6.59E-02
2200X-1-CJ-WCV5-024	2.0 - 4.0	1.01E-01	0.00E+00	2.48E-01	8.50E-02	0.00E+00	0.00E+00	1.51E-01
2200X-1-CJ-WCV6-024	4.0 - 6.0	4.73E-01	0.00E+00	**0.00E+00**	1.78E-01	3.09E-02	1.16E-01	0.00E+00

Location 25								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-025	0.0 - 0.5	**8.61E-01**	1.81E-01	**4.09E+01**	0.00E+00	4.41E-01	7.81E-01	0.00E+00
2200X-1-CJ-FCV2-025	0.5 - 1.0	9.51E-02	0.00E+00	**1.69E-01**	0.00E+00	0.00E+00	7.45E-02	0.00E+00
2200X-1-CJ-FCV3-025	1.0 - 1.5	2.75E-01	0.00E+00	**6.58E-01**	0.00E+00	0.00E+00	1.27E-01	0.00E+00

Location 26								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-026	0.0 - 0.5	1.76E-01	0.00E+00	**2.18E+04**	0.00E+00	0.00E+00	5.50E+00	0.00E+00
2200X-1-CJ-FCV2-026	0.5 - 1.0	1.42E-01	1.73E-01	**3.85E+02**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-FCV3-026	1.0 - 1.5	1.21E-01	0.00E+00	**1.61E+02**	5.64E-01	0.00E+00	1.24E-01	0.00E+00

Location 27								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-027	0.0 - 0.5	8.80E-02	0.00E+00	**6.18E+00**	0.00E+00	2.99E-02	0.00E+00	0.00E+00
2200X-1-CJ-WCV2-027	0.5 - 1.0	1.04E-01	0.00E+00	**4.15E-01**	0.00E+00	1.13E-01	0.00E+00	0.00E+00
2200X-1-CJ-WCV3-027	1.0 - 1.5	1.94E-01	0.00E+00	**4.82E-01**	0.00E+00	0.00E+00	2.74E-01	0.00E+00

Location 28								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-028	0.0 - 0.5	2.59E-01	5.50E-01	**2.81E+00**	6.28E-02	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-FCV2-028	0.5 - 1.0	7.83E-02	0.00E+00	5.01E-01	2.65E-01	6.84E-02	4.38E-01	0.00E+00
2200X-1-CJ-FCV3-028	1.0 - 1.5	2.18E-01	0.00E+00	**2.96E-01**	2.59E-03	0.00E+00	1.51E-01	0.00E+00

Location 29								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-029	0.0 - 0.5	1.76E-01	9.40E-04	**3.00E+02**	1.28E-01	9.73E-02	0.00E+00	0.00E+00
2200X-1-CJ-FCV2-029	0.5 - 1.0	5.18E-02	4.86E-02	**3.64E+01**	0.00E+00	0.00E+00	1.65E-01	1.55E-01
2200X-1-CJ-FCV3-029	1.0 - 1.5	8.12E-02	0.00E+00	**6.76E+01**	1.14E-01	0.00E+00	4.15E-02	8.24E-02

Location 30								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-030	0.0 - 0.5	**1.41E+02**	0.00E+00	**3.10E+02**	0.00E+00	0.00E+00	1.02E+00	7.74E-01
2200X-1-CJ-FCV2-030	0.5 - 1.0	**4.76E+00**	2.44E-02	**6.94E+00**	0.00E+00	0.00E+00	1.39E-01	1.26E-01
2200X-1-CJ-FCV3-030	1.0 - 1.5	**2.14E+00**	4.79E-02	**3.05E+00**	0.00E+00	0.00E+00	0.00E+00	6.44E-02
2200X-1-CJ-FCV4-030	1.5 - 2.0	**3.39E+00**	0.00E+00	**5.90E+00**	0.00E+00	0.00E+00	1.71E-01	0.00E+00
2200X-1-CJ-FCV5-030	2.0 - 4.0	2.04E-01	0.00E+00	**2.18E-01**	0.00E+00	7.95E-02	2.41E-01	0.00E+00
2200X-1-CJ-FCV6-030	4.0 - 6.0	1.56E-01	0.00E+00	5.08E-01	0.00E+00	2.41E-01	0.00E+00	1.16E-01
Location 31								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-031	0.0 - 0.5	**8.00E+00**	4.86E-01	**4.14E+02**	0.00E+00	0.00E+00	1.86E+00	0.00E+00
2200X-1-CJ-FCV2-031	0.5 - 1.0	**1.54E-01**	2.21E-01	**5.73E+01**	0.00E+00	0.00E+00	0.00E+00	7.98E-02
2200X-1-CJ-FCV3-031	1.0 - 1.5	**5.76E-01**	0.00E+00	**3.55E+02**	3.30E-01	2.76E-01	6.27E-01	0.00E+00
Location 32								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-032	0.0 - 0.5	5.60E-02	0.00E+00	**3.36E+00**	1.63E-01	0.00E+00	3.21E-01	1.18E-02
2200X-1-CJ-WCV2-032	0.5 - 1.0	1.68E-01	0.00E+00	**2.54E-01**	1.79E-01	2.55E-01	3.48E-01	0.00E+00
2200X-1-CJ-WCV3-032	1.0 - 1.5	3.70E-02	0.00E+00	**9.09E-02**	5.78E-02	9.28E-02	2.31E-02	0.00E+00
2200X-1-CJ-WCV4-032	1.5 - 2.0	1.38E-01	0.00E+00	1.76E-01	3.71E-02	0.00E+00	3.51E-02	5.78E-02
2200X-1-CJ-WCV5-032	2.0 - 4.0	3.06E-01	0.00E+00	**1.85E-01**	0.00E+00	3.03E-03	0.00E+00	0.00E+00
2200X-1-CJ-WCV6-032	4.0 - 6.0	8.12E-02	0.00E+00	**0.00E+00**	6.48E-02	0.00E+00	0.00E+00	0.00E+00
Location 33								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-033	0.0 - 0.5	0.00E+00	2.70E-02	2.63E-01	1.50E-01	3.15E-01	0.00E+00	1.01E-01
2200X-1-CJ-WCV2-033	0.5 - 1.0	1.45E-01	0.00E+00	5.83E-02	1.07E-01	1.88E-02	1.20E-01	0.00E+00
2200X-1-CJ-WCV3-033	1.0 - 1.5	3.82E-01	0.00E+00	**1.10E-01**	6.36E-03	1.24E-01	1.66E-01	0.00E+00
2200X-1-CJ-WCV4-033	1.5 - 2.0	1.44E-01	0.00E+00	8.04E-02	0.00E+00	1.16E-02	0.00E+00	2.34E-02
2200X-1-CJ-WCV5-033	2.0 - 4.0	7.53E-02	0.00E+00	6.68E-02	2.05E-01	1.13E-01	8.68E-02	0.00E+00
2200X-1-CJ-WCV6-033	4.0 - 6.0	1.48E-01	0.00E+00	**3.42E-01**	1.07E-01	4.21E-01	0.00E+00	0.00E+00

Location 34								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-034	0.0 - 0.5	**1.53E-01**	0.00E+00	**8.21E-01**	2.03E-01	0.00E+00	8.03E-02	0.00E+00
2200X-1-CJ-WCV2-034	0.5 - 1.0	1.20E-02	0.00E+00	**2.29E-01**	0.00E+00	2.81E-02	1.61E-01	0.00E+00
2200X-1-CJ-WCV3-034	1.0 - 1.5	0.00E+00	0.00E+00	**4.97E-02**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-WCV4-034	1.5 - 2.0	2.36E-01	0.00E+00	**5.33E-03**	1.51E-01	0.00E+00	3.55E-02	0.00E+00
2200X-1-CJ-WCV5-034	2.0 - 4.0	2.90E-01	0.00E+00	**1.18E-01**	2.18E-02	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-WCV6-034	4.0 - 6.0	1.89E-01	0.00E+00	**0.00E+00**	3.36E-01	0.00E+00	2.50E-01	0.00E+00
Location 35								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-035	0.0 - 0.5	**1.85E+01**	7.54E-02	**8.51E+02**	3.15E-01	0.00E+00	0.00E+00	5.39E-01
2200X-1-CJ-FCV2-035	0.5 - 1.0	**3.03E+00**	2.25E-01	**1.87E+02**	6.15E-01	8.48E-01	3.97E-01	0.00E+00
2200X-1-CJ-FCV3-035	1.0 - 1.5	**1.38E+00**	0.00E+00	**4.03E+01**	0.00E+00	1.62E-01	6.09E-02	0.00E+00
Location 36								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-036	0.0 - 0.5	**4.20E+00**	4.16E-01	**1.08E+02**	0.00E+00	0.00E+00	0.00E+00	6.43E-02
2200X-1-CJ-FCV2-036	0.5 - 1.0	**2.35E-01**	0.00E+00	**9.50E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-FCV3-036	1.0 - 1.5	**1.54E-01**	6.55E-03	**4.39E+00**	1.22E-01	1.64E-01	1.04E-01	0.00E+00
Location 37								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-WCV1-037	0.0 - 0.5	6.84E-02	0.00E+00	**4.29E+00**	0.00E+00	1.83E-02	0.00E+00	0.00E+00
2200X-1-CJ-WCV2-037	0.5 - 1.0	1.98E-01	0.00E+00	**3.31E-01**	0.00E+00	0.00E+00	0.00E+00	1.85E-01
2200X-1-CJ-WCV3-037	1.0 - 1.5	1.06E-01	0.00E+00	**2.29E-01**	0.00E+00	0.00E+00	0.00E+00	2.13E-01
2200X-1-CJ-WCV4-037	1.5 - 2.0	3.71E-02	0.00E+00	1.93E-01	0.00E+00	0.00E+00	2.13E-01	1.66E-01
2200X-1-CJ-WCV5-037	2.0 - 4.0	0.00E+00	0.00E+00	5.24E-02	0.00E+00	8.29E-02	4.74E-02	0.00E+00
2200X-1-CJ-WCV6-037	4.0 - 6.0	2.91E-02	9.63E-02	**3.70E-01**	0.00E+00	1.29E-01	4.00E-02	4.50E-02
Location 38								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2200X-1-CJ-FCV1-038	0.0 - 0.5	**4.73E+00**	0.00E+00	**1.45E+02**	0.00E+00	1.93E-01	0.00E+00	0.00E+00
2200X-1-CJ-FCV2-038	0.5 - 1.0	**3.01E-01**	0.00E+00	**1.49E+01**	3.90E-01	0.00E+00	4.81E-01	0.00E+00
2200X-1-CJ-FCV3-038	1.0 - 1.5	6.23E-02	0.00E+00	**3.46E+00**	1.25E-01	9.63E-02	2.69E-01	0.00E+00

Location 39								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2200X-1-CJ-FCV1-039	0.0 - 0.5	**4.85E+00**	0.00E+00	**1.49E+02**	5.43E-02	2.00E-02	2.94E-01	0.00E+00
2200X-1-CJ-FCV2-039	0.5 - 1.0	8.39E-02	0.00E+00	**2.13E+00**	0.00E+00	0.00E+00	2.01E-01	0.00E+00
2200X-1-CJ-FCV3-039	1.0 - 1.5	5.33E-02	5.75E-02	**1.20E+00**	3.85E-02	0.00E+00	1.36E-01	0.00E+00
2200X-1-CJ-FCV4-039	1.5 - 2.0	2.03E-01	1.56E-02	**3.64E+00**	2.39E-02	1.07E-02	7.85E-02	5.84E-03
2200X-1-CJ-FCV5-039	2.0 - 4.0	1.18E-01	0.00E+00	**1.69E-01**	9.56E-02	3.29E-03	0.00E+00	0.00E+00
2200X-1-CJ-FCV6-039	4.0 - 6.0	1.44E-01	0.00E+00	6.35E-02	8.84E-02	3.30E-02	7.81E-02	0.00E+00
Location 40								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2200X-1-CJ-FCV1-040	0.0 - 0.5	5.73E-02	0.00E+00	**1.90E+01**	5.40E-02	0.00E+00	2.80E-01	0.00E+00
2200X-1-CJ-FCV2-040	0.5 - 1.0	5.06E-02	0.00E+00	**1.90E-01**	0.00E+00	0.00E+00	1.65E-01	0.00E+00
2200X-1-CJ-FCV3-040	1.0 - 1.5	0.00E+00	3.38E-01	5.86E-02	2.09E-01	8.04E-02	4.69E-02	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-27 2200 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	1.83E+00	2.28E+00	1.44E-01	1.29E+01
Cs-134	0.00E+00	2.60E+01	2.91E-01	0.00E+00	2.22E+00
Cs-137	0.00E+00	2.18E+04	4.74E+02	2.23E+00	2.62E+03
Eu-152	0.00E+00	3.50E+00	1.09E-01	0.00E+00	3.28E-01
Eu-154	0.00E+00	1.04E+00	6.93E-02	0.00E+00	1.49E-01
Eu-155	0.00E+00	5.50E+00	2.16E-01	4.71E-02	5.82E-01
Am-241	0.00E+00	1.12E+00	1.09E-01	0.00E+00	1.49E-01

Table 2-28 2300 Gamma Spectroscopy Results for Concrete Samples

Sample ID	Depth (inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
Location 1								
		Activity (pCi/g)						
2300X-1-CJ-FCV1-001	0.0 - 0.5	**9.04E-01**	0.00E+00	**7.51E+03**	3.63E+00	0.00E+00	5.04E+00	2.64E-02
2300X-1-CJ-FCV2-001	0.5 - 1.0	2.16E-01	0.00E+00	**1.18E+02**	1.99E-01	1.53E-01	2.50E-01	0.00E+00
2300X-1-CJ-FCV3-001	1.0 - 1.5	7.70E-02	0.00E+00	**2.59E+01**	0.00E+00	0.00E+00	1.49E-01	0.00E+00
Location 2								
Sample ID	**Depth (inches)**	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2300X-1-CJ-FCV1-002	0.0 - 0.5	**6.99E+00**	3.54E+00	**9.83E+03**	0.00E+00	2.60E+00	2.94E+00	7.86E-01
2300X-1-CJ-FCV2-002	0.5 - 1.0	8.21E-02	0.00E+00	**2.11E+02**	0.00E+00	0.00E+00	2.46E-01	2.15E-01
2300X-1-CJ-FCV3-002	1.0 - 1.5	1.31E-01	1.18E-01	**2.96E+02**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Location 3								
Sample ID	**Depth (inches)**	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2300X-1-CJ-FCV1-003	0.0 - 0.5	**1.40E+02**	1.44E+00	**6.61E+03**	2.00E+00	2.13E+00	9.13E-02	7.56E-01
2300X-1-CJ-FCV2-003	0.5 - 1.0	**1.01E+01**	0.00E+00	**3.26E+02**	2.03E-02	0.00E+00	7.00E-02	5.68E-01
2300X-1-CJ-FCV3-003	1.0 - 1.5	**2.96E+00**	0.00E+00	**1.11E+02**	2.83E-01	0.00E+00	2.36E-01	4.00E-01
Location 4								
Sample ID	**Depth (inches)**	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2300X-1-CJ-FCV1-004	0.0 - 0.5	2.66E-01	4.71E-01	**2.75E+02**	2.59E-01	1.80E-01	1.39E+00	5.46E-02
2300X-1-CJ-FCV2-004	0.5 - 1.0	7.70E-02	1.86E-01	**3.59E+01**	5.95E-02	9.21E-02	0.00E+00	4.28E-01
2300X-1-CJ-FCV3-004	1.0 - 1.5	1.45E-01	7.54E-02	**3.44E+01**	0.00E+00	3.56E-01	2.36E-01	0.00E+00
Location 5								
Sample ID	**Depth (inches)**	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2300X-1-CJ-FCV1-005	0.0 - 0.5	**2.36E+00**	5.29E-01	**3.90E+03**	0.00E+00	0.00E+00	7.10E-01	0.00E+00
2300X-1-CJ-FCV2-005	0.5 - 1.0	**3.78E-01**	2.43E-01	**6.81E+02**	0.00E+00	7.14E-01	0.00E+00	0.00E+00
2300X-1-CJ-FCV3-005	1.0 - 1.5	1.58E-01	0.00E+00	**3.94E+02**	2.98E-01	3.76E-01	3.05E-01	2.51E-01
Location 6								
Sample ID	**Depth (inches)**	**Co-60**	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
2300X-1-CJ-FCV1-006	0.0 - 0.5	7.43E-02	0.00E+00	**3.80E+01**	0.00E+00	0.00E+00	3.83E-01	0.00E+00
2300X-1-CJ-FCV2-006	0.5 - 1.0	0.00E+00	0.00E+00	**6.84E+01**	0.00E+00	0.00E+00	0.00E+00	1.45E-01
2300X-1-CJ-FCV3-006	1.0 - 1.5	1.26E-01	0.00E+00	**5.14E+01**	0.00E+00	0.00E+00	3.56E-01	2.98E-01

Location 7								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-007	0.0 - 0.5	**9.95E+00**	**1.99E+00**	**8.88E+02**	1.35E+00	2.89E-01	0.00E+00	2.68E-01
2300X-1-CJ-FCV2-007	0.5 - 1.0	**1.29E+00**	1.90E-01	**1.28E+02**	3.10E-01	1.86E-02	0.00E+00	6.81E-02
2300X-1-CJ-FCV3-007	1.0 - 1.5	4.74E-01	1.40E-02	**2.50E+01**	2.89E-01	1.54E-01	3.48E-01	0.00E+00

Location 8								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-008	0.0 - 0.5	**7.76E+00**	3.45E-01	**5.18E+01**	0.00E+00	3.16E-02	0.00E+00	0.00E+00
2300X-1-CJ-FCV2-008	0.5 - 1.0	**1.78E-01**	1.40E-01	**5.21E-01**	0.00E+00	8.08E-03	9.69E-02	0.00E+00
2300X-1-CJ-FCV3-008	1.0 - 1.5	3.61E-01	0.00E+00	**1.48E+00**	0.00E+00	0.00E+00	1.98E-01	0.00E+00

Location 9								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-009	0.0 - 0.5	3.55E-01	1.71E-01	**1.43E+01**	2.66E-01	8.80E-02	2.21E-01	0.00E+00
2300X-1-CJ-FCV2-009	0.5 - 1.0	8.91E-02	2.74E-01	**9.09E-01**	0.00E+00	0.00E+00	0.00E+00	1.48E-01
2300X-1-CJ-FCV3-009	1.0 - 1.5	1.05E-01	1.26E-01	**4.04E-01**	1.36E-01	0.00E+00	1.56E-01	0.00E+00

Location 10								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-010	0.0 - 0.5	**9.86E-01**	5.56E-01	**4.94E+01**	0.00E+00	0.00E+00	1.64E-01	0.00E+00
2300X-1-CJ-FCV2-010	0.5 - 1.0	1.63E-01	3.50E-02	**5.78E+00**	0.00E+00	1.44E-01	5.30E-02	0.00E+00
2300X-1-CJ-FCV3-010	1.0 - 1.5	1.35E-01	5.20E-02	1.28E-01	0.00E+00	0.00E+00	0.00E+00	1.38E-02

Location 11								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-011	0.0 - 0.5	**1.55E+00**	3.84E-03	**1.56E+01**	1.53E-01	0.00E+00	6.54E-03	7.33E-02
2300X-1-CJ-FCV2-011	0.5 - 1.0	1.83E-01	1.46E-01	**6.13E-01**	0.00E+00	1.07E-01	0.00E+00	1.30E-01
2300X-1-CJ-FCV3-011	1.0 - 1.5	1.75E-01	0.00E+00	**1.03E+00**	3.13E-03	1.26E-01	0.00E+00	0.00E+00

Location 12								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-012	0.0 - 0.5	7.45E-02	2.84E-02	**5.00E+01**	2.18E-01	0.00E+00	1.83E-01	0.00E+00
2300X-1-CJ-FCV2-012	0.5 - 1.0	9.80E-02	0.00E+00	**1.56E+00**	0.00E+00	0.00E+00	1.64E-01	0.00E+00
2300X-1-CJ-FCV3-012	1.0 - 1.5	6.53E-02	0.00E+00	**1.02E+00**	0.00E+00	0.00E+00	0.00E+00	1.33E-01

Location 13								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-013	0.0 - 0.5	2.08E-01	1.35E-01	**1.16E+01**	0.00E+00	2.78E-01	0.00E+00	1.29E-01
2300X-1-CJ-FCV2-013	0.5 - 1.0	0.00E+00	1.36E-01	**4.21E-01**	1.99E-01	2.53E-01	4.59E-01	3.21E-01
2300X-1-CJ-FCV3-013	1.0 - 1.5	1.12E-01	0.00E+00	1.13E-01	5.59E-02	3.10E-02	0.00E+00	1.21E-01
Location 14								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-014	0.0 - 0.5	2.26E-01	8.94E-02	**2.21E+01**	1.13E-02	1.09E-01	0.00E+00	0.00E+00
2300X-1-CJ-FCV2-014	0.5 - 1.0	1.09E-01	0.00E+00	2.28E-02	1.95E-01	0.00E+00	1.17E-01	0.00E+00
2300X-1-CJ-FCV3-014	1.0 - 1.5	1.24E-01	5.53E-03	3.76E-02	1.18E-01	5.10E-02	0.00E+00	1.00E-01
Location 15								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-015	0.0 - 0.5	1.20E-01	0.00E+00	**8.53E+00**	2.33E-02	0.00E+00	3.33E-01	0.00E+00
2300X-1-CJ-FCV2-015	0.5 - 1.0	2.10E-01	0.00E+00	7.61E-02	3.49E-03	1.09E-01	2.96E-01	3.73E-02
2300X-1-CJ-FCV3-015	1.0 - 1.5	0.00E+00	1.07E-01	**2.55E-01**	0.00E+00	0.00E+00	1.15E-01	0.00E+00
Location 16								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2300X-1-CJ-FCV1-016	0.0 - 0.5	3.21E-01	4.59E-03	**4.58E+01**	0.00E+00	0.00E+00	0.00E+00	8.74E-02
2300X-1-CJ-FCV2-016	0.5 - 1.0	0.00E+00	2.13E-03	**2.88E+00**	7.81E-02	3.64E-02	1.14E-01	0.00E+00
2300X-1-CJ-FCV3-016	1.0 - 1.5	1.26E-01	2.93E-01	**8.11E-01**	0.00E+00	4.01E-02	0.00E+00	4.99E-02

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-29 2300 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	1.40E+02	3.97E+00	1.69E-01	2.02E+01
Cs-134	0.00E+00	3.54E+00	2.38E-01	3.17E-02	6.05E-01
Cs-137	2.28E-02	9.83E+03	6.63E+02	2.54E+01	2.02E+03
Eu-152	0.00E+00	3.63E+00	2.11E-01	1.56E-03	6.11E-01
Eu-154	0.00E+00	2.60E+00	1.76E-01	1.34E-02	4.83E-01
Eu-155	0.00E+00	5.04E+00	3.21E-01	1.14E-01	8.37E-01
Am-241	0.00E+00	7.86E-01	2.11E-01	2.01E-02	1.91E-01

Table 2-30 2600 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2600X-1-CJ-FCV1-001	0.0 - 0.5	1.75E-01	0.00E+00	**6.74E+00**	1.48E-01	5.09E-02	0.00E+00	0.00E+00
2600X-1-CJ-FCV2-001	0.5 - 1.0	1.30E-01	0.00E+00	**4.04E-01**	9.45E-02	3.26E-02	2.04E-01	0.00E+00
2600X-1-CJ-FCV3-001	1.0 - 1.5	8.21E-02	6.05E-02	**2.89E-01**	3.14E-01	1.17E-01	6.24E-02	0.00E+00
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2600X-1-CJ-FCV1-002	0.0 - 0.5	**1.34E+00**	0.00E+00	**6.54E+01**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2600X-1-CJ-FCV2-002	0.5 - 1.0	1.68E-01	0.00E+00	**1.13E+00**	0.00E+00	0.00E+00	1.70E-01	3.74E-02
2600X-1-CJ-FCV3-002	1.0 - 1.5	2.59E-01	0.00E+00	**2.25E+00**	0.00E+00	0.00E+00	1.70E-01	0.00E+00
Location 3								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2600X-1-CJ-FCV1-003	0.0 - 0.5	8.34E-02	0.00E+00	**5.51E+00**	0.00E+00	0.00E+00	4.88E-01	0.00E+00
2600X-1-CJ-FCV2-003	0.5 - 1.0	8.81E-02	0.00E+00	3.86E-01	0.00E+00	9.63E-02	6.83E-02	0.00E+00
2600X-1-CJ-FCV3-003	1.0 - 1.5	1.03E-01	6.68E-02	1.99E-01	1.98E-01	1.65E-01	0.00E+00	0.00E+00
Location 4								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
2600X-1-CJ-FCV1-004	0.0 - 0.5	6.68E-02	1.43E-02	**1.55E+00**	0.00E+00	0.00E+00	2.23E-01	9.96E-03
2600X-1-CJ-FCV2-004	0.5 - 1.0	7.45E-02	0.00E+00	1.40E-01	0.00E+00	5.09E-02	3.20E-01	0.00E+00
2600X-1-CJ-FCV3-004	1.0 - 1.5	9.25E-02	1.25E-02	2.55E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-31 2600 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	6.68E-02	1.34E+00	2.22E-01	9.76E-02	3.56E-01
Cs-134	0.00E+00	6.68E-02	1.28E-02	0.00E+00	2.43E-02
Cs-137	1.40E-01	6.54E+01	7.02E+00	7.69E-01	1.85E+01
Eu-152	0.00E+00	3.14E-01	6.28E-02	0.00E+00	1.05E-01
Eu-154	0.00E+00	1.65E-01	4.28E-02	1.63E-02	5.61E-02
Eu-155	0.00E+00	4.88E-01	1.42E-01	1.19E-01	1.52E-01
Am-241	0.00E+00	3.74E-02	6.28E-02	0.00E+00	1.09E-02

Table 2-32 3100 Gamma Spectroscopy Results for Concrete Samples

Sample ID	Depth (inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
Location 1								
3100X-3-CJ-FCV1-001	0.0 - 0.5	3.81E-02	0.00E+00	0.00E+00	6.40E-02	3.19E-02	2.46E-01	2.56E-01
Location 2								
3100X-3-CJ-FCV1-002	0.0 - 0.5	4.69E-02	2.59E-01	3.00E-02	0.00E+00	1.61E-01	7.28E-02	0.00E+00
Location 3								
3100X-3-CJ-FCV1-003	0.0 - 0.5	6.41E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.60E-01
Location 4								
3100X-3-CJ-FCV1-004	0.0 - 0.5	5.00E-02	0.00E+00	5.45E-02	2.86E-01	0.00E+00	0.00E+00	0.00E+00
Location 5								
3100X-3-CJ-FCV1-005	0.0 - 0.5	5.53E-02	0.00E+00	0.00E+00	6.39E-02	1.34E-01	9.83E-02	0.00E+00
Location 6								
3100X-3-CJ-FCV1-006	0.0 - 0.5	1.11E-01	0.00E+00	7.63E-02	0.00E+00	0.00E+00	6.31E-02	3.88E-02
Location 7								
3100X-3-CJ-FCV1-007	0.0 - 0.5	3.39E-02	0.00E+00	5.31E-02	2.14E-01	5.76E-02	1.66E-01	0.00E+00
Location 8								
3100X-3-CJ-FCV1-008	0.0 - 0.5	0.00E+00	1.88E-01	0.00E+00	1.13E-02	8.34E-02	3.86E-02	0.00E+00
Location 9								
3100X-3-CJ-FCV1-009	0.0 - 0.5	0.00E+00	4.01E-02	6.01E-02	9.25E-02	7.71E-02	1.43E-01	0.00E+00

Activity values in pCi/g.

Location 10								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
3100X-3-CJ-FCV1-010	0.0 - 0.5	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.60E-02	1.50E-01	0.00E+00
Location 11								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
3100X-3-CJ-FCV1-011	0.0 - 0.5	2.48E-02	0.00E+00	9.59E-02	7.99E-02	6.88E-02	0.00E+00	0.00E+00
Location 12								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
3100X-3-CJ-FCV1-012	0.0 - 0.5	0.00E+00	8.51E-02	0.00E+00	1.39E-01	7.98E-03	2.96E-02	0.00E+00
Location 13								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
3100X-3-CJ-FCV1-013	0.0 - 0.5	1.28E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.55E-01	0.00E+00
Location 14								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
3100X-3-CJ-FCV1-014	0.0 - 0.5	5.49E-02	0.00E+00	1.60E-01	0.00E+00	0.00E+00	1.12E-01	1.43E-01
Location 15								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
3100X-3-CJ-FCV1-015	0.0 - 0.5	1.13E-01	2.34E-02	9.61E-02	0.00E+00	0.00E+00	1.06E-01	0.00E+00
Location 16								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
3100X-3-CJ-FCV1-016	0.0 - 0.5	5.01E-04	0.00E+00	0.00E+00	1.78E-01	0.00E+00	3.64E-01	0.00E+00
Location 17								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
3100X-3-CJ-FCV1-017	0.0 - 0.5	9.34E-02	0.00E+00	0.00E+00	0.00E+00	5.16E-02	0.00E+00	0.00E+00
Location 18								
Sample ID	**Depth (inches)**	**Activity (pCi/g)**						
		Co-60	**Cs-134**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**Am-241**
3100X-3-CJ-FCV1-018	0.0 - 0.5	6.05E-02	0.00E+00	0.00E+00	1.13E-03	0.00E+00	2.79E-01	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-33 3100 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	1.11E-01	4.85E-02	4.84E-02	4.16E-02
Cs-134	0.00E+00	2.59E-01	3.30E-02	0.00E+00	7.35E-02
Cs-137	0.00E+00	1.60E-01	3.48E-02	0.00E+00	4.75E-02
Eu-152	0.00E+00	2.86E-01	6.27E-02	6.21E-03	8.77E-02
Eu-154	0.00E+00	1.61E-01	4.22E-02	1.99E-02	5.08E-02
Eu-155	0.00E+00	3.64E-01	1.18E-01	1.02E-01	1.09E-01
Am-241	0.00E+00	2.56E-01	6.27E-02	0.00E+00	7.41E-02

Table 2-34 4100 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4100X-1-CJ-FCV1-001	0.0 - 0.5	1.88E-01	0.00E+00	**8.64E-01**	0.00E+00	0.00E+00	3.45E-01	0.00E+00
4100X-1-CJ-FCV2-001	0.5 - 1.0	1.47E-01	0.00E+00	**0.00E+00**	5.03E-01	1.73E-01	2.51E-03	0.00E+00
4100X-1-CJ-FCV3-001	1.0 - 1.5	1.25E-01	0.00E+00	**6.82E-02**	0.00E+00	1.31E-01	7.73E-02	7.09E-02
Location 2								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4100X-1-CJ-FCV1-002	0.0 - 0.5	**4.21E-01**	0.00E+00	**2.04E-01**	0.00E+00	1.90E-01	3.30E-02	0.00E+00
4100X-1-CJ-FCV2-002	0.5 - 1.0	0.00E+00	0.00E+00	**0.00E+00**	0.00E+00	0.00E+00	3.88E-01	0.00E+00
4100X-1-CJ-FCV3-002	1.0 - 1.5	1.38E-01	0.00E+00	**1.40E-02**	0.00E+00	1.81E-01	1.12E-01	0.00E+00
Location 3								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4100X-1-CJ-FCV1-003	0.0 - 0.5	**5.36E+00**	1.94E-01	**4.51E+00**	1.88E-01	1.33E-01	1.79E-01	0.00E+00
4100X-1-CJ-FCV2-003	0.5 - 1.0	**1.88E-01**	0.00E+00	**3.30E-01**	0.00E+00	2.29E-01	0.00E+00	0.00E+00
4100X-1-CJ-FCV3-003	1.0 - 1.5	4.73E-01	0.00E+00	**1.46E-01**	9.85E-03	0.00E+00	0.00E+00	0.00E+00
Location 4								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-65	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-246
4100X-1-CJ-FCV1-004	0.0 - 0.5	3.21E-02	0.00E+00	**6.79E+00**	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4100X-1-CJ-FCV2-004	0.5 - 1.0	9.06E-02	0.00E+00	**5.54E-01**	0.00E+00	0.00E+00	7.82E-02	0.00E+00
4100X-1-CJ-FCV3-004	1.0 - 1.5	1.61E-01	7.73E-02	**2.80E-01**	0.00E+00	0.00E+00	2.01E-02	0.00E+00
Location 5								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4100X-1-CJ-FCV1-005	0.0 - 0.5	**1.86E+00**	0.00E+00	**5.73E+00**	1.99E-01	2.56E-01	1.43E-01	0.00E+00
4100X-1-CJ-FCV2-005	0.5 - 1.0	4.00E-01	0.00E+00	**5.30E-01**	9.54E-02	0.00E+00	0.00E+00	0.00E+00
4100X-1-CJ-FCV3-005	1.0 - 1.5	1.06E-01	0.00E+00	**0.00E+00**	3.45E-03	1.03E-01	1.33E-01	0.00E+00
Location 6								
Sample ID	Depth	Activity (pCi/g)						
	(inches)	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4100X-1-CJ-FCV1-006	0.0 - 0.5	0.00E+00	0.00E+00	**1.53E-01**	0.00E+00	0.00E+00	5.58E-01	0.00E+00
4100X-1-CJ-FCV2-006	0.5 - 1.0	**1.44E-01**	0.00E+00	**1.30E-01**	1.41E-01	0.00E+00	2.82E-01	0.00E+00
4100X-1-CJ-FCV3-006	1.0 - 1.5	1.34E-01	0.00E+00	**2.97E-01**	1.77E-01	1.32E-01	1.98E-01	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-35 4100 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	5.36E+00	5.54E-01	1.45E-01	1.27E+00
Cs-134	0.00E+00	1.94E-01	1.50E-02	0.00E+00	4.81E-02
Cs-137	0.00E+00	6.79E+00	1.14E+00	2.42E-01	2.13E+00
Eu-152	0.00E+00	5.03E-01	7.32E-02	0.00E+00	1.31E-01
Eu-154	0.00E+00	2.56E-01	8.49E-02	5.14E-02	9.39E-02
Eu-155	0.00E+00	5.58E-01	1.42E-01	9.53E-02	1.60E-01
Am-241	0.00E+00	7.09E-02	7.32E-02	0.00E+00	1.67E-02

Table 2-36 4200 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4200X-3-CJ-FCV1-001	0.0 - 0.5	7.85E-02	0.00E+00	2.56E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Location 2								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4200X-3-CJ-FCV1-002	0.0 - 0.5	1.78E-01	0.00E+00	0.00E+00	1.60E-01	0.00E+00	1.79E-02	0.00E+00
Location 3								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4200X-3-CJ-FCV1-003	0.0 - 0.5	1.33E-01	1.63E-01	1.39E-01	5.08E-02	1.05E-01	0.00E+00	4.14E-02
Location 4								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4200X-3-CJ-FCV1-004	0.0 - 0.5	0.00E+00	4.51E-01	4.83E-04	0.00E+00	2.36E-02	0.00E+00	1.36E-01
Location 5								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4200X-3-CJ-FCV1-005	0.0 - 0.5	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.09E-01	0.00E+00
Location 6								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4200X-3-CJ-FCV1-006	0.0 - 0.5	1.54E-01	1.85E-02	0.00E+00	0.00E+00	3.39E-02	0.00E+00	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-37 4200 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	1.78E-01	9.04E-02	1.06E-01	7.73E-02
Cs-134	0.00E+00	4.51E-01	1.05E-01	9.25E-03	1.81E-01
Cs-137	0.00E+00	1.39E-01	2.75E-02	2.41E-04	5.55E-02
Eu-152	0.00E+00	1.60E-01	3.51E-02	0.00E+00	6.45E-02
Eu-154	0.00E+00	1.05E-01	2.70E-02	1.18E-02	4.07E-02
Eu-155	0.00E+00	2.09E-01	3.78E-02	0.00E+00	8.41E-02
Am-241	0.00E+00	1.36E-01	3.51E-02	0.00E+00	5.48E-02

Table 2-38 4300 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4300X-3-CJ-FCV1-001	0.0 - 0.5	8.66E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.74E-01	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-39 4400 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4400X-3-CJ-FCV1-001	0.0 - 0.5	1.65E-02	0.00E+00	7.13E-02	1.68E-02	6.39E-02	1.39E-01	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-40 4500 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4500X-3-CJ-FCV1-001	0.0 - 0.5	9.71E-02	4.20E-01	0.00E+00	0.00E+00	0.00E+00	1.51E-01	1.54E-01

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-41 4600 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4600X-3-CJ-FCV1-001	0.0 - 0.5	1.05E-01	0.00E+00	1.59E-02	1.12E-01	2.93E-03	0.00E+00	0.00E+00
Location 2								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4600X-3-CJ-FCV1-002	0.0 - 0.5	0.00E+00	1.84E-01	0.00E+00	0.00E+00	1.85E-01	1.90E-01	6.93E-02

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-42 4600 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	1.05E-01	5.23E-02	5.23E-02	7.39E-02
Cs-134	0.00E+00	1.84E-01	9.19E-02	9.19E-02	1.30E-01
Cs-137	0.00E+00	1.59E-02	7.94E-03	7.94E-03	1.12E-02
Eu-152	0.00E+00	1.12E-01	5.59E-02	5.59E-02	7.91E-02
Eu-154	2.93E-03	1.85E-01	9.40E-02	9.40E-02	1.29E-01
Eu-155	0.00E+00	1.90E-01	9.50E-02	9.50E-02	1.34E-01
Am-241	0.00E+00	6.93E-02	5.59E-02	3.46E-02	4.90E-02

Table 2-43 4700 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4700X-3-CJ-FCV1-001	0.0 - 0.5	1.14E-01	6.91E-03	1.54E-01	1.94E-01	0.00E+00	3.26E-01	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-44 4800 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4800X-3-CJ-FCV1-001	0.0 - 0.5	9.21E-02	2.35E-01	0.00E+00	1.38E-01	0.00E+00	8.69E-02	0.00E+00
Location 2								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4800X-3-CJ-FCV1-002	0.0 - 0.5	2.95E-02	0.00E+00	1.16E-01	0.00E+00	0.00E+00	8.96E-02	0.00E+00
Location 3								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4800X-3-CJ-FCV1-003	0.0 - 0.5	1.08E-01	7.48E-02	**7.67E-02**	0.00E+00	3.76E-02	1.28E-02	0.00E+00
4800X-3-CJ-FCV2-003	0.5 - 1.0	2.20E-01	0.00E+00	**1.26E-01**	0.00E+00	0.00E+00	1.48E-01	0.00E+00
4800X-3-CJ-FCV3-003	1.0 - 1.5	8.64E-02	0.00E+00	2.78E-01	1.58E-01	0.00E+00	1.52E-01	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-45 4800 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	2.95E-02	2.20E-01	1.07E-01	9.21E-02	6.97E-02
Cs-134	0.00E+00	2.35E-01	6.20E-02	0.00E+00	1.02E-01
Cs-137	0.00E+00	2.78E-01	1.19E-01	1.16E-01	1.02E-01
Eu-152	0.00E+00	1.58E-01	5.90E-02	0.00E+00	8.11E-02
Eu-154	0.00E+00	3.76E-02	7.51E-03	0.00E+00	1.68E-02
Eu-155	1.28E-02	1.52E-01	9.79E-02	8.96E-02	5.67E-02
Am-241	0.00E+00	0.00E+00	5.90E-02	0.00E+00	0.00E+00

Table 2-46 4900 Gamma Spectroscopy Results for Concrete Samples

Location 1								
Sample ID	Depth (inches)	Activity (pCi/g)						
		Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Eu-155	Am-241
4900X-3-CJ-FCV1-001	0.0 - 0.5	6.10E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Table 2-47 GEL Laboratories Results for Concrete Samples (pCi/g)

Sample ID	H-3	C-14	Mn-54	Fe-55	Co-57	Co-58	Ni-59	Ni-63	Co-60	Zn-65	Nb-94	Tc-99	Sr-90	Ag-110m	Sb-125
1100X-1-CJ-WCV1-004	**2.99E+03**	**1.06E+03**	4.86E+00	**3.99E+03**	0.00E+00	2.06E+01	**3.10E+01**	**2.36E+03**	**2.23E+03**	0.00E+00	2.13E+00	**2.56E+00**	**9.88E+00**	4.58E+00	0.00E+00
1100X-1-CJ-FCV1-005	**1.79E+03**	**1.56E+03**	9.36E-01	**3.77E+03**	0.00E+00	0.00E+00	**1.12E+01**	**1.05E+03**	**1.09E+03**	0.00E+00	1.87E-01	4.51E-02	**6.20E+00**	1.17E+00	0.00E+00
1200X-1-CJ-FCV1-001	**7.89E+02**	**1.53E+03**	0.00E+00	0.00E+00	2.07E-02	3.55E-01	7.12E-02	**1.30E+02**	**4.78E+01**	2.40E-01	9.98E-02	6.85E-01	**7.05E-01**	4.18E-01	5.76E-01
1200X-1-CJ-FCV1-002	**7.68E+02**	**2.67E+03**	6.89E-01	5.49E+00	1.42E-01	0.00E+00	0.00E+00	**5.43E+02**	**8.67E+01**	5.17E-01	1.36E-01	4.11E-01	**2.19E+00**	0.00E+00	0.00E+00
1200X-1-CJ-FCV1-003	**7.38E+02**	**1.99E+03**	5.61E-01	0.00E+00	5.06E-02	1.93E-02	2.23E+00	**2.03E+02**	**4.18E+01**	3.58E-02	3.32E-01	5.62E-01	**4.30E+00**	2.40E-01	3.07E-01
1200X-1-CJ-WCV1-005	**3.36E+02**	**7.19E+02**	2.13E-02	0.00E+00	0.00E+00	0.00E+00	4.68E-02	**2.17E+01**	**2.43E+00**	1.07E-01	1.96E-02	2.21E-01	**1.07E-01**	0.00E+00	3.76E-01
1200X-1-CJ-WCV1-009	**2.46E+02**	**2.62E+03**	0.00E+00	0.00E+00	7.03E-02	2.95E-01	5.12E-01	**1.08E+01**	**9.35E-01**	0.00E+00	4.43E-03	6.60E-02	8.16E-02	9.26E-02	0.00E+00
1200X-1-CJ-FCV1-010	**9.29E+02**	**1.45E+02**	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	**1.41E+02**	**1.26E+01**	9.71E-01	0.00E+00	4.07E-01	**5.07E-01**	1.53E-01	0.00E+00
1200X-1-CJ-FCV1-018	**8.99E+02**	**1.59E+03**	3.37E-02	6.27E+00	2.50E-02	3.50E-02	0.00E+00	**5.76E+01**	**1.81E+00**	0.00E+00	5.93E-02	8.56E-02	**4.51E-01**	0.00E+00	1.22E+00
1200X-1-CJ-FCV1-023	**2.80E+02**	**8.09E+01**	4.50E-01	0.00E+00	1.45E+00	2.19E-01	**1.09E+01**	**1.13E+03**	**7.85E+01**	0.00E+00	1.06E-01	2.09E-01	**1.72E+01**	1.17E-01	6.66E+00
1200X-1-CJ-FCV1-025	**6.97E+02**	**2.70E+03**	1.20E-01	0.00E+00	2.44E-01	0.00E+00	2.31E-01	**1.19E+02**	**1.21E+01**	4.96E-03	0.00E+00	2.43E-01	**2.08E+00**	0.00E+00	3.11E+00
1200X-1-CJ-FCV1-027	**5.38E+02**	**1.62E+03**	3.04E-01	0.00E+00	4.08E-02	4.15E-01	8.96E-01	**1.00E+02**	**3.18E+01**	0.00E+00	2.04E-01	0.00E+00	**1.05E+00**	0.00E+00	1.73E+00
1300X-1-CJ-FCV1-003	**4.11E+02**	**1.41E+03**	9.31E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	**1.47E+02**	**1.84E+01**	0.00E+00	1.76E-01	4.44E-01	**5.09E-01**	3.16E-01	7.01E-01
1300X-1-CJ-FCV1-006	**1.95E+02**	**1.72E+02**	1.53E-02	5.29E+00	1.22E-01	9.73E-02	0.00E+00	**3.54E+01**	**2.48E+00**	3.60E-02	0.00E+00	0.00E+00	**2.42E-01**	0.00E+00	3.95E-01
1300X-1-CJ-FCV1-008	**8.12E+01**	**3.03E+02**	1.41E-01	2.72E+00	4.60E-01	2.85E-01	1.69E+00	**4.21E+01**	**3.07E+00**	0.00E+00	0.00E+00	1.01E-01	**1.43E+00**	0.00E+00	2.06E-01
1400X-1-CJ-FCV1-002	**3.88E+02**	**8.55E+03**	0.00E+00	0.00E+00	0.00E+00	2.06E-02	2.03E+00	**4.86E+01**	**3.47E+00**	6.77E-02	0.00E+00	1.72E-01	**1.03E+01**	0.00E+00	4.18E-01
1400X-1-CJ-FCV1-021	**5.08E+02**	**2.07E+03**	4.51E-03	0.00E+00	1.18E-01	0.00E+00	1.86E-01	**6.37E+00**	**3.55E+00**	0.00E+00	5.45E-02	1.76E-01	0.00E+00	9.97E-02	0.00E+00
2100X-1-CJ-FCV1-006	**2.99E+01**	2.87E+00	6.03E-02	0.00E+00	0.00E+00	1.60E-01	0.00E+00	**3.24E+01**	**2.73E+00**	0.00E+00	0.00E+00	1.14E-01	**4.42E-01**	1.95E-01	0.00E+00
2100X-1-CJ-FCV1-011	7.05E+00	3.64E+00	1.38E-02	0.00E+00	0.00E+00	0.00E+00	4.02E-01	**6.20E+01**	**4.32E+00**	0.00E+00	4.43E-02	**5.53E+00**	**1.99E-01**	7.20E-02	0.00E+00
2100X-1-CJ-FCV1-014	1.13E+00	**7.85E+00**	0.00E+00	0.00E+00	2.23E-02	0.00E+00	0.00E+00	**4.81E+01**	**9.76E-01**	0.00E+00	4.14E-02	**2.60E+01**	**2.47E-01**	3.82E-02	1.18E-01
2200X-1-CJ-FCV1-010	**1.85E+01**	**9.15E+00**	9.11E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	**1.36E+02**	**1.00E+01**	0.00E+00	2.38E-02	0.00E+00	**1.33E+00**	0.00E+00	1.40E+00
2200X-1-CJ-WCV1-009	6.19E+00	**1.88E+01**	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.08E+00	**6.26E+00**	**4.47E-01**	0.00E+00	0.00E+00	4.35E-01	0.00E+00	0.00E+00	0.00E+00
2300X-1-CJ-FCV1-007	**4.48E+01**	**5.02E+01**	1.01E-01	0.00E+00	3.00E-01	1.98E-01	0.00E+00	**5.70E+01**	**7.75E+00**	0.00E+00	1.18E-01	0.00E+00	**3.70E+00**	0.00E+00	1.17E+00
2600X-1-CJ-FCV1-002	**1.46E+01**	**2.42E+01**	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	**1.91E+01**	**1.93E+00**	1.04E-01	2.89E-02	1.51E-01	5.47E-02	2.09E-02	1.33E-02
2200X-1-CJ-FCV1-008	**9.34E+01**	**2.38E+01**	3.11E-01	0.00E+00	2.00E-01	2.68E-01	0.00E+00	**9.47E+01**	**4.74E+01**	0.00E+00	0.00E+00	0.00E+00	**1.73E+00**	6.29E-02	1.73E+00
2200X-1-CJ-FCV1-020	**5.84E+01**	**1.99E+01**	2.59E-01	3.76E+00	0.00E+00	5.69E-01	**4.91E+00**	**1.42E+03**	**2.34E+01**	2.88E-01	0.00E+00	**1.61E+00**	**7.69E+00**	0.00E+00	3.62E+00
2200X-1-CJ-FCV1-022	**1.61E+01**	**2.28E+01**	0.00E+00	4.81E-01	2.86E-01	0.00E+00	**6.56E+00**	**6.42E+02**	**1.29E+00**	4.19E-03	7.17E-02	0.00E+00	**1.78E+01**	0.00E+00	0.00E+00
2200X-1-CJ-FCV1-026	**1.68E+01**	**1.31E+01**	0.00E+00	6.41E-01	0.00E+00	2.02E-01	0.00E+00	**1.90E+02**	0.00E+00	7.41E-02	0.00E+00	0.00E+00	**2.25E+01**	0.00E+00	0.00E+00
2200X-1-CJ-FCV1-031	**2.40E+02**	**4.97E+00**	4.35E-02	8.58E+00	6.14E-02	7.05E-02	**2.37E+00**	**2.27E+02**	**7.21E+00**	2.06E-02	9.88E-02	0.00E+00	**1.18E+00**	0.00E+00	0.00E+00
2200X-1-CJ-FCV1-035	**1.27E+01**	**1.93E+01**	8.52E-03	**1.23E+01**	2.44E-01	1.24E-01	**3.21E+00**	**4.06E+02**	**1.84E+01**	0.00E+00	4.60E-02	0.00E+00	**1.77E+00**	2.53E-01	5.09E-01

Sample ID	H-3	C-14	Mn-54	Fe-55	Co-57	Co-58	Ni-59	Ni-63	Co-60	Zn-65	Nb-94	Tc-99	Sr-90	Ag-110m	Sb-125
2300X-1-CJ-FCV1-001	2.51E+00	**7.99E+01**	6.44E-02	**1.43E+01**	0.00E+00	0.00E+00	**3.41E+00**	**3.14E+02**	**1.01E+00**	0.00E+00	1.08E-01	0.00E+00	**1.81E+01**	0.00E+00	0.00E+00
2300X-1-CJ-FCV1-002	**1.40E+01**	**5.28E+01**	4.61E-03	1.37E+00	0.00E+00	0.00E+00	**6.57E+00**	**7.73E+02**	**8.40E+00**	0.00E+00	0.00E+00	0.00E+00	**2.37E+01**	0.00E+00	7.27E-01
2300X-1-CJ-FCV1-005	5.52E+00	**3.25E+01**	1.33E-01	**2.35E+01**	8.58E-02	0.00E+00	**4.10E+00**	**4.14E+02**	**2.81E+00**	2.50E-01	9.35E-02	1.81E-01	**6.47E+00**	4.15E-03	0.00E+00
3100X-3-CJ-FCV1-006	0.00E+00	3.26E+00	8.23E-02	1.60E+00	0.00E+00	3.06E-02	0.00E+00	0.00E+00	2.52E-02	0.00E+00	9.05E-04	0.00E+00	1.06E-01	0.00E+00	4.87E-02
3100X-3-CJ-FCV1-014	0.00E+00	1.82E+00	0.00E+00	1.90E-01	2.99E-02	2.68E-02	0.00E+00	1.71E+00	5.57E-02	0.00E+00	0.00E+00	0.00E+00	6.38E-02	5.65E-02	1.27E-02
4100X-1-CJ-FCV1-004	0.00E+00	**8.05E+00**	0.00E+00	2.35E+00	6.70E-03	0.00E+00	7.16E-01	1.86E+00	2.88E-01	0.00E+00	5.86E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4100X-1-CJ-FCV1-005	0.00E+00	8.47E-01	0.00E+00	3.41E+00	1.15E-03	1.45E-02	1.65E-04	4.01E-01	**6.26E-01**	0.00E+00	0.00E+00	0.00E+00	**1.31E+00**	0.00E+00	1.51E-01

Sample ID	Cs-134	Cs-137	Ce-144	Eu-152	Eu-154	Eu-155	Pu-238	Pu-239/ 240	Pu-241	Am-241	Cm-242	Cm-243/ 244
1100X-1-CJ-WCV1-004	**1.08E+02**	**5.34E+03**	0.00E+00	**4.84E+03**	**5.52E+02**	7.20E+00	**1.46E+00**	**1.72E+00**	**2.42E+01**	**2.51E+00**	2.12E-02	**7.43E-01**
1100X-1-CJ-FCV1-005	**1.29E+02**	**4.87E+03**	0.00E+00	**5.57E+03**	**6.11E+02**	**1.83E+01**	**3.02E-01**	**1.23E+00**	**9.86E+00**	**3.17E-01**	1.12E-02	**1.06E-01**
1200X-1-CJ-FCV1-001	1.07E-01	**1.24E+02**	3.03E-01	0.00E+00	4.50E-01	1.96E-01	4.51E-02	0.00E+00	6.68E+00	3.99E-02	0.00E+00	5.43E-03
1200X-1-CJ-FCV1-002	4.30E-01	**4.68E+02**	0.00E+00	1.68E-01	0.00E+00	0.00E+00	**2.61E-01**	**8.37E-02**	**5.38E+00**	**4.90E-01**	0.00E+00	**1.68E-01**
1200X-1-CJ-FCV1-003	**1.15E+00**	**1.07E+03**	2.25E-01	0.00E+00	4.91E-01	0.00E+00	**6.04E-02**	**4.92E-02**	**4.86E+00**	**9.72E-02**	0.00E+00	**3.94E-02**
1200X-1-CJ-WCV1-005	3.54E-02	**9.35E+01**	8.90E-01	0.00E+00	2.78E-01	2.01E-01	8.10E-03	0.00E+00	5.67E+00	1.30E-02	0.00E+00	0.00E+00
1200X-1-CJ-WCV1-009	5.72E-02	**2.97E+02**	0.00E+00	1.34E+00	2.40E-01	0.00E+00	1.60E-02	0.00E+00	3.65E+00	3.02E-02	0.00E+00	2.15E-02
1200X-1-CJ-FCV1-010	6.39E-02	**3.51E+02**	7.99E-01	**2.36E+00**	1.70E-02	4.72E-02	1.58E-02	0.00E+00	2.43E+00	3.55E-02	0.00E+00	1.31E-02
1200X-1-CJ-FCV1-018	1.61E-01	**2.50E+02**	0.00E+00	1.56E-02	0.00E+00	0.00E+00	2.10E-02	0.00E+00	2.40E+00	9.22E-03	6.81E-03	3.14E-03
1200X-1-CJ-FCV1-023	**3.78E+00**	**5.90E+03**	2.82E+00	2.61E-01	1.38E-01	0.00E+00	**7.22E-01**	**3.98E-01**	**1.55E+01**	**9.46E-01**	0.00E+00	**1.16E-01**
1200X-1-CJ-FCV1-025	4.01E-01	**9.55E+02**	1.68E+00	0.00E+00	0.00E+00	0.00E+00	1.87E-03	1.01E-02	3.79E+00	9.23E-03	8.10E-03	0.00E+00
1200X-1-CJ-FCV1-027	**3.58E+00**	**4.47E+03**	0.00E+00	4.01E-01	0.00E+00	0.00E+00	7.86E-03	1.07E-02	5.32E+00	8.05E-03	8.98E-03	1.70E-03
1300X-1-CJ-FCV1-003	**3.94E+00**	**5.35E+03**	0.00E+00	0.00E+00	7.25E-02	6.63E-01	2.22E-02	1.13E-02	**4.73E+00**	3.34E-03	0.00E+00	0.00E+00
1300X-1-CJ-FCV1-006	**9.01E-01**	**8.79E+02**	9.97E-02	0.00E+00	0.00E+00	0.00E+00	6.54E-03	6.53E-04	4.18E+00	0.00E+00	0.00E+00	6.91E-03
1300X-1-CJ-FCV1-008	**5.73E-01**	**1.26E+03**	1.30E+00	1.89E+00	3.43E-01	9.31E-02	2.97E-02	2.31E-02	1.99E+00	**5.54E-02**	0.00E+00	1.86E-03
1400X-1-CJ-FCV1-002	2.35E-01	**2.79E+03**	2.56E+00	0.00E+00	3.15E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.44E-02	0.00E+00	0.00E+00
1400X-1-CJ-FCV1-021	0.00E+00	**9.14E+01**	0.00E+00	5.22E-01	9.85E-02	1.37E-01	1.77E-02	4.34E-02	0.00E+00	1.42E-02	0.00E+00	0.00E+00
2100X-1-CJ-FCV1-006	1.26E-01	**1.14E+02**	1.76E-01	4.23E-01	8.57E-02	9.29E-01	1.59E-02	0.00E+00	1.73E+00	1.04E-02	0.00E+00	0.00E+00
2100X-1-CJ-FCV1-011	1.45E-01	**6.02E+01**	0.00E+00	0.00E+00	4.15E-03	0.00E+00	0.00E+00	0.00E+00	9.98E-02	0.00E+00	0.00E+00	0.00E+00
2100X-1-CJ-FCV1-014	8.66E-02	**1.33E+02**	1.03E+00	0.00E+00	0.00E+00	1.19E-01	2.39E-02	0.00E+00	0.00E+00	0.00E+00	1.99E-02	6.71E-04
2200X-1-CJ-FCV1-010	0.00E+00	**3.35E+02**	0.00E+00	3.35E-01	0.00E+00	4.87E-01	0.00E+00	1.25E-02	0.00E+00	0.00E+00	1.19E-02	0.00E+00
2200X-1-CJ-WCV1-009	9.61E-02	**3.82E+01**	0.00E+00	0.00E+00	0.00E+00	3.02E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.46E-02	0.00E+00

Sample ID	Cs-134	Cs-137	Ce-144	Eu-152	Eu-154	Eu-155	Pu-238	Pu-239/ 240	Pu-241	Am-241	Cm-242	Cm-243/ 244
2300X-1-CJ-FCV1-007	**2.32E+00**	**7.80E+02**	2.53E+00	5.48E-01	2.54E-01	0.00E+00	2.24E-02	3.24E-04	1.13E+00	2.48E-01	0.00E+00	0.00E+00
2600X-1-CJ-FCV1-002	1.15E-01	**6.77E+01**	0.00E+00	9.12E-01	1.10E-01	9.29E-02	0.00E+00	1.39E-02	2.51E-01	2.32E-02	0.00E+00	5.97E-03
2200X-1-CJ-FCV1-008	**1.46E+00**	**1.22E+03**	9.64E-02	0.00E+00	1.88E-01	0.00E+00	1.80E-03	1.35E-02	0.00E+00	1.51E-02	0.00E+00	0.00E+00
2200X-1-CJ-FCV1-020	**2.63E+01**	**1.22E+04**	8.33E-01	3.16E+00	5.91E-02	0.00E+00	8.89E-03	1.40E-02	0.00E+00	2.04E-02	6.83E-03	0.00E+00
2200X-1-CJ-FCV1-022	2.61E-01	**1.87E+04**	0.00E+00	1.61E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-FCV1-026	2.25E-01	**2.12E+04**	4.58E+00	2.41E+00	0.00E+00	0.00E+00	4.16E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2200X-1-CJ-FCV1-031	2.16E-01	**3.93E+02**	5.79E-01	0.00E+00	7.82E-02	0.00E+00	0.00E+00	1.35E-02	0.00E+00	9.34E-03	0.00E+00	1.30E-02
2200X-1-CJ-FCV1-035	1.86E-01	**8.44E+02**	1.67E-01	1.40E-01	6.70E-02	0.00E+00	4.11E-02	3.19E-02	0.00E+00	3.78E-02	0.00E+00	4.90E-03
2300X-1-CJ-FCV1-001	0.00E+00	**7.57E+03**	4.15E-01	0.00E+00	6.70E-02	1.67E+00	0.00E+00	0.00E+00	0.00E+00	7.48E-03	4.27E-03	0.00E+00
2300X-1-CJ-FCV1-002	2.08E-01	**1.05E+04**	0.00E+00	1.76E+00	7.23E-02	0.00E+00	**5.16E-01**	**3.29E-01**	4.08E+00	**8.98E-01**	1.99E-03	**1.10E-01**
2300X-1-CJ-FCV1-005	8.13E-02	**3.61E+03**	4.12E-01	0.00E+00	0.00E+00	0.00E+00	4.02E-03	0.00E+00	0.00E+00	4.79E-02	0.00E+00	0.00E+00
3100X-3-CJ-FCV1-006	0.00E+00	**3.54E-01**	1.55E-01	8.14E-02	0.00E+00	0.00E+00	2.19E-02	6.94E-03	0.00E+00	1.12E-02	6.55E-03	9.32E-03
3100X-3-CJ-FCV1-014	1.20E-01	1.54E-01	5.46E-01	0.00E+00	3.48E-01	5.88E-02	2.16E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4100X-1-CJ-FCV1-004	8.95E-02	**6.85E+00**	0.00E+00	7.01E-02	2.14E-01	0.00E+00	1.15E-02	1.15E-02	0.00E+00	8.82E-04	6.51E-03	0.00E+00
4100X-1-CJ-FCV1-005	0.00E+00	**3.36E+00**	1.34E-01	8.72E-02	1.03E-01	0.00E+00	0.00E+00	0.00E+00	3.96E-01	2.75E-03	0.00E+00	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Attachment 2-2 Results for Open Land Area Soil Samples

Figure 2-26 Survey Unit 8100 Random Sample and Scan Locations

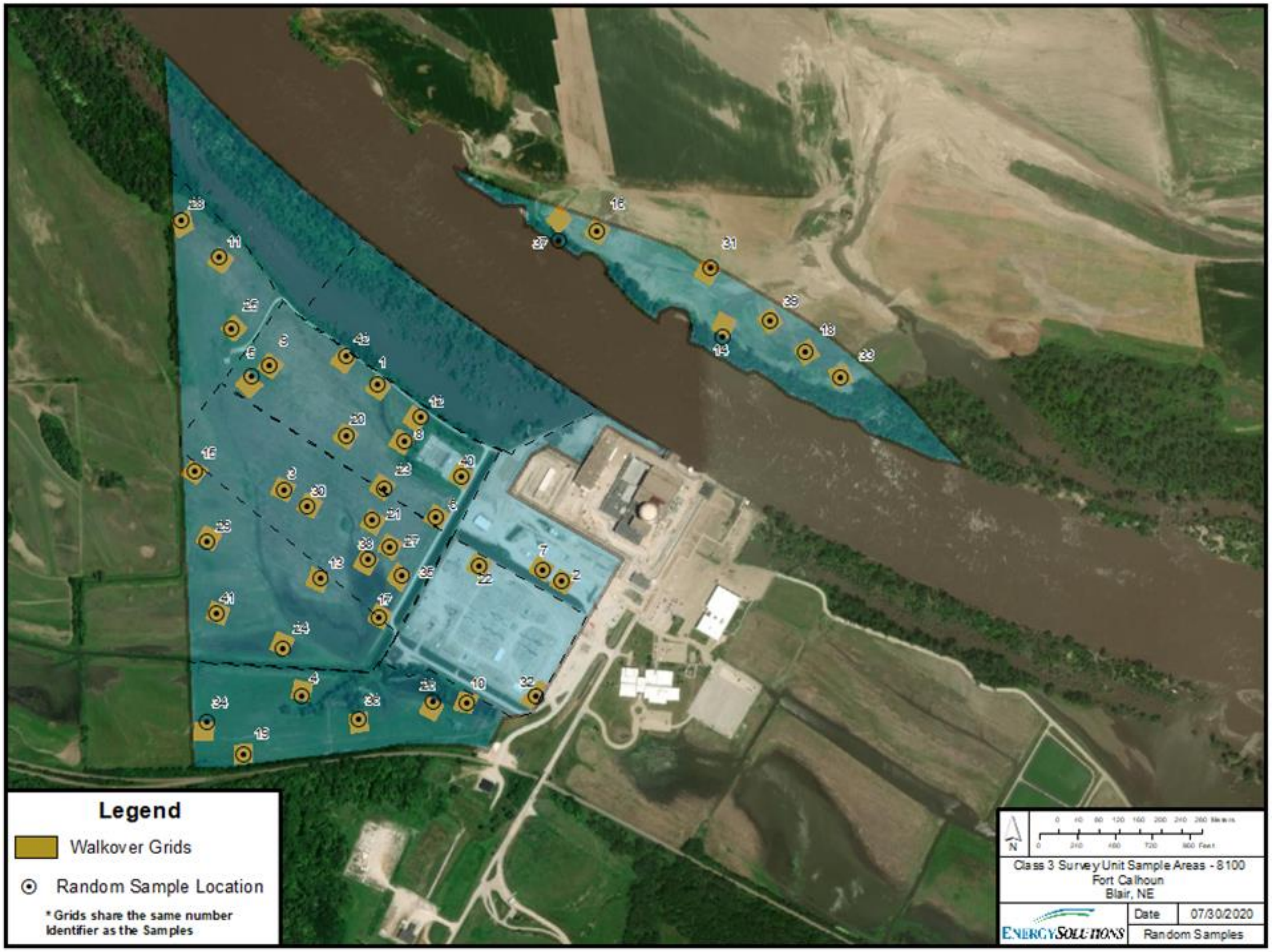


Figure 2-27 Survey Unit 8100 Judgmental Sample and Scan Locations

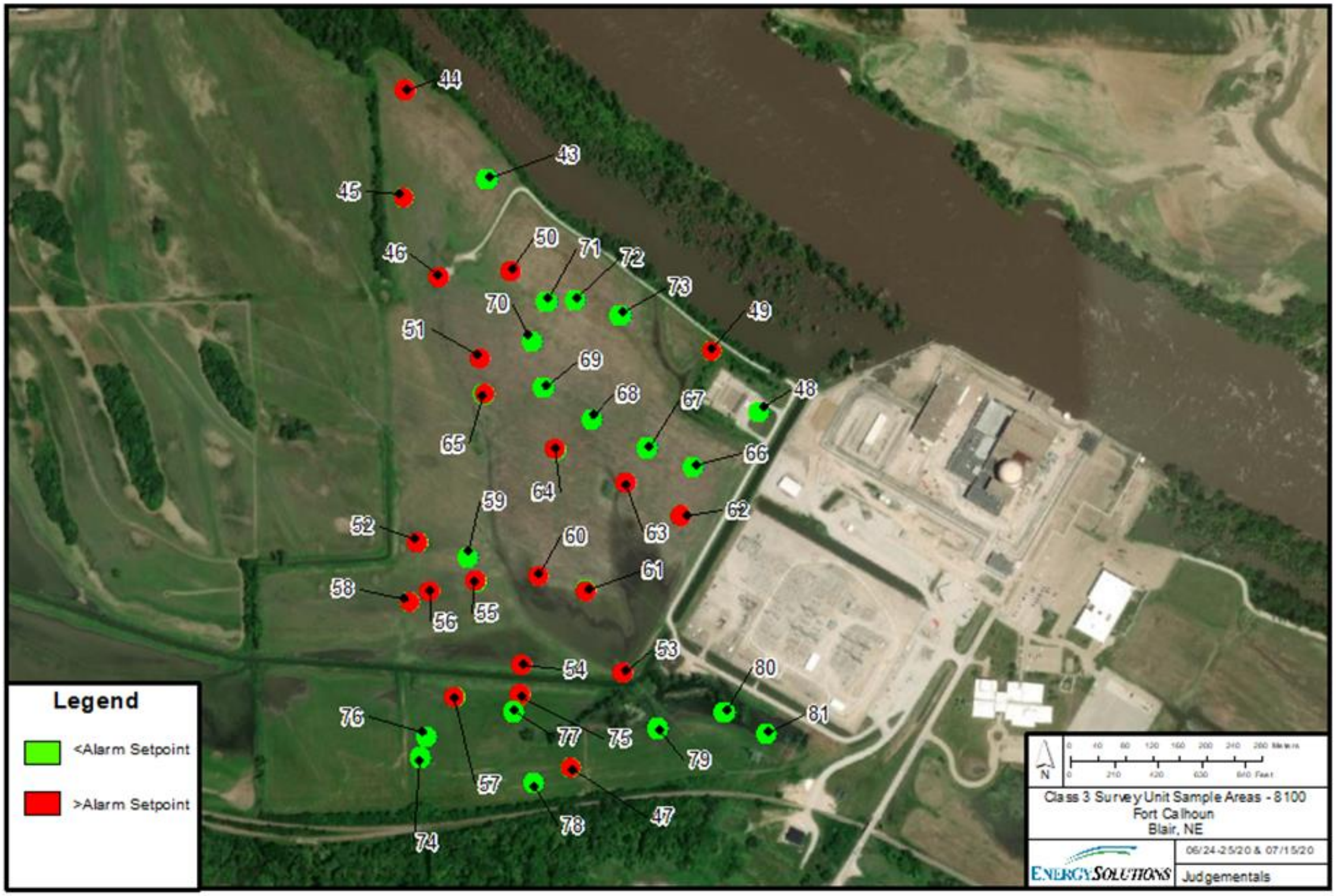


Table 2-48 8100 Gamma Spectroscopy Results for Random Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8100X-3-CR-GSSX-001	Co-60	2.13E-02	N/A	U	1.16E-01	4.96E-02	3.80E+00	5.61E-03	0.0160
	Cs-134	0.00E+00	N/A	U	1.39E-01	8.36E-02	5.70E+00	0.00E+00	
	Cs-137	1.03E-01	2.47E-02		7.41E-02	3.05E-02	1.10E+01	9.36E-03	
	Eu-152	0.00E+00	N/A	U	2.61E-01	1.22E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.14E-01	1.34E-01	8.00E+00	0.00E+00	
	Eu-155	2.79E-01	N/A	U	3.96E-01	1.90E-01	2.80E+02	9.96E-04	
	Am-241	0.00E+00	N/A	U	3.28E-01	1.56E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-002 ROCK	Co-60	7.65E-03	N/A	U	6.08E-02	2.55E-02	3.80E+00	2.01E-03	0.0070
	Cs-134	0.00E+00	N/A	U	7.48E-02	5.70E-02	5.70E+00	0.00E+00	
	Cs-137	1.50E-05	N/A	U	8.03E-02	3.63E-02	1.10E+01	1.36E-06	
	Eu-152	3.94E-02	N/A	U	1.72E-01	8.11E-02	8.70E+00	4.53E-03	
	Eu-154	0.00E+00	N/A	U	1.58E-01	6.53E-02	8.00E+00	0.00E+00	
	Eu-155	1.19E-01	N/A	U	2.20E-01	1.05E-01	2.80E+02	4.25E-04	
	Am-241	0.00E+00	N/A	U	1.73E-01	8.19E-02	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-003	Co-60	3.72E-02	N/A	U	1.66E-01	8.69E-02	3.80E+00	9.79E-03	0.0377
	Cs-134	0.00E+00	N/A	U	2.04E-01	1.37E-01	5.70E+00	0.00E+00	
	Cs-137	2.05E-01	4.29E-02		1.32E-01	5.74E-02	1.10E+01	1.86E-02	
	Eu-152	7.58E-02	N/A	U	4.04E-01	1.91E-01	8.70E+00	8.71E-03	
	Eu-154	0.00E+00	N/A	U	4.52E-01	1.96E-01	8.00E+00	0.00E+00	
	Eu-155	1.50E-01	N/A	U	5.26E-01	2.61E-01	2.80E+02	5.36E-04	
	Am-241	0.00E+00	N/A	U	4.13E-01	1.96E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-004	Co-60	3.34E-02	N/A	U	1.19E-01	5.18E-02	3.80E+00	8.79E-03	0.0405
	Cs-134	0.00E+00	N/A	U	1.38E-01	8.45E-02	5.70E+00	0.00E+00	
	Cs-137	6.36E-02	2.84E-02		1.08E-01	4.78E-02	1.10E+01	5.78E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	3.93E-02	N/A	U	2.96E-01	1.40E-01	8.70E+00	4.52E-03	
	Eu-154	1.62E-01	N/A	U	3.64E-01	1.60E-01	8.00E+00	2.03E-02	
	Eu-155	3.37E-01	N/A	U	4.07E-01	1.96E-01	2.80E+02	1.20E-03	
	Am-241	0.00E+00	N/A	U	3.40E-01	1.62E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-005	Co-60	5.78E-02	N/A	U	1.38E-01	9.52E-02	3.80E+00	1.52E-02	0.0750
	Cs-134	0.00E+00	N/A	U	2.08E-01	1.23E-01	5.70E+00	0.00E+00	
	Cs-137	1.67E-01	4.15E-02		1.35E-01	5.87E-02	1.10E+01	1.52E-02	
	Eu-152	1.06E-01	N/A	U	3.64E-01	1.71E-01	8.70E+00	1.22E-02	
	Eu-154	2.49E-01	N/A	U	5.68E-01	2.52E-01	8.00E+00	3.11E-02	
	Eu-155	3.52E-01	N/A	U	5.26E-01	2.53E-01	2.80E+02	1.26E-03	
	Am-241	0.00E+00	N/A	U	4.51E-01	2.15E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-006	Co-60	5.02E-03	N/A	U	1.31E-01	5.57E-02	3.80E+00	1.32E-03	0.0131
	Cs-134	0.00E+00	N/A	U	1.47E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	1.24E-01	3.25E-02		1.06E-01	4.55E-02	1.10E+01	1.13E-02	
	Eu-152	0.00E+00	N/A	U	3.22E-01	1.51E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.91E-01	1.69E-01	8.00E+00	0.00E+00	
	Eu-155	1.37E-01	N/A	U	2.06E-01	9.37E-02	2.80E+02	4.89E-04	
	Am-241	0.00E+00	N/A	U	3.67E-01	1.74E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-007 ROCK	Co-60	2.80E-02	N/A	U	6.36E-02	3.67E-02	3.80E+00	7.37E-03	0.0083
	Cs-134	0.00E+00	N/A	U	7.08E-02	5.80E-02	5.70E+00	0.00E+00	
	Cs-137	4.99E-03	N/A	U	7.54E-02	3.28E-02	1.10E+01	4.54E-04	
	Eu-152	1.24E-03	N/A	U	1.66E-01	7.66E-02	8.70E+00	1.43E-04	
	Eu-154	2.27E-03	N/A	U	1.77E-01	7.11E-02	8.00E+00	2.84E-04	
	Eu-155	1.96E-02	N/A	U	2.12E-01	1.00E-01	2.80E+02	7.00E-05	
	Am-241	0.00E+00	N/A	U	1.57E-01	7.23E-02	2.10E+00	0.00E+00	
8100X-3-CR-GSSA-008	Co-60	4.80E-02	N/A	U	1.19E-01	6.07E-02	3.80E+00	1.26E-02	0.0610

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.41E-01	8.99E-02	5.70E+00	0.00E+00	
	Cs-137	9.39E-02	N/A	U	1.52E-01	6.96E-02	1.10E+01	8.54E-03	
	Eu-152	0.00E+00	N/A	U	3.04E-01	1.43E-01	8.70E+00	0.00E+00	
	Eu-154	4.08E-02	N/A	U	3.40E-01	1.47E-01	8.00E+00	5.10E-03	
	Eu-155	1.89E-01	N/A	U	3.95E-01	1.92E-01	2.80E+02	6.75E-04	
	Am-241	7.15E-02	N/A	U	3.48E-01	1.66E-01	2.10E+00	3.40E-02	
8100X-3-CR-GSSA-009	Co-60	4.64E-02	N/A	U	7.91E-02	6.53E-02	3.80E+00	1.22E-02	0.0562
	Cs-134	0.00E+00	N/A	U	1.66E-01	9.58E-02	5.70E+00	0.00E+00	
	Cs-137	1.70E-01	N/A	U	1.97E-01	9.15E-02	1.10E+01	1.55E-02	
	Eu-152	9.39E-03	N/A	U	3.16E-01	1.49E-01	8.70E+00	1.08E-03	
	Eu-154	3.64E-02	N/A	U	3.85E-01	1.68E-01	8.00E+00	4.55E-03	
	Eu-155	4.17E-02	N/A	U	3.06E-01	1.44E-01	2.80E+02	1.49E-04	
	Am-241	4.78E-02	N/A	U	3.64E-01	1.74E-01	2.10E+00	2.28E-02	
8100X-3-CR-GSSX-010	Co-60	8.15E-03	N/A	U	9.39E-02	3.91E-02	3.80E+00	2.14E-03	0.0297
	Cs-134	0.00E+00	N/A	U	1.21E-01	9.37E-02	5.70E+00	0.00E+00	
	Cs-137	1.46E-01	N/A	U	1.64E-01	7.60E-02	1.10E+01	1.33E-02	
	Eu-152	0.00E+00	N/A	U	3.00E-01	1.42E-01	8.70E+00	0.00E+00	
	Eu-154	1.13E-01	N/A	U	3.99E-01	1.78E-01	8.00E+00	1.41E-02	
	Eu-155	5.78E-02	N/A	U	3.83E-01	1.84E-01	2.80E+02	2.06E-04	
	Am-241	0.00E+00	N/A	U	3.35E-01	1.60E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-011	Co-60	4.12E-02	N/A	U	9.55E-02	7.03E-02	3.80E+00	1.08E-02	0.0428
	Cs-134	0.00E+00	N/A	U	1.65E-01	9.87E-02	5.70E+00	0.00E+00	
	Cs-137	1.58E-01	3.79E-02		1.26E-01	5.58E-02	1.10E+01	1.44E-02	
	Eu-152	9.48E-03	N/A	U	3.22E-01	1.52E-01	8.70E+00	1.09E-03	
	Eu-154	1.21E-01	N/A	U	3.47E-01	1.49E-01	8.00E+00	1.51E-02	
	Eu-155	3.99E-01	N/A	U	3.10E-01	1.46E-01	2.80E+02	1.43E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	3.52E-01	1.67E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-012	Co-60	0.00E+00	N/A	U	1.27E-01	6.63E-02	3.80E+00	0.00E+00	0.0165
	Cs-134	0.00E+00	N/A	U	1.52E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	1.34E-01	3.43E-02		1.12E-01	4.86E-02	1.10E+01	1.22E-02	
	Eu-152	2.71E-02	N/A	U	3.17E-01	1.48E-01	8.70E+00	3.11E-03	
	Eu-154	0.00E+00	N/A	U	3.55E-01	1.50E-01	8.00E+00	0.00E+00	
	Eu-155	3.29E-01	N/A	U	4.33E-01	2.07E-01	2.80E+02	1.18E-03	
	Am-241	0.00E+00	N/A	U	3.55E-01	1.68E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-013	Co-60	1.15E-01	N/A	U	1.24E-01	9.36E-02	3.80E+00	3.03E-02	0.0626
	Cs-134	0.00E+00	N/A	U	1.86E-01	1.39E-01	5.70E+00	0.00E+00	
	Cs-137	1.73E-01	N/A	U	2.48E-01	1.15E-01	1.10E+01	1.57E-02	
	Eu-152	1.31E-01	N/A	U	3.89E-01	1.84E-01	8.70E+00	1.51E-02	
	Eu-154	0.00E+00	N/A	U	5.08E-01	2.23E-01	8.00E+00	0.00E+00	
	Eu-155	4.32E-01	N/A	U	5.33E-01	2.71E-01	2.80E+02	1.54E-03	
	Am-241	0.00E+00	N/A	U	4.10E-01	1.95E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-014	Co-60	1.62E-02	N/A	U	9.54E-02	4.17E-02	3.80E+00	4.26E-03	0.0119
	Cs-134	0.00E+00	N/A	U	1.06E-01	6.01E-02	5.70E+00	0.00E+00	
	Cs-137	7.24E-02	N/A	U	1.15E-01	5.19E-02	1.10E+01	6.58E-03	
	Eu-152	7.40E-03	N/A	U	2.23E-01	1.05E-01	8.70E+00	8.51E-04	
	Eu-154	0.00E+00	N/A	U	3.03E-01	1.32E-01	8.00E+00	0.00E+00	
	Eu-155	5.15E-02	N/A	U	2.70E-01	1.28E-01	2.80E+02	1.84E-04	
	Am-241	0.00E+00	N/A	U	2.43E-01	1.15E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-015	Co-60	0.00E+00	N/A	U	1.45E-01	6.62E-02	3.80E+00	0.00E+00	0.0224
	Cs-134	0.00E+00	N/A	U	2.09E-01	1.26E-01	5.70E+00	0.00E+00	
	Cs-137	2.38E-01	N/A	U	2.48E-01	1.16E-01	1.10E+01	2.16E-02	
	Eu-152	0.00E+00	N/A	U	3.74E-01	1.76E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	5.03E-01	2.21E-01	8.00E+00	0.00E+00	
	Eu-155	2.11E-01	N/A	U	4.00E-01	1.90E-01	2.80E+02	7.54E-04	
	Am-241	0.00E+00	N/A	U	4.29E-01	2.05E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-016	Co-60	1.33E-02	N/A	U	9.53E-02	5.42E-02	3.80E+00	3.50E-03	0.0248
	Cs-134	0.00E+00	N/A	U	1.19E-01	7.51E-02	5.70E+00	0.00E+00	
	Cs-137	5.98E-02	N/A	U	1.12E-01	5.02E-02	1.10E+01	5.44E-03	
	Eu-152	3.11E-02	N/A	U	2.39E-01	1.12E-01	8.70E+00	3.57E-03	
	Eu-154	9.54E-02	N/A	U	3.30E-01	1.45E-01	8.00E+00	1.19E-02	
	Eu-155	9.47E-02	N/A	U	1.87E-01	8.65E-02	2.80E+02	3.38E-04	
	Am-241	0.00E+00	N/A	U	2.70E-01	1.28E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-017	Co-60	0.00E+00	N/A	U	9.92E-02	5.59E-02	3.80E+00	0.00E+00	0.0236
	Cs-134	0.00E+00	N/A	U	1.48E-01	9.02E-02	5.70E+00	0.00E+00	
	Cs-137	1.82E-01	N/A	U	1.93E-01	9.02E-02	1.10E+01	1.65E-02	
	Eu-152	5.24E-02	N/A	U	2.91E-01	1.37E-01	8.70E+00	6.02E-03	
	Eu-154	0.00E+00	N/A	U	3.60E-01	1.57E-01	8.00E+00	0.00E+00	
	Eu-155	2.91E-01	N/A	U	4.18E-01	2.01E-01	2.80E+02	1.04E-03	
	Am-241	0.00E+00	N/A	U	3.37E-01	1.61E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-018	Co-60	2.70E-02	N/A	U	8.84E-02	3.75E-02	3.80E+00	7.11E-03	0.0278
	Cs-134	0.00E+00	N/A	U	1.08E-01	7.80E-02	5.70E+00	0.00E+00	
	Cs-137	6.90E-02	N/A	U	1.17E-01	5.30E-02	1.10E+01	6.27E-03	
	Eu-152	5.59E-02	N/A	U	2.34E-01	1.10E-01	8.70E+00	6.43E-03	
	Eu-154	5.86E-02	N/A	U	2.90E-01	1.26E-01	8.00E+00	7.33E-03	
	Eu-155	1.99E-01	N/A	U	1.83E-01	8.53E-02	2.80E+02	7.11E-04	
	Am-241	0.00E+00	N/A	U	2.62E-01	1.25E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-019	Co-60	2.56E-02	N/A	U	1.23E-01	5.17E-02	3.80E+00	6.74E-03	0.0599
	Cs-134	0.00E+00	N/A	U	1.47E-01	1.04E-01	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	7.28E-02	N/A	U	1.54E-01	6.91E-02	1.10E+01	6.62E-03	
	Eu-152	0.00E+00	N/A	U	3.15E-01	1.48E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.03E-01	1.74E-01	8.00E+00	0.00E+00	
	Eu-155	1.36E-01	N/A	U	3.12E-01	1.47E-01	2.80E+02	4.86E-04	
	Am-241	9.67E-02	N/A	U	3.97E-01	1.89E-01	2.10E+00	4.60E-02	
8100X-3-CR-GSSX-020	Co-60	0.00E+00	N/A	U	1.09E-01	6.87E-02	3.80E+00	0.00E+00	0.0265
	Cs-134	2.03E-02	N/A	U	1.66E-01	1.16E-01	5.70E+00	3.56E-03	
	Cs-137	1.83E-01	3.90E-02		1.20E-01	5.18E-02	1.10E+01	1.66E-02	
	Eu-152	0.00E+00	N/A	U	3.73E-01	1.76E-01	8.70E+00	0.00E+00	
	Eu-154	4.44E-02	N/A	U	4.50E-01	1.97E-01	8.00E+00	5.55E-03	
	Eu-155	2.03E-01	N/A	U	4.73E-01	2.31E-01	2.80E+02	7.25E-04	
	Am-241	0.00E+00	N/A	U	3.97E-01	1.88E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-021	Co-60	4.81E-02	N/A	U	1.48E-01	6.32E-02	3.80E+00	1.27E-02	0.0606
	Cs-134	0.00E+00	N/A	U	1.72E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	2.10E-01	4.13E-02		1.21E-01	5.23E-02	1.10E+01	1.91E-02	
	Eu-152	1.41E-02	N/A	U	3.40E-01	1.59E-01	8.70E+00	1.62E-03	
	Eu-154	0.00E+00	N/A	U	4.77E-01	2.09E-01	8.00E+00	0.00E+00	
	Eu-155	7.57E-02	N/A	U	3.54E-01	1.67E-01	2.80E+02	2.70E-04	
	Am-241	5.66E-02	N/A	U	4.23E-01	2.01E-01	2.10E+00	2.70E-02	
8100X-3-CR-GSSX-022	Co-60	3.01E-03	N/A	U	1.17E-01	6.00E-02	3.80E+00	7.92E-04	0.0029
	Cs-134	0.00E+00	N/A	U	1.40E-01	9.43E-02	5.70E+00	0.00E+00	
	Cs-137	1.14E-02	N/A	U	1.27E-01	5.72E-02	1.10E+01	1.04E-03	
	Eu-152	0.00E+00	N/A	U	2.80E-01	1.32E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.60E-01	1.58E-01	8.00E+00	0.00E+00	
	Eu-155	3.04E-01	N/A	U	3.98E-01	2.07E-01	2.80E+02	1.09E-03	
	Am-241	0.00E+00	N/A	U	3.42E-01	1.63E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8100X-3-CR-GSSX-023	Co-60	7.84E-02	N/A	U	1.33E-01	7.13E-02	3.80E+00	2.06E-02	0.0372
	Cs-134	0.00E+00	N/A	U	1.76E-01	1.01E-01	5.70E+00	0.00E+00	
	Cs-137	1.72E-01	N/A	U	1.94E-01	8.94E-02	1.10E+01	1.56E-02	
	Eu-152	0.00E+00	N/A	U	2.71E-01	1.26E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.26E-01	1.86E-01	8.00E+00	0.00E+00	
	Eu-155	2.65E-01	N/A	U	4.63E-01	2.23E-01	2.80E+02	9.46E-04	
	Am-241	0.00E+00	N/A	U	3.84E-01	1.82E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-024	Co-60	7.57E-05	N/A	U	1.17E-01	4.92E-02	3.80E+00	1.99E-05	0.0101
	Cs-134	0.00E+00	N/A	U	1.57E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	1.01E-01	3.10E-02		1.07E-01	4.61E-02	1.10E+01	9.18E-03	
	Eu-152	0.00E+00	N/A	U	3.28E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.75E-01	1.61E-01	8.00E+00	0.00E+00	
	Eu-155	2.61E-01	N/A	U	3.63E-01	1.72E-01	2.80E+02	9.32E-04	
	Am-241	0.00E+00	N/A	U	3.60E-01	1.71E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-025	Co-60	3.75E-02	N/A	U	1.16E-01	4.84E-02	3.80E+00	9.87E-03	0.0993
	Cs-134	0.00E+00	N/A	U	1.81E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	1.60E-01	3.63E-02		1.14E-01	4.95E-02	1.10E+01	1.45E-02	
	Eu-152	2.65E-02	N/A	U	3.05E-01	1.43E-01	8.70E+00	3.05E-03	
	Eu-154	1.19E-01	N/A	U	3.85E-01	1.65E-01	8.00E+00	1.49E-02	
	Eu-155	3.41E-01	N/A	U	4.13E-01	2.17E-01	2.80E+02	1.22E-03	
	Am-241	1.17E-01	N/A	U	3.90E-01	1.86E-01	2.10E+00	5.57E-02	
8100X-3-CR-GSSX-026	Co-60	2.38E-02	N/A	U	1.19E-01	5.62E-02	3.80E+00	6.26E-03	0.0339
	Cs-134	0.00E+00	N/A	U	1.53E-01	9.76E-02	5.70E+00	0.00E+00	
	Cs-137	1.03E-01	N/A	U	1.58E-01	7.25E-02	1.10E+01	9.36E-03	
	Eu-152	1.59E-01	N/A	U	3.20E-01	1.52E-01	8.70E+00	1.83E-02	
	Eu-154	0.00E+00	N/A	U	3.81E-01	1.68E-01	8.00E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-155	9.03E-03	N/A	U	2.53E-01	1.19E-01	2.80E+02	3.23E-05	
	Am-241	0.00E+00	N/A	U	3.15E-01	1.50E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-027	Co-60	3.56E-02	N/A	U	1.35E-01	5.85E-02	3.80E+00	9.37E-03	0.0500
	Cs-134	0.00E+00	N/A	U	1.60E-01	9.32E-02	5.70E+00	0.00E+00	
	Cs-137	2.07E-01	3.90E-02		1.16E-01	5.10E-02	1.10E+01	1.88E-02	
	Eu-152	1.12E-02	N/A	U	3.24E-01	1.53E-01	8.70E+00	1.29E-03	
	Eu-154	1.58E-01	N/A	U	4.62E-01	2.06E-01	8.00E+00	1.98E-02	
	Eu-155	2.07E-01	N/A	U	4.45E-01	2.14E-01	2.80E+02	7.39E-04	
	Am-241	0.00E+00	N/A	U	3.55E-01	1.69E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-028	Co-60	0.00E+00	N/A	U	8.30E-02	3.33E-02	3.80E+00	0.00E+00	0.0163
	Cs-134	0.00E+00	N/A	U	1.34E-01	9.49E-02	5.70E+00	0.00E+00	
	Cs-137	1.65E-01	N/A	U	1.76E-01	8.16E-02	1.10E+01	1.50E-02	
	Eu-152	0.00E+00	N/A	U	2.91E-01	1.37E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.61E-01	1.58E-01	8.00E+00	0.00E+00	
	Eu-155	3.60E-01	N/A	U	4.06E-01	1.95E-01	2.80E+02	1.29E-03	
	Am-241	0.00E+00	N/A	U	3.23E-01	1.54E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-029	Co-60	2.96E-02	N/A	U	1.15E-01	4.90E-02	3.80E+00	7.79E-03	0.0507
	Cs-134	0.00E+00	N/A	U	1.57E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	2.33E-01	3.83E-02		1.05E-01	4.54E-02	1.10E+01	2.12E-02	
	Eu-152	1.75E-01	N/A	U	3.35E+01	1.59E-01	8.70E+00	2.01E-02	
	Eu-154	8.67E-03	N/A	U	4.28E-01	1.90E-01	8.00E+00	1.08E-03	
	Eu-155	1.58E-01	N/A	U	4.50E-01	2.16E-01	2.80E+02	5.64E-04	
	Am-241	0.00E+00	N/A	U	3.88E-01	1.85E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-030	Co-60	5.54E-02	N/A	U	1.56E-01	8.64E-02	3.80E+00	1.46E-02	0.0311
	Cs-134	0.00E+00	N/A	U	2.34E-01	1.35E-01	5.70E+00	0.00E+00	
	Cs-137	1.77E-01	4.36E-02		1.44E-01	6.33E-02	1.10E+01	1.61E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.44E-01	1.61E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.43E-01	1.90E-01	8.00E+00	0.00E+00	
	Eu-155	1.30E-01	N/A	U	3.32E-01	1.56E-01	2.80E+02	4.64E-04	
	Am-241	0.00E+00	N/A	U	4.23E-01	2.02E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-031	Co-60	2.65E-02	N/A	U	9.11E-02	5.98E-02	3.80E+00	6.97E-03	0.0181
	Cs-134	0.00E+00	N/A	U	1.13E-01	7.45E-02	5.70E+00	0.00E+00	
	Cs-137	8.57E-02	N/A	U	1.26E-01	5.74E-02	1.10E+01	7.79E-03	
	Eu-152	0.00E+00	N/A	U	2.30E-01	1.08E-01	8.70E+00	0.00E+00	
	Eu-154	2.53E-02	N/A	U	2.75E-01	1.18E-01	8.00E+00	3.16E-03	
	Eu-155	4.74E-02	N/A	U	3.09E-01	1.48E-01	2.80E+02	1.69E-04	
	Am-241	0.00E+00	N/A	U	2.70E-01	1.28E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-032	Co-60	4.31E-02	N/A	U	1.11E-01	4.84E-02	3.80E+00	1.13E-02	0.0221
	Cs-134	0.00E+00	N/A	U	1.31E-01	7.93E-02	5.70E+00	0.00E+00	
	Cs-137	1.06E-01	N/A	U	1.51E-01	6.99E-02	1.10E+01	9.64E-03	
	Eu-152	0.00E+00	N/A	U	2.85E-01	1.35E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.16E-01	1.38E-01	8.00E+00	0.00E+00	
	Eu-155	3.20E-01	N/A	U	3.94E-01	1.90E-01	2.80E+02	1.14E-03	
	Am-241	0.00E+00	N/A	U	3.40E-01	1.63E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-033	Co-60	1.58E-02	N/A	U	8.55E-02	3.66E-02	3.80E+00	4.16E-03	0.0075
	Cs-134	0.00E+00	N/A	U	9.14E-02	6.03E-02	5.70E+00	0.00E+00	
	Cs-137	1.92E-02	N/A	U	8.40E-02	3.71E-02	1.10E+01	1.75E-03	
	Eu-152	8.07E-03	N/A	U	1.98E-01	9.26E-02	8.70E+00	9.28E-04	
	Eu-154	0.00E+00	N/A	U	2.76E-01	1.21E-01	8.00E+00	0.00E+00	
	Eu-155	1.88E-01	N/A	U	2.74E-01	1.31E-01	2.80E+02	6.71E-04	
	Am-241	0.00E+00	N/A	U	2.27E-01	1.07E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-034	Co-60	1.33E-02	N/A	U	1.13E-01	6.40E-02	3.80E+00	3.50E-03	0.0051

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.28E-01	9.07E-02	5.70E+00	0.00E+00	
	Cs-137	8.36E-03	N/A	U	1.33E-01	6.02E-02	1.10E+01	7.60E-04	
	Eu-152	0.00E+00	N/A	U	2.87E-01	1.35E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.74E-01	1.15E-01	8.00E+00	0.00E+00	
	Eu-155	2.44E-01	N/A	U	4.04E-01	1.94E-01	2.80E+02	8.71E-04	
	Am-241	0.00E+00	N/A	U	3.40E-01	1.62E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-035	Co-60	0.00E+00	N/A	U	1.11E-01	5.88E-02	3.80E+00	0.00E+00	0.0165
	Cs-134	0.00E+00	N/A	U	1.50E-01	9.01E-02	5.70E+00	0.00E+00	
	Cs-137	1.75E-01	3.36E-02		9.63E-02	4.11E-02	1.10E+01	1.59E-02	
	Eu-152	0.00E+00	N/A	U	3.08E-01	1.45E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.62E-01	1.56E-01	8.00E+00	0.00E+00	
	Eu-155	1.56E-01	N/A	U	4.22E-01	2.03E-01	2.80E+02	5.57E-04	
	Am-241	0.00E+00	N/A	U	3.24E-01	1.53E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-036	Co-60	6.15E-02	N/A	U	1.11E-01	7.83E-02	3.80E+00	1.62E-02	0.0266
	Cs-134	0.00E+00	N/A	U	1.44E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	1.03E-01	N/A	U	1.76E-01	8.12E-02	1.10E+01	9.36E-03	
	Eu-152	3.82E-03	N/A	U	3.22E-01	1.52E-01	8.70E+00	4.39E-04	
	Eu-154	0.00E+00	N/A	U	3.97E-01	1.74E-01	8.00E+00	0.00E+00	
	Eu-155	1.85E-01	N/A	U	3.00E-01	1.42E-01	2.80E+02	6.61E-04	
	Am-241	0.00E+00	N/A	U	3.49E-01	1.66E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-037	Co-60	1.16E-02	N/A	U	8.49E-02	4.85E-02	3.80E+00	3.05E-03	0.0254
	Cs-134	0.00E+00	N/A	U	1.07E-01	6.75E-02	5.70E+00	0.00E+00	
	Cs-137	3.01E-02	N/A	U	1.04E-01	4.69E-02	1.10E+01	2.74E-03	
	Eu-152	3.48E-03	N/A	U	2.19E-01	1.03E-01	8.70E+00	4.00E-04	
	Eu-154	5.44E-02	N/A	U	2.36E-01	1.01E-01	8.00E+00	6.80E-03	
	Eu-155	4.89E-02	N/A	U	2.70E-01	1.29E-01	2.80E+02	1.75E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	2.56E-02	N/A	U	2.37E-01	1.12E-01	2.10E+00	1.22E-02	
8100X-3-CR-GSSX-038	Co-60	1.99E-02	N/A	U	1.73E-01	7.60E-02	3.80E+00	5.24E-03	0.0288
	Cs-134	0.00E+00	N/A	U	1.83E-01	1.22E-01	5.70E+00	0.00E+00	
	Cs-137	2.49E-01	4.51E-02		1.33E-01	5.81E-02	1.10E+01	2.26E-02	
	Eu-152	0.00E+00	N/A	U	3.93E-01	1.86E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.28E-01	1.85E-01	8.00E+00	0.00E+00	
	Eu-155	2.69E-01	N/A	U	5.48E-01	2.64E-01	2.80E+02	9.61E-04	
	Am-241	0.00E+00	N/A	U	4.60E-01	2.20E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-039	Co-60	2.02E-02	N/A	U	7.60E-02	5.20E-02	3.80E+00	5.32E-03	0.0186
	Cs-134	0.00E+00	N/A	U	9.97E-02	7.65E-02	5.70E+00	0.00E+00	
	Cs-137	5.83E-02	2.39E-02		8.95E-02	3.94E-02	1.10E+01	5.30E-03	
	Eu-152	6.77E-02	N/A	U	2.47E-01	1.17E-01	8.70E+00	7.78E-03	
	Eu-154	0.00E+00	N/A	U	2.83E-01	1.23E-01	8.00E+00	0.00E+00	
	Eu-155	5.55E-02	N/A	U	1.94E-01	9.04E-02	2.80E+02	1.98E-04	
	Am-241	0.00E+00	N/A	U	2.65E-01	1.26E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-040	Co-60	0.00E+00	N/A	U	1.08E-01	6.12E-02	3.80E+00	0.00E+00	0.0101
	Cs-134	0.00E+00	N/A	U	1.62E-01	1.17E-01	5.70E+00	0.00E+00	
	Cs-137	9.93E-02	N/A	U	1.88E-01	8..61E-2	1.10E+01	9.03E-03	
	Eu-152	0.00E+00	N/A	U	3.14E-01	1.47E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.83E-01	1.63E-01	8.00E+00	0.00E+00	
	Eu-155	3.12E-01	N/A	U	4.71E-01	2.27E-01	2.80E+02	1.11E-03	
	Am-241	0.00E+00	N/A	U	3.85E-01	1.83E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-041	Co-60	0.00E+00	N/A	U	1.12E-01	4.78E-02	3.80E+00	0.00E+00	0.0176
	Cs-134	0.00E+00	N/A	U	1.40E-01	9.53E-02	5.70E+00	0.00E+00	
	Cs-137	1.86E-01	3.53E-02		1.06E-01	4.68E-02	1.10E+01	1.69E-02	
	Eu-152	0.00E+00	N/A	U	2.97E-01	1.40E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	3.03E-01	1.29E-01	8.00E+00	0.00E+00	
	Eu-155	1.88E-01	N/A	U	4.02E-01	1.93E-01	2.80E+02	6.71E-04	
	Am-241	0.00E+00	N/A	U	3.36E-01	1.60E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSX-042	Co-60	3.39E-02	N/A	U	1.21E-01	5.14E-02	3.80E+00	8.92E-03	0.0247
	Cs-134	0.00E+00	N/A	U	1.52E-01	9.50E-02	5.70E+00	0.00E+00	
	Cs-137	1.65E-01	3.41E-02		1.02E-01	4.35E-02	1.10E+01	1.50E-02	
	Eu-152	0.00E+00	N/A	U	3.03E-01	1.42E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.47E-01	1.48E-01	8.00E+00	0.00E+00	
	Eu-155	2.14E-01	N/A	U	3.10E-01	1.46E-01	2.80E+02	7.64E-04	
	Am-241	0.00E+00	N/A	U	3.49E-01	1.65E-01	2.10E+00	0.00E+00	

Table 2-49 8100 Gamma Spectroscopy Results for QC Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8100X-3-CQ-GSSX-001	Co-60	0.00E+00	N/A	U	1.04E-01	4.30E-02	3.80E+00	0.00E+00	0.0182
	Cs-134	0.00E+00	N/A	U	1.53E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	1.07E-01	2.57E-02		7.54E-02	3.06E-02	1.10E+01	9.73E-03	
	Eu-152	0.00E+00	N/A	U	2.90E-01	1.36E-01	8.70E+00	0.00E+00	
	Eu-154	6.52E-02	N/A	U	4.37E-01	2.10E-01	8.00E+00	8.15E-03	
	Eu-155	9.81E-02	N/A	U	4.37E-01	2.10E-01	2.80E+02	3.50E-04	
	Am-241	0.00E+00	N/A	U	3.55E-01	1.69E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSB-008	Co-60	0.00E+00	N/A	U	1.18E-01	4.92E-02	3.80E+00	0.00E+00	0.0170
	Cs-134	0.00E+00	N/A	U	1.38E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.08E-01	N/A	U	1.85E-01	8.49E-02	1.10E+01	9.82E-03	
	Eu-152	0.00E+00	N/A	U	3.29E-01	1.54E-01	8.70E+00	0.00E+00	
	Eu-154	5.22E-02	N/A	U	4.17E-01	1.81E-01	8.00E+00	6.53E-03	
	Eu-155	1.96E-01	N/A	U	3.52E-01	1.67E-01	2.80E+02	7.00E-04	
	Am-241	0.00E+00	N/A	U	3.99E-01	1.90E-01	2.10E+00	0.00E+00	
8100X-3-CR-GSSB-009	Co-60	1.20E-02	N/A	U	1.21E-01	5.24E-02	3.80E+00	3.16E-03	0.0236
	Cs-134	0.00E+00	N/A	U	1.24E-01	8.90E-02	5.70E+00	0.00E+00	
	Cs-137	1.13E-01	3.43E-02		1.22E-01	5.45E-02	1.10E+01	1.03E-02	
	Eu-152	0.00E+00	N/A	U	2.86E-01	1.35E-01	8.70E+00	0.00E+00	
	Eu-154	7.07E-02	N/A	U	3.78E-01	1.67E-01	8.00E+00	8.84E-03	
	Eu-155	3.66E-01	N/A	U	4.08E-01	1.96E-01	2.80E+02	1.31E-03	
	Am-241	0.00E+00	N/A	U	3.26E-01	1.55E-01	2.10E+00	0.00E+00	
8100X-3-CQ-GSSX-013	Co-60	4.72E-02	N/A	U	1.49E-01	8.70E-02	3.80E+00	1.24E-02	0.0325
	Cs-134	0.00E+00	N/A	U	2.16E-01	1.34E-01	5.70E+00	0.00E+00	
	Cs-137	2.16E-01	4.54E-02		1.41E-01	6.18E-02	1.10E+01	1.96E-02	
	Eu-152	0.00E+00	N/A	U	3.71E-01	1.75E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	3.88E-01	1.63E-01	8.00E+00	0.00E+00	
	Eu-155	1.29E-01	N/A	U	3.35E-01	1.57E-01	2.80E+02	4.61E-04	
	Am-241	0.00E+00	N/A	U	3.99E-01	1.89E-01	2.10E+00	0.00E+00	
8100X-3-CQ-GSSX-027	Co-60	0.00E+00	N/A	U	1.04E-01	4.37E-02	3.80E+00	0.00E+00	0.0470
	Cs-134	0.00E+00	N/A	U	1.50E-01	9.06E-02	5.70E+00	0.00E+00	
	Cs-137	2.77E-01	4.00E-02		1.05E-01	4.61E-02	1.10E+01	2.52E-02	
	Eu-152	0.00E+00	N/A	U	2.88E-01	1.36E-01	8.70E+00	0.00E+00	
	Eu-154	5.37E-02	N/A	U	3.83E-01	1.69E-01	8.00E+00	6.71E-03	
	Eu-155	1.17E-01	N/A	U	4.06E-01	1.95E-01	2.80E+02	4.18E-04	
	Am-241	3.09E-02	N/A	U	3.49E-01	1.66E-01	2.10E+00	1.47E-02	
8100X-3-CQ-GSSX-040	Co-60	7.23E-02	N/A	U	1.17E-01	5.93E-02	3.80E+00	1.90E-02	0.0344
	Cs-134	0.00E+00	N/A	U	1.55E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	1.30E-01	N/A	U	1.87E-01	8.57E-02	1.10E+01	1.18E-02	
	Eu-152	0.00E+00	N/A	U	2.92E-01	1.36E-01	8.70E+00	0.00E+00	
	Eu-154	2.13E-02	N/A	U	4.37E-01	1.91E-01	8.00E+00	2.66E-03	
	Eu-155	2.62E-01	N/A	U	4.49E-01	2.23E-01	2.80E+02	9.36E-04	
	Am-241	0.00E+00	N/A	U	3.63E-01	1.71E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSB-062	Co-60	0.00E+00	N/A	U	1.34E-01	8.14E-02	3.80E+00	0.00E+00	0.0206
	Cs-134	0.00E+00	N/A	U	2.33E-01	1.23E-01	5.70E+00	0.00E+00	
	Cs-137	1.60E-01	N/A	U	2.46E-01	1.12E-01	1.10E+01	1.45E-02	
	Eu-152	4.31E-02	N/A	U	4.08E-01	1.91E-01	8.70E+00	4.95E-03	
	Eu-154	0.00E+00	N/A	U	5.09E-01	2.18E-01	8.00E+00	0.00E+00	
	Eu-155	3.05E-01	N/A	U	5.53E-01	2.64E-01	2.80E+02	1.09E-03	
	Am-241	0.00E+00	N/A	U	4.80E-01	2.28E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSB-063	Co-60	1.83E-02	N/A	U	1.18E-01	7.97E-02	3.80E+00	4.82E-03	0.0247
	Cs-134	0.00E+00	N/A	U	1.81E-01	1.01E-01	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	1.81E-01	N/A	U	2.13E-01	9.83E-02	1.10E+01	1.65E-02	
	Eu-152	2.23E-02	N/A	U	3.40E-01	1.60E-01	8.70E+00	2.56E-03	
	Eu-154	0.00E+00	N/A	U	4.50E-01	1.95E-01	8.00E+00	0.00E+00	
	Eu-155	2.53E-01	N/A	U	4.78E-01	2.29E-01	2.80E+02	9.04E-04	
	Am-241	0.00E+00	N/A	U	4.05E-01	1.93E-01	2.10E+00	0.00E+00	
8100X-3-CQ-GSSX-069	Co-60	2.07E-02	N/A	U	1.54E-01	9.26E-02	3.80E+00	5.45E-03	0.0273
	Cs-134	0.00E+00	N/A	U	2.14E-01	1.29E-01	5.70E+00	0.00E+00	
	Cs-137	2.31E-01	N/A	U	2.60E-01	1.20E-01	1.10E+01	2.10E-02	
	Eu-152	0.00E+00	N/A	U	3.90E-01	1.82E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	5.29E-01	2.29E-01	8.00E+00	0.00E+00	
	Eu-155	2.38E-01	N/A	U	5.18E-01	2.61E-01	2.80E+02	8.50E-04	
	Am-241	0.00E+00	N/A	U	4.40E-01	2.09E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSB-081	Co-60	5.80E-02	N/A	U	1.23E-01	5.15E-02	3.80E+00	1.53E-02	0.0619
	Cs-134	0.00E+00	N/A	U	1.40E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	1.91E-01	N/A	U	2.01E-01	9.29E-02	1.10E+01	1.74E-02	
	Eu-152	6.21E-02	N/A	U	3.52E-01	1.66E-01	8.70E+00	7.14E-03	
	Eu-154	1.72E-01	N/A	U	4.27E-01	1.86E-01	8.00E+00	2.15E-02	
	Eu-155	1.84E-01	N/A	U	3.85E-01	1.83E-01	2.80E+02	6.57E-04	
	Am-241	0.00E+00	N/A	U	4.10E-01	1.95E-01	2.10E+00	0.00E+00	

Table 2-50 8100 Gamma Spectroscopy Results for Judgmental Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8100X-3-CJ-GSSX-043	Co-60	1.63E-02	N/A	U	8.35E-02	3.39E-02	3.80E+00	4.29E-03	0.0190
	Cs-134	0.00E+00	N/A	U	9.54E-02	9.43E-02	5.70E+00	0.00E+00	
	Cs-137	4.29E-02	N/A	U	1.11E-01	4.94E-02	1.10E+01	3.90E-03	
	Eu-152	9.04E-02	N/A	U	2.58E-01	1.21E-01	8.70E+00	1.04E-02	
	Eu-154	0.00E+00	N/A	U	2.32E-01	9.43E-02	8.00E+00	0.00E+00	
	Eu-155	1.30E-01	N/A	U	3.48E-01	1.66E-01	2.80E+02	4.64E-04	
	Am-241	0.00E+00	N/A	U	2.76E-01	1.30E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-044	Co-60	2.06E-02	N/A	U	1.44E-01	6.23E-02	3.80E+00	5.42E-03	0.0410
	Cs-134	0.00E+00	N/A	U	1.88E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	2.15E-01	4.13E-02		1.24E-01	5.43E-02	1.10E+01	1.95E-02	
	Eu-152	0.00E+00	N/A	U	3.08E-01	1.44E-01	8.70E+00	0.00E+00	
	Eu-154	1.15E-01	N/A	U	4.27E-01	1.86E-01	8.00E+00	1.44E-02	
	Eu-155	4.55E-01	N/A	U	4.76E-01	2.42E-01	2.80E+02	1.63E-03	
	Am-241	0.00E+00	N/A	U	3.73E-01	1.77E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-045	Co-60	4.74E-02	N/A	U	1.47E-01	9.43E-02	3.80E+00	1.25E-02	0.0303
	Cs-134	0.00E+00	N/A	U	2.23E-01	1.33E-01	5.70E+00	0.00E+00	
	Cs-137	1.21E-01	N/A	U	2.41E-01	1.11E-01	1.10E+01	1.10E-02	
	Eu-152	0.00E+00	N/A	U	3.86E-01	1.81E-01	8.70E+00	0.00E+00	
	Eu-154	4.87E-02	N/A	U	4.83E-01	2.08E-01	8.00E+00	6.09E-03	
	Eu-155	2.07E-01	N/A	U	5.28E-01	2.53E-01	2.80E+02	7.39E-04	
	Am-241	0.00E+00	N/A	U	4.08E-01	1.94E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-046	Co-60	6.56E-03	N/A	U	1.48E-01	7.96E-02	3.80E+00	1.73E-03	0.0614
	Cs-134	0.00E+00	N/A	U	1.80E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	3.16E-01	N/A	U	2.80E-01	1.30E-01	1.10E+01	2.87E-02	
	Eu-152	0.00E+00	N/A	U	3.56E-01	1.66E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	2.32E-01	N/A	U	5.93E-01	2.62E-01	8.00E+00	2.90E-02	
	Eu-155	5.48E-01	N/A	U	5.58E-01	2.73E-01	2.80E+02	1.96E-03	
	Am-241	0.00E+00	N/A	U	4.28E-01	2.03E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-047	Co-60	6.32E-02	N/A	U	1.39E-01	6.76E-02	3.80E+00	1.66E-02	0.0250
	Cs-134	0.00E+00	N/A	U	1.48E-01	9.42E-02	5.70E+00	0.00E+00	
	Cs-137	8.53E-02	N/A	U	1.67E-01	7.61E-02	1.10E+01	7.75E-03	
	Eu-152	0.00E+00	N/A	U	3.02E-01	1.41E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.34E-01	1.91E-01	8.00E+00	0.00E+00	
	Eu-155	1.77E-01	N/A	U	4.14E-01	1.98E-01	2.80E+02	6.32E-04	
	Am-241	0.00E+00	N/A	U	3.52E-01	1.67E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-048	Co-60	8.24E-02	N/A	U	1.37E-01	6.54E-02	3.80E+00	2.17E-02	0.0420
	Cs-134	0.00E+00	N/A	U	1.74E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.12E-01	N/A	U	1.87E-01	8.59E-02	1.10E+01	1.02E-02	
	Eu-152	8.14E-02	N/A	U	3.46E-01	1.63E-01	8.70E+00	9.36E-03	
	Eu-154	0.00E+00	N/A	U	4.18E-01	1.82E-01	8.00E+00	0.00E+00	
	Eu-155	2.22E-01	N/A	U	4.61E-01	2.21E-01	2.80E+02	7.93E-04	
	Am-241	0.00E+00	N/A	U	3.69E-01	1.76E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-049	Co-60	3.71E-02	N/A	U	1.27E-01	7.08E-02	3.80E+00	9.76E-03	0.0309
	Cs-134	0.00E+00	N/A	U	1.55E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	7.46E-02	N/A	U	1.63E-01	7.37E-02	1.10E+01	6.78E-03	
	Eu-152	0.00E+00	N/A	U	3.31E-01	1.56E-01	8.70E+00	0.00E+00	
	Eu-154	1.15E-01	N/A	U	3.93E-01	1.69E-01	8.00E+00	1.44E-02	
	Eu-155	0.00E+00	N/A	U	4.09E-01	1.95E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	3.62E-01	1.72E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-050	Co-60	1.12E-02	N/A	U	1.26E-01	5.21E-02	3.80E+00	2.95E-03	0.0166
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.20E-01	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	1.43E-01	N/A	U	2.23E-01	1.03E-01	1.10E+01	1.30E-02	
	Eu-152	5.02E-03	N/A	U	3.50E-01	1.64E-01	8.70E+00	5.77E-04	
	Eu-154	0.00E+00	N/A	U	3.87E-01	1.63E-01	8.00E+00	0.00E+00	
	Eu-155	2.08E-02	N/A	U	4.89E-01	2.35E-01	2.80E+02	7.43E-05	
	Am-241	0.00E+00	N/A	U	4.07E-01	1.93E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-051	Co-60	1.95E-02	N/A	U	1.33E-01	7.63E-02	3.80E+00	5.13E-03	0.0652
	Cs-134	1.25E-01	N/A	U	1.79E-01	1.08E-01	5.70E+00	2.19E-02	
	Cs-137	1.68E-01	N/A	U	2.22E-01	1.02E-01	1.10E+01	1.53E-02	
	Eu-152	0.00E+00	N/A	U	3.24E-01	1.52E-01	8.70E+00	0.00E+00	
	Eu-154	1.79E-01	N/A	U	4.72E-01	2.06E-01	8.00E+00	2.24E-02	
	Eu-155	1.32E-01	N/A	U	4.00E-01	1.90E-01	2.80E+02	4.71E-04	
	Am-241	0.00E+00	N/A	U	3.81E-01	1.81E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-052	Co-60	0.00E+00	N/A	U	1.22E-01	7.75E-02	3.80E+00	0.00E+00	0.0250
	Cs-134	0.00E+00	N/A	U	2.06E-01	1.09E-01	5.70E+00	0.00E+00	
	Cs-137	2.73E-01	N/A	U	2.33E-01	1.09E-01	1.10E+01	2.48E-02	
	Eu-152	0.00E+00	N/A	U	3.42E-01	1.61E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.73E-01	1.58E-01	8.00E+00	0.00E+00	
	Eu-155	5.76E-02	N/A	U	2.79E-01	1.30E-01	2.80E+02	2.06E-04	
	Am-241	0.00E+00	N/A	U	3.68E-01	1.75E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-053	Co-60	3.27E-02	N/A	U	1.23E-01	6.75E-02	3.80E+00	8.61E-03	0.0280
	Cs-134	0.00E+00	N/A	U	1.68E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	2.05E-01	4.33E-02		1.36E-01	6.01E-02	1.10E+01	1.86E-02	
	Eu-152	0.00E+00	N/A	U	3.46E-01	1.63E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.86E-01	1.65E-01	8.00E+00	0.00E+00	
	Eu-155	2.26E-01	N/A	U	4.76E-01	2.28E-01	2.80E+02	8.07E-04	
	Am-241	0.00E+00	N/A	U	4.03E-01	1.92E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8100X-3-CJ-GSSX-054	Co-60	1.73E-03	N/A	U	1.34E-01	5.61E-02	3.80E+00	4.55E-04	0.0374
	Cs-134	0.00E+00	N/A	U	1.94E-01	1.36E-01	5.70E+00	0.00E+00	
	Cs-137	1.01E-01	N/A	U	2.00E-01	9.13E-02	1.10E+01	9.18E-03	
	Eu-152	1.29E-01	N/A	U	3.95E-01	1.87E-01	8.70E+00	1.48E-02	
	Eu-154	1.00E-01	N/A	U	5.35E-01	2.37E-01	8.00E+00	1.25E-02	
	Eu-155	1.33E-01	N/A	U	5.34E-01	2.57E-01	2.80E+02	4.75E-04	
	Am-241	0.00E+00	N/A	U	4.50E-01	2.15E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-055	Co-60	4.16E-02	N/A	U	1.49E-01	6.31E-02	3.80E+00	1.09E-02	0.0161
	Cs-134	0.00E+00	N/A	U	2.03E-01	1.28E-01	5.70E+00	0.00E+00	
	Cs-137	2.21E-02	N/A	U	1.90E-01	8.60E-02	1.10E+01	2.01E-03	
	Eu-152	1.98E-02	N/A	U	3.82E-01	1.80E-01	8.70E+00	2.28E-03	
	Eu-154	0.00E+00	N/A	U	4.37E-01	1.87E-01	8.00E+00	0.00E+00	
	Eu-155	2.44E-01	N/A	U	4.54E-01	2.17E-01	2.80E+02	8.71E-04	
	Am-241	0.00E+00	N/A	U	4.45E-01	2.12E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-056	Co-60	3.64E-02	N/A	U	1.40E-01	8.30E-02	3.80E+00	9.58E-03	0.0300
	Cs-134	0.00E+00	N/A	U	1.63E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	1.66E-01	4.19E-02		1.39E-01	6.11E-02	1.10E+01	1.51E-02	
	Eu-152	3.74E-02	N/A	U	4.22E-01	2.00E-01	8.70E+00	4.30E-03	
	Eu-154	4.43E-03	N/A	U	4.53E-01	1.96E-01	8.00E+00	5.54E-04	
	Eu-155	1.27E-01	N/A	U	5.00E-01	2.40E-01	2.80E+02	4.54E-04	
	Am-241	0.00E+00	N/A	U	4.15E-01	1.98E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-057	Co-60	0.00E+00	N/A	U	9.92E-02	6.49E-02	3.80E+00	0.00E+00	0.0027
	Cs-134	0.00E+00	N/A	U	1.77E-01	9.43E-02	5.70E+00	0.00E+00	
	Cs-137	1.81E-02	N/A	U	1.53E-01	6.82E-02	1.10E+01	1.65E-03	
	Eu-152	0.00E+00	N/A	U	3.28E-01	1.54E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.44E-01	1.43E-01	8.00E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-155	2.93E-01	N/A	U	4.54E-01	2.18E-01	2.80E+02	1.05E-03	
	Am-241	0.00E+00	N/A	U	3.74E-01	1.78E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-058	Co-60	5.09E-02	N/A	U	1.54E-01	7.49E-02	3.80E+00	1.34E-02	0.0482
	Cs-134	0.00E+00	N/A	U	1.62E-01	1.13E-01	5.70E+00	0.00E+00	
	Cs-137	2.16E-01	4.01E-02		1.13E-01	4.81E-02	1.10E+01	1.96E-02	
	Eu-152	0.00E+00	N/A	U	3.67E-01	1.73E-01	8.70E+00	0.00E+00	
	Eu-154	1.19E-01	N/A	U	4.37E-01	1.89E-01	8.00E+00	1.49E-02	
	Eu-155	9.29E-02	N/A	U	4.91E-01	2.36E-01	2.80E+02	3.32E-04	
	Am-241	0.00E+00	N/A	U	3.93E-01	1.87E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-059	Co-60	0.00E+00	N/A	U	1.13E-01	4.68E-02	3.80E+00	0.00E+00	0.0593
	Cs-134	0.00E+00	N/A	U	1.51E-01	1.00E-01	5.70E+00	0.00E+00	
	Cs-137	8.75E-02	N/A	U	1.68E-01	7.65E-02	1.10E+01	7.95E-03	
	Eu-152	0.00E+00	N/A	U	3.12E-01	1.46E-01	8.70E+00	0.00E+00	
	Eu-154	5.33E-02	N/A	U	4.55E-01	2.00E-01	8.00E+00	6.66E-03	
	Eu-155	3.20E-02	N/A	U	2.10E-01	9.58E-02	2.80E+02	1.14E-04	
	Am-241	9.36E-02	N/A	U	3.90E-01	1.86E-01	2.10E+00	4.46E-02	
8100X-3-CJ-GSSX-060	Co-60	1.63E-02	N/A	U	1.39E-01	5.91E-02	3.80E+00	4.29E-03	0.0285
	Cs-134	0.00E+00	N/A	U	1.82E-01	1.18E-01	5.70E+00	0.00E+00	
	Cs-137	2.61E-01	N/A	U	2.37E-01	1.10E-01	1.10E+01	2.37E-02	
	Eu-152	0.00E+00	N/A	U	3.75E-01	1.77E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.09E-01	1.75E-01	8.00E+00	0.00E+00	
	Eu-155	1.26E-01	N/A	U	4.90E-01	2.35E-01	2.80E+02	4.50E-04	
	Am-241	0.00E+00	N/A	U	4.24E-01	2.02E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-061	Co-60	0.00E+00	N/A	U	1.25E-01	7.27E-02	3.80E+00	0.00E+00	0.0317
	Cs-134	2.14E-02	N/A	U	2.06E-01	1.13E-01	5.70E+00	3.75E-03	
	Cs-137	1.54E-01	3.97E-02		1.33E-01	5.82E-02	1.10E+01	1.40E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.32E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.84E-01	2.13E-01	8.00E+00	0.00E+00	
	Eu-155	3.28E-01	N/A	U	4.98E-01	2.46E-01	2.80E+02	1.17E-03	
	Am-241	2.68E-02	N/A	U	4.06E-01	1.94E-01	2.10E+00	1.28E-02	
8100X-3-CJ-GSSX-062	Co-60	1.59E-02	N/A	U	1.50E-01	6.46E-02	3.80E+00	4.18E-03	0.0313
	Cs-134	0.00E+00	N/A	U	1.96E-01	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	8.23E-02	N/A	U	1.94E-01	8.90E-02	1.10E+01	7.48E-03	
	Eu-152	0.00E+00	N/A	U	3.82E-01	1.80E-01	8.70E+00	0.00E+00	
	Eu-154	1.49E-01	N/A	U	4.53E-01	1.98E-01	8.00E+00	1.86E-02	
	Eu-155	2.77E-01	N/A	U	3.72E-01	1.76E-01	2.80E+02	9.89E-04	
	Am-241	0.00E+00	N/A	U	4.33E-01	2.06E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-063	Co-60	1.45E-02	N/A	U	1.12E-01	4.53E-02	3.80E+00	3.82E-03	0.0368
	Cs-134	0.00E+00	N/A	U	1.77E-01	1.09E-01	5.70E+00	0.00E+00	
	Cs-137	1.01E-01	3.48E-02		1.25E-01	5.41E-02	1.10E+01	9.18E-03	
	Eu-152	0.00E+00	N/A	U	3.70E-01	1.75E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.50E-01	1.46E-01	8.00E+00	0.00E+00	
	Eu-155	7.85E-02	N/A	U	3.78E-01	1.80E-01	2.80E+02	2.80E-04	
	Am-241	4.95E-02	N/A	U	4.32E-01	2.06E-01	2.10E+00	2.36E-02	
8100X-3-CJ-GSSX-064	Co-60	0.00E+00	N/A	U	1.38E-01	5.92E-02	3.80E+00	0.00E+00	0.0209
	Cs-134	0.00E+00	N/A	U	2.06E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	2.16E-01	4.60E-02		1.47E-01	6.54E-02	1.10E+01	1.96E-02	
	Eu-152	0.00E+00	N/A	U	3.55E-01	1.67E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.66E-01	2.04E-01	8.00E+00	0.00E+00	
	Eu-155	3.46E-01	N/A	U	4.68E-01	2.40E-01	2.80E+02	1.24E-03	
	Am-241	0.00E+00	N/A	U	3.86E-01	1.83E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-065	Co-60	2.14E-02	N/A	U	1.09E-01	8.15E-02	3.80E+00	5.63E-03	0.0367

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	2.09E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	2.00E-01	4.10E-02		1.24E-01	5.33E-02	1.10E+01	1.82E-02	
	Eu-152	0.00E+00	N/A	U	3.31E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	9.32E-02	N/A	U	4.54E-01	1.97E-01	8.00E+00	1.17E-02	
	Eu-155	3.56E-01	N/A	U	4.93E-01	2.44E-01	2.80E+02	1.27E-03	
	Am-241	0.00E+00	N/A	U	3.84E-01	1.82E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-066	Co-60	3.91E-02	N/A	U	1.12E-01	7.24E-02	3.80E+00	1.03E-02	0.0211
	Cs-134	0.00E+00	N/A	U	1.81E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	1.60E-02	4.32E-02		1.51E-01	6.78E-02	1.10E+01	1.45E-03	
	Eu-152	7.87E-02	N/A	U	3.57E-01	1.69E-01	8.70E+00	9.05E-03	
	Eu-154	0.00E+00	N/A	U	4.26E-01	1.86E-01	8.00E+00	0.00E+00	
	Eu-155	7.36E-02	N/A	U	4.58E-01	2.21E-01	2.80E+02	2.63E-04	
	Am-241	0.00E+00	N/A	U	3.75E-01	1.78E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-067	Co-60	0.00E+00	N/A	U	1.21E-01	5.97E-02	3.80E+00	0.00E+00	0.0155
	Cs-134	0.00E+00	N/A	U	1.56E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	1.56E-01	3.50E-02		1.10E-01	4.75E-02	1.10E+01	1.42E-02	
	Eu-152	0.00E+00	N/A	U	3.36E-01	1.59E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.25E-01	1.86E-01	8.00E+00	0.00E+00	
	Eu-155	3.75E-01	N/A	U	4.10E-01	2.16E-01	2.80E+02	1.34E-03	
	Am-241	0.00E+00	N/A	U	3.54E-01	1.68E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-068	Co-60	2.94E-02	N/A	U	1.31E-01	5.57E-02	3.80E+00	7.74E-03	0.0331
	Cs-134	0.00E+00	N/A	U	1.74E-01	1.12E-01	5.70E+00	0.00E+00	
	Cs-137	1.64E-01	N/A	U	2.03E-01	9.39E-02	1.10E+01	1.49E-02	
	Eu-152	8.39E-02	N/A	U	3.43E-01	1.62E-01	8.70E+00	9.64E-03	
	Eu-154	0.00E+00	N/A	U	4.86E-01	2.15E-01	8.00E+00	0.00E+00	
	Eu-155	2.30E-01	N/A	U	3.05E-01	1.43E-01	2.80E+02	8.21E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	3.55E-01	1.68E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-069	Co-60	6.42E-02	N/A	U	1.58E-01	6.89E-02	3.80E+00	1.69E-02	0.0393
	Cs-134	0.00E+00	N/A	U	1.85E-01	1.22E-01	5.70E+00	0.00E+00	
	Cs-137	1.87E-01	4.53E-02		1.53E-01	6.88E-02	1.10E+01	1.70E-02	
	Eu-152	0.00E+00	N/A	U	3.31E-01	1.56E-01	8.70E+00	0.00E+00	
	Eu-154	3.35E-02	N/A	U	4.12E-01	1.78E-01	8.00E+00	4.19E-03	
	Eu-155	3.45E-01	N/A	U	4.84E-01	2.33E-01	2.80E+02	1.23E-03	
	Am-241	0.00E+00	N/A	U	3.76E-01	1.79E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-070	Co-60	0.00E+00	N/A	U	1.07E-01	4.39E-02	3.80E+00	0.00E+00	0.0247
	Cs-134	0.00E+00	N/A	U	1.81E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.25E-01	3.48E-02		1.18E-01	5.14E-02	1.10E+01	1.14E-02	
	Eu-152	0.00E+00	N/A	U	3.25E-01	1.53E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.68E-01	2.08E-01	8.00E+00	0.00E+00	
	Eu-155	2.54E-01	N/A	U	4.74E-01	2.30E-01	2.80E+02	9.07E-04	
	Am-241	2.61E-02	N/A	U	4.02E-01	1.92E-01	2.10E+00	1.24E-02	
8100X-3-CJ-GSSX-071	Co-60	5.80E-02	N/A	U	1.19E-01	7.64E-02	3.80E+00	1.53E-02	0.0352
	Cs-134	0.00E+00	N/A	U	1.79E-01	1.30E-01	5.70E+00	0.00E+00	
	Cs-137	2.14E-01	N/A	U	2.26E-01	1.04E-01	1.10E+01	1.95E-02	
	Eu-152	0.00E+00	N/A	U	3.26E-01	1.52E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.93E-01	1.67E-01	8.00E+00	0.00E+00	
	Eu-155	1.43E-01	N/A	U	4.78E-01	2.29E-01	2.80E+02	5.11E-04	
	Am-241	0.00E+00	N/A	U	4.29E-01	2.05E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-072	Co-60	2.86E-02	N/A	U	1.49E-01	7.15E-02	3.80E+00	7.53E-03	0.0446
	Cs-134	0.00E+00	N/A	U	2.01E-01	1.12E-01	5.70E+00	0.00E+00	
	Cs-137	1.23E-01	N/A	U	2.14E-01	9.90E-02	1.10E+01	1.12E-02	
	Eu-152	2.15E-01	N/A	U	3.59E-01	1.70E-01	8.70E+00	2.47E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	3.82E-01	1.62E-01	8.00E+00	0.00E+00	
	Eu-155	3.22E-01	N/A	U	4.43E-01	2.26E-01	2.80E+02	1.15E-03	
	Am-241	0.00E+00	N/A	U	3.87E-01	1.84E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-073	Co-60	4.68E-02	N/A	U	1.29E-01	7.21E-02	3.80E+00	1.23E-02	0.0212
	Cs-134	0.00E+00	N/A	U	1.83E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	9.75E-02	3.58E-02		1.32E-01	5.81E-02	1.10E+01	8.86E-03	
	Eu-152	0.00E+00	N/A	U	3.35E-01	1.58E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.68E-01	1.57E-01	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	4.24E-01	2.03E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	3.26E-01	1.54E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-074	Co-60	4.48E-02	N/A	U	1.09E-01	7.39E-02	3.80E+00	1.18E-02	0.0267
	Cs-134	0.00E+00	N/A	U	1.61E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	1.15E-01	N/A	U	1.68E-01	7.55E-02	1.10E+01	1.05E-02	
	Eu-152	0.00E+00	N/A	U	2.93E-01	1.36E-01	8.70E+00	0.00E+00	
	Eu-154	3.12E-02	N/A	U	4.17E-01	1.78E-01	8.00E+00	3.90E-03	
	Eu-155	1.52E-01	N/A	U	4.30E-01	2.05E-01	2.80E+02	5.43E-04	
	Am-241	0.00E+00	N/A	U	3.81E-01	1.80E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-075	Co-60	9.44E-03	N/A	U	1.12E-01	5.85E-02	3.80E+00	2.48E-03	0.0067
	Cs-134	0.00E+00	N/A	U	1.93E-01	9.91E-02	5.70E+00	0.00E+00	
	Cs-137	4.46E-02	N/A	U	1.68E-01	7.61E-02	1.10E+01	4.05E-03	
	Eu-152	0.00E+00	N/A	U	3.14E-01	1.47E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.60E-01	2.03E-01	8.00E+00	0.00E+00	
	Eu-155	5.56E-02	N/A	U	4.37E-01	2.09E-01	2.80E+02	1.99E-04	
	Am-241	0.00E+00	N/A	U	3.61E-01	1.71E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-076	Co-60	3.54E-02	N/A	U	1.28E-01	6.45E-02	3.80E+00	9.32E-03	0.0149
	Cs-134	0.00E+00	N/A	U	1.54E-01	9.99E-02	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	5.59E-02	N/A	U	1.39E-01	6.28E-02	1.10E+01	5.08E-03	
	Eu-152	0.00E+00	N/A	U	3.11E-01	1.47E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.93E-01	1.72E-01	8.00E+00	0.00E+00	
	Eu-155	1.29E-01	N/A	U	4.25E-01	2.04E-01	2.80E+02	4.61E-04	
	Am-241	0.00E+00	N/A	U	3.60E-01	1.72E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-077	Co-60	1.04E-02	N/A	U	1.13E-01	6.29E-02	3.80E+00	2.74E-03	0.0276
	Cs-134	0.00E+00	N/A	U	1.76E-01	9.56E-02	5.70E+00	0.00E+00	
	Cs-137	4.54E-02	2.75E-02		1.09E-01	4.75E-02	1.10E+01	4.13E-03	
	Eu-152	0.00E+00	N/A	U	3.05E-01	1.43E-01	8.70E+00	0.00E+00	
	Eu-154	1.55E-01	N/A	U	4.24E-01	1.87E-01	8.00E+00	1.94E-02	
	Eu-155	3.93E-01	N/A	U	4.30E-01	2.15E-01	2.80E+02	1.40E-03	
	Am-241	0.00E+00	N/A	U	3.58E-01	1.70E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-078	Co-60	3.98E-02	N/A	U	1.24E-01	5.32E-02	3.80E+00	1.05E-02	0.0179
	Cs-134	0.00E+00	N/A	U	1.62E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	4.76E-02	N/A	U	1.68E-01	7.71E-02	1.10E+01	4.33E-03	
	Eu-152	9.51E-03	N/A	U	3.37E-01	1.59E-01	8.70E+00	1.09E-03	
	Eu-154	6.15E-03	N/A	U	4.01E-01	1.76E-01	8.00E+00	7.69E-04	
	Eu-155	3.55E-01	N/A	U	4.56E-01	2.21E-01	2.80E+02	1.27E-03	
	Am-241	0.00E+00	N/A	U	3.68E-01	1.75E-01	2.10E+00	0.00E+00	
8100X-3-CJ-GSSX-079	Co-60	2.22E-02	N/A	U	1.01E-01	6.69E-02	3.80E+00	5.84E-03	0.0164
	Cs-134	0.00E+00	N/A	U	1.35E-01	1.00E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.34E-01	6.06E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.23E-01	1.53E-01	8.70E+00	0.00E+00	
	Eu-154	7.28E-02	N/A	U	4.21E-01	1.87E-01	8.00E+00	9.10E-03	
	Eu-155	4.03E-01	N/A	U	4.56E-01	2.20E-01	2.80E+02	1.44E-03	
	Am-241	0.00E+00	N/A	U	3.83E-01	1.83E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8100X-3-CJ-GSSX-080	Co-60	0.00E+00	N/A	U	1.31E-01	5.79E-02	3.80E+00	0.00E+00	0.0318
	Cs-134	0.00E+00	N/A	U	1.66E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	1.18E-01	N/A	U	1.98E-01	9.07E-02	1.10E+01	1.07E-02	
	Eu-152	0.00E+00	N/A	U	3.20E-01	1.50E-01	8.70E+00	0.00E+00	
	Eu-154	1.50E-01	N/A	U	4.29E-01	1.86E-01	8.00E+00	1.88E-02	
	Eu-155	1.99E-01	N/A	U	4.74E-01	2.27E-01	2.80E+02	7.11E-04	
	Am-241	3.39E-03	N/A	U	3.92E-01	1.87E-01	2.10E+00	1.61E-03	
8100X-3-CJ-GSSX-081	Co-60	2.69E-02	N/A	U	1.26E-01	5.37E-02	3.80E+00	7.08E-03	0.0383
	Cs-134	0.00E+00	N/A	U	1.70E-01	1.00E-01	5.70E+00	0.00E+00	
	Cs-137	1.26E-01	3.99E-02		1.44E-01	6.46E-02	1.10E+01	1.15E-02	
	Eu-152	1.53E-02	N/A	U	3.55E-01	1.68E-01	8.70E+00	1.76E-03	
	Eu-154	1.25E-01	N/A	U	4.46E-01	1.97E-01	8.00E+00	1.56E-02	
	Eu-155	6.75E-01	N/A	U	5.13E-01	2.71E-01	2.80E+02	2.41E-03	
	Am-241	0.00E+00	N/A	U	3.95E-01	1.88E-01	2.10E+00	0.00E+00	
8100X-3-CI-GSSX-13A	Co-60	7.49E-02	N/A	U	1.50E-01	6.46E-02	3.80E+00	1.97E-02	0.0565
	Cs-134	0.00E+00	N/A	U	2.05E-01	1.13E-01	5.70E+00	0.00E+00	
	Cs-137	2.28E-01	4.15E-02		1.17E-01	5.00E-02	1.10E+01	2.07E-02	
	Eu-152	0.00E+00	N/A	U	3.48E-01	1.63E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.65E-01	2.03E-01	8.00E+00	0.00E+00	
	Eu-155	3.66E-01	N/A	U	5.10E-01	2.45E-01	2.80E+02	1.31E-03	
	Am-241	3.10E-02	N/A	U	4.29E-01	2.04E-01	2.10E+00	1.48E-02	

Table 2-51 8100 Summary Statistics

Combined					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	1.15E-01	2.59E-02	2.10E-02	2.37E-02
Cs-134	0.00E+00	1.25E-01	1.81E-03	0.00E+00	1.33E-02
Cs-137	0.00E+00	3.16E-01	1.33E-01	1.26E-01	7.21E-02
Eu-152	0.00E+00	2.15E-01	2.12E-02	0.00E+00	4.25E-02
Eu-154	0.00E+00	2.49E-01	3.81E-02	0.00E+00	6.02E-02
Eu-155	0.00E+00	6.75E-01	2.13E-01	2.01E-01	1.30E-01
Am-241	0.00E+00	1.17E-01	7.35E-03	0.00E+00	2.20E-02
Random					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	1.15E-01	2.59E-02	2.26E-02	2.46E-02
Cs-134	0.00E+00	2.03E-02	4.83E-04	0.00E+00	3.13E-03
Cs-137	1.50E-05	2.49E-01	1.25E-01	1.29E-01	6.87E-02
Eu-152	0.00E+00	1.75E-01	2.51E-02	2.36E-03	4.42E-02
Eu-154	0.00E+00	2.49E-01	3.07E-02	0.00E+00	5.77E-02
Eu-155	9.03E-03	4.32E-01	1.99E-01	1.94E-01	1.13E-01
Am-241	0.00E+00	1.17E-01	9.89E-03	0.00E+00	2.71E-02
Judgmental					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	8.24E-02	2.66E-02	2.18E-02	2.26E-02
Cs-134	0.00E+00	1.25E-01	3.66E-03	0.00E+00	2.00E-02
Cs-137	0.00E+00	3.16E-01	1.31E-01	1.22E-01	7.70E-02
Eu-152	0.00E+00	2.15E-01	1.91E-02	0.00E+00	4.48E-02
Eu-154	0.00E+00	2.32E-01	4.46E-02	0.00E+00	6.46E-02
Eu-155	0.00E+00	6.75E-01	2.27E-01	2.15E-01	1.54E-01
Am-241	0.00E+00	9.36E-02	5.76E-03	0.00E+00	1.77E-02

Total Number of Samples	
Random	42
Judgmental	40
QC	10

Random	
SOF >0.5	0
Maximum SOF	0.0993
Minimum SOF	0.0029

Judgmental	
SOF >0.5	0
Maximum SOF	0.0652
Minimum SOF	0.0027

Figure 2-28 Survey Unit 8200 Random Sample and Scan Locations

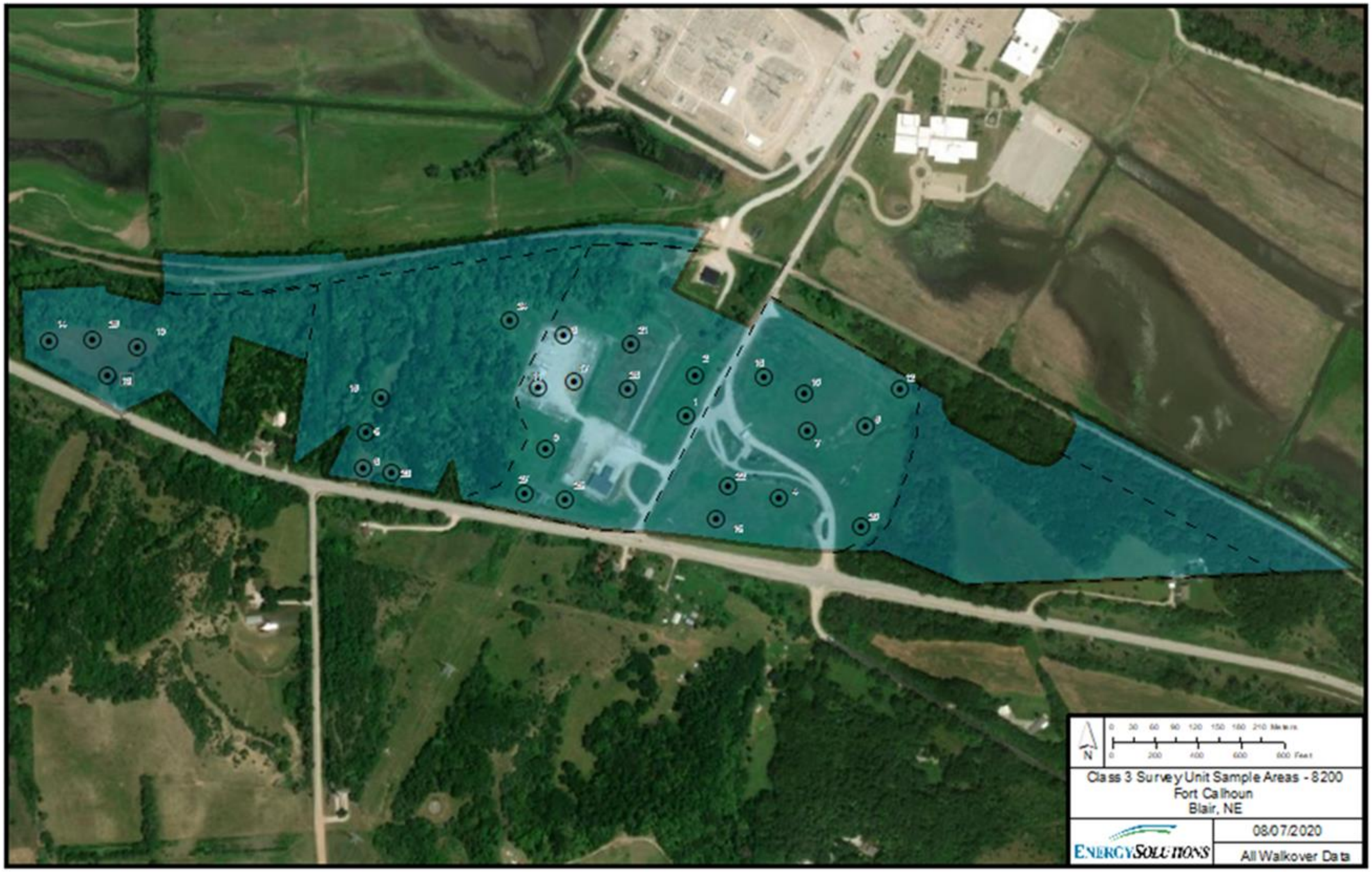


Figure 2-29 Survey Unit 8200 Judgmental Sample and Scan Locations

Table 2-52 8200 Gamma Spectroscopy Results for Random Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8200X-3-CR-GSSX-001	Co-60	0.00E+00	N/A	U	1.38E-01	7.11E-02	3.80E+00	0.00E+00	0.0126
	Cs-134	0.00E+00	N/A	U	1.56E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.04E-01	N/A	U	2.00E-01	9.16E-02	1.10E+01	9.45E-03	
	Eu-152	0.00E+00	N/A	U	3.05E-01	1.41E-01	8.70E+00	0.00E+00	
	Eu-154	1.56E-02	N/A	U	5.02E-01	2.21E-01	8.00E+00	1.95E-03	
	Eu-155	3.31E-01	N/A	U	4.66E-01	2.36E-01	2.80E+02	1.18E-03	
	Am-241	0.00E+00	N/A	U	3.90E-01	1.84E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-002	Co-60	9.14E-03	N/A	U	1.29E-01	5.44E-02	3.80E+00	2.41E-03	0.0230
	Cs-134	0.00E+00	N/A	U	1.69E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	1.35E-01	N/A	U	1.79E-01	8.16E-02	1.10E+01	1.23E-02	
	Eu-152	6.71E-02	N/A	U	3.21E-01	1.50E-01	8.70E+00	7.71E-03	
	Eu-154	0.00E+00	N/A	U	3.70E-01	1.57E-01	8.00E+00	0.00E+00	
	Eu-155	1.80E-01	N/A	U	4.60E-01	2.20E-01	2.80E+02	6.43E-04	
	Am-241	0.00E+00	N/A	U	3.87E-01	1.84E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-003 ROCK	Co-60	0.00E+00	N/A	U	9.50E-02	4.69E-02	3.80E+00	0.00E+00	0.0041
	Cs-134	0.00E+00	N/A	U	1.01E-01	1.33E-01	5.70E+00	0.00E+00	
	Cs-137	3.71E-02	N/A	U	1.28E-01	5.77E-02	1.10E+01	3.37E-03	
	Eu-152	0.00E+00	N/A	U	3.15E-01	1.49E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.18E-01	1.36E-01	8.00E+00	0.00E+00	
	Eu-155	1.90E-01	N/A	U	2.78E-01	1.31E-01	2.80E+02	6.79E-04	
	Am-241	0.00E+00	N/A	U	3.41E-01	1.62E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-004	Co-60	3.01E-02	N/A	U	1.47E-01	6.30E-02	3.80E+00	7.92E-03	0.0162
	Cs-134	0.00E+00	N/A	U	1.46E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.52E-01	6.78E-02	1.10E+01	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	6.66E-02	N/A	U	3.54E-01	1.67E-01	8.70E+00	7.66E-03	
	Eu-154	0.00E+00	N/A	U	4.09E-01	1.76E-01	8.00E+00	0.00E+00	
	Eu-155	1.80E-01	N/A	U	4.48E-01	2.25E-01	2.80E+02	6.43E-04	
	Am-241	0.00E+00	N/A	U	3.81E-01	1.81E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-005	Co-60	7.42E-02	N/A	U	1.25E-01	6.29E-02	3.80E+00	1.95E-02	0.0472
	Cs-134	0.00E+00	N/A	U	1.47E-01	9.69E-02	5.70E+00	0.00E+00	
	Cs-137	6.85E-02	N/A	U	1.60E-01	7.28E-02	1.10E+01	6.23E-03	
	Eu-152	0.00E+00	N/A	U	2.84E-01	1.33E-01	8.70E+00	0.00E+00	
	Eu-154	1.66E-01	N/A	U	4.11E-01	1.81E-01	8.00E+00	2.08E-02	
	Eu-155	2.01E-01	N/A	U	4.30E-01	2.06E-01	2.80E+02	7.18E-04	
	Am-241	0.00E+00	N/A	U	3.70E-01	1.76E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-006	Co-60	0.00E+00	N/A	U	1.17E-01	4.92E-02	3.80E+00	0.00E+00	0.0086
	Cs-134	0.00E+00	N/A	U	1.43E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	9.06E-02	N/A	U	1.79E-01	8.21E-02	1.10E+01	8.24E-03	
	Eu-152	0.00E+00	N/A	U	3.06E-01	1.44E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.94E-01	1.72E-01	8.00E+00	0.00E+00	
	Eu-155	9.62E-02	N/A	U	4.30E-01	2.06E-01	2.80E+02	3.44E-04	
	Am-241	0.00E+00	N/A	U	3.36E-01	1.59E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-007	Co-60	3.92E-02	N/A	U	1.27E-01	7.55E-02	3.80E+00	1.03E-02	0.0104
	Cs-134	0.00E+00	N/A	U	1.84E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.71E-01	7.74E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.41E-01	1.60E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.61E-01	2.02E-01	8.00E+00	0.00E+00	
	Eu-155	1.93E-02	N/A	U	4.52E-01	2.16E-01	2.80E+02	6.89E-05	
	Am-241	0.00E+00	N/A	U	3.85E-01	1.83E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-008	Co-60	2.07E-02	N/A	U	1.20E-01	5.15E-02	3.80E+00	5.45E-03	0.0311

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.34E-01	9.08E-02	5.70E+00	0.00E+00	
	Cs-137	1.20E-01	2.79E-02		8.54E-02	3.61E-02	1.10E+01	1.09E-02	
	Eu-152	0.00E+00	N/A	U	3.01E-01	1.42E-01	8.70E+00	0.00E+00	
	Eu-154	1.10E-01	N/A	U	3.78E-01	1.66E-01	8.00E+00	1.38E-02	
	Eu-155	2.67E-01	N/A	U	4.12E-01	1.98E-01	2.80E+02	9.54E-04	
	Am-241	0.00E+00	N/A	U	3.31E-01	1.57E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-009	Co-60	0.00E+00	N/A	U	1.36E-01	5.68E-02	3.80E+00	0.00E+00	0.0100
	Cs-134	0.00E+00	N/A	U	1.63E-01	1.17E-01	5.70E+00	0.00E+00	
	Cs-137	9.80E-02	N/A	U	2.00E-01	9.10E-02	1.10E+01	8.91E-03	
	Eu-152	0.00E+00	N/A	U	3.28E-01	1.53E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.81E-01	2.09E-01	8.00E+00	0.00E+00	
	Eu-155	2.98E-01	N/A	U	4.73E-01	2.26E-01	2.80E+02	1.06E-03	
	Am-241	0.00E+00	N/A	U	3.91E-01	1.85E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-010	Co-60	3.31E-02	N/A	U	1.03E-01	6.24E-02	3.80E+00	8.71E-03	0.0305
	Cs-134	0.00E+00	N/A	U	1.58E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	6.73E-02	N/A	U	1.43E-01	6.46E-02	1.10E+01	6.12E-03	
	Eu-152	1.32E-01	N/A	U	3.29E-01	1.55E-01	8.70E+00	1.52E-02	
	Eu-154	0.00E+00	N/A	U	3.78E-01	1.64E-01	8.00E+00	0.00E+00	
	Eu-155	1.35E-01	N/A	U	2.83E-01	1.33E-01	2.80E+02	4.82E-04	
	Am-241	0.00E+00	N/A	U	3.28E-01	1.55E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-011 ROCK	Co-60	0.00E+00	N/A	U	8.92E-02	3.66E-02	3.80E+00	0.00E+00	0.0241
	Cs-134	0.00E+00	N/A	U	1.05E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.13E-01	4.94E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.77E-01	1.29E-01	8.70E+00	0.00E+00	
	Eu-154	1.88E-01	N/A	U	3.33E-01	1.42E-01	8.00E+00	2.35E-02	
	Eu-155	1.56E-01	N/A	U	3.48E-01	1.66E-01	2.80E+02	5.57E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	2.67E-01	1.25E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-012	Co-60	0.00E+00	N/A	U	1.36E-01	6.19E-02	3.80E+00	0.00E+00	0.0178
	Cs-134	0.00E+00	N/A	U	1.73E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	1.28E-01	N/A	U	1.91E-01	8.74E-02	1.10E+01	1.16E-02	
	Eu-152	5.21E-02	N/A	U	3.52E-01	1.70E-01	8.70E+00	5.99E-03	
	Eu-154	0.00E+00	N/A	U	5.05E-01	2.24E-01	8.00E+00	0.00E+00	
	Eu-155	5.10E-02	N/A	U	3.14E-01	1.47E-01	2.80E+02	1.82E-04	
	Am-241	0.00E+00	N/A	U	3.79E-01	1.80E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-013	Co-60	6.76E-02	N/A	U	1.33E-01	6.42E-02	3.80E+00	1.78E-02	0.0477
	Cs-134	0.00E+00	N/A	U	1.66E-01	8.92E-02	5.70E+00	0.00E+00	
	Cs-137	8.24E-02	2.94E-02		1.06E-01	4.62E-02	1.10E+01	7.49E-03	
	Eu-152	1.29E-02	N/A	U	2.98E-01	1.40E-01	8.70E+00	1.48E-03	
	Eu-154	1.62E-01	N/A	U	3.99E-01	1.75E-01	8.00E+00	2.03E-02	
	Eu-155	2.03E-01	N/A	U	4.10E-01	1.96E-01	2.80E+02	7.25E-04	
	Am-241	0.00E+00	N/A	U	3.56E-01	1.69E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSA-014	Co-60	1.36E-02	N/A	U	1.40E-01	6.07E-02	3.80E+00	3.58E-03	0.0148
	Cs-134	0.00E+00	N/A	U	1.64E-01	1.01E-01	5.70E+00	0.00E+00	
	Cs-137	1.10E-01	3.42E-02		1.21E-01	5.31E-02	1.10E+01	1.00E-02	
	Eu-152	0.00E+00	N/A	U	3.26E-01	1.54E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.89E-01	1.69E-01	8.00E+00	0.00E+00	
	Eu-155	3.39E-01	N/A	U	4.54E-01	2.18E-01	2.80E+02	1.21E-03	
	Am-241	0.00E+00	N/A	U	3.82E-01	1.82E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-015	Co-60	3.50E-02	N/A	U	1.48E-01	7.41E-02	3.80E+00	9.21E-03	0.0348
	Cs-134	0.00E+00	N/A	U	1.93E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	5.74E-02	N/A	U	1.87E-01	8.48E-02	1.10E+01	5.22E-03	
	Eu-152	1.70E-01	N/A	U	3.69E-01	1.73E-01	8.70E+00	1.95E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	4.67E-01	2.02E-01	8.00E+00	0.00E+00	
	Eu-155	2.44E-01	N/A	U	3.36E-01	1.58E-01	2.80E+02	8.71E-04	
	Am-241	0.00E+00	N/A	U	4.01E-01	1.90E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-016	Co-60	5.19E-02	N/A	U	1.41E-01	7.10E-02	3.80E+00	1.37E-02	0.0355
	Cs-134	0.00E+00	N/A	U	1.60E-01	1.09E-01	5.70E+00	0.00E+00	
	Cs-137	1.40E-01	N/A	U	2.06E-01	9.54E-02	1.10E+01	1.27E-02	
	Eu-152	0.00E+00	N/A	U	3.11E-01	1.46E-01	8.70E+00	0.00E+00	
	Eu-154	6.20E-02	N/A	U	3.91E-01	1.69E-01	8.00E+00	7.75E-03	
	Eu-155	3.76E-01	N/A	U	4.56E-01	2.25E-01	2.80E+02	1.34E-03	
	Am-241	0.00E+00	N/A	U	3.46E-01	1.63E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-017	Co-60	4.21E-02	N/A	U	1.22E-01	7.08E-02	3.80E+00	1.11E-02	0.0430
	Cs-134	0.00E+00	N/A	U	1.53E-01	1.13E-01	5.70E+00	0.00E+00	
	Cs-137	4.17E-02	N/A	U	1.42E-01	6.45E-02	1.10E+01	3.79E-03	
	Eu-152	0.00E+00	N/A	U	3.14E-01	1.48E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.71E-01	1.62E-01	8.00E+00	0.00E+00	
	Eu-155	2.12E-01	N/A	U	3.12E-01	1.48E-01	2.80E+02	7.57E-04	
	Am-241	5.74E-02	N/A	U	3.77E-01	1.80E-01	2.10E+00	2.73E-02	
8200X-3-CR-GSSX-018	Co-60	2.17E-02	N/A	U	1.12E-01	4.64E-02	3.80E+00	5.71E-03	0.0456
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.13E-01	5.70E+00	0.00E+00	
	Cs-137	9.51E-02	N/A	U	1.74E-01	7.94E-02	1.10E+01	8.65E-03	
	Eu-152	1.53E-02	N/A	U	3.34E-01	1.57E-01	8.70E+00	1.76E-03	
	Eu-154	0.00E+00	N/A	U	3.76E-01	1.61E-01	8.00E+00	0.00E+00	
	Eu-155	4.25E-01	N/A	U	4.63E-01	2.22E-01	2.80E+02	1.52E-03	
	Am-241	5.88E-02	N/A	U	3.82E-01	1.81E-01	2.10E+00	2.80E-02	
8200X-3-CR-GSSX-019	Co-60	1.23E-03	N/A	U	1.16E-01	4.96E-02	3.80E+00	3.24E-04	0.0181
	Cs-134	0.00E+00	N/A	U	1.43E-01	9.40E-02	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	1.31E-01	3.25E-02		1.06E-01	4.64E-02	1.10E+01	1.19E-02	
	Eu-152	4.26E-02	N/A	U	3.22E-01	1.52E-01	8.70E+00	4.90E-03	
	Eu-154	0.00E+00	N/A	U	3.49E-01	1.51E-01	8.00E+00	0.00E+00	
	Eu-155	2.84E-01	N/A	U	4.24E-01	2.04E-01	2.80E+02	1.01E-03	
	Am-241	0.00E+00	N/A	U	3.66E-01	1.74E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-020	Co-60	2.39E-02	N/A	U	1.25E-01	5.32E-02	3.80E+00	6.29E-03	0.0201
	Cs-134	0.00E+00	N/A	U	1.29E-01	9.09E-02	5.70E+00	0.00E+00	
	Cs-137	2.49E-02	N/A	U	1.45E-01	6.51E-02	1.10E+01	2.26E-03	
	Eu-152	0.00E+00	N/A	U	2.78E-01	1.30E-01	8.70E+00	0.00E+00	
	Eu-154	8.41E-02	N/A	U	4.15E-01	1.82E-01	8.00E+00	1.05E-02	
	Eu-155	2.82E-01	N/A	U	4.02E-01	2.02E-01	2.80E+02	1.01E-03	
	Am-241	0.00E+00	N/A	U	3.51E-01	1.66E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-021	Co-60	4.91E-03	N/A	U	1.09E-01	4.51E-02	3.80E+00	1.29E-03	0.0594
	Cs-134	0.00E+00	N/A	U	1.83E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	4.12E-02	N/A	U	1.44E-01	6.48E-02	1.10E+01	3.75E-03	
	Eu-152	0.00E+00	N/A	U	2.78E-01	1.30E-01	8.70E+00	0.00E+00	
	Eu-154	8.37E-02	N/A	U	3.83E-01	1.65E-01	8.00E+00	1.05E-02	
	Eu-155	1.12E-01	N/A	U	1.99E-01	9.03E-02	2.80E+02	4.00E-04	
	Am-241	9.14E-02	N/A	U	4.02E-01	1.92E-01	2.10E+00	4.35E-02	
8200X-3-CR-GSSX-022	Co-60	6.58E-02	N/A	U	1.46E-01	6.27E-02	3.80E+00	1.73E-02	0.0365
	Cs-134	0.00E+00	N/A	U	1.32E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	8.84E-02	N/A	U	1.76E-01	7.95E-02	1.10E+01	8.04E-03	
	Eu-152	0.00E+00	N/A	U	3.15E-01	1.47E-01	8.70E+00	0.00E+00	
	Eu-154	8.71E-02	N/A	U	4.18E-01	1.80E-01	8.00E+00	1.09E-02	
	Eu-155	7.94E-02	N/A	U	4.51E-01	2.16E-01	2.80E+02	2.84E-04	
	Am-241	0.00E+00	N/A	U	3.70E-01	1.75E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8200X-3-CR-GSSX-023	Co-60	3.37E-02	N/A	U	1.29E-01	5.51E-02	3.80E+00	8.87E-03	0.0305
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	1.51E-01	3.49E-02		1.11E-01	4.82E-02	1.10E+01	1.37E-02	
	Eu-152	0.00E+00	N/A	U	3.24E-01	1.53E-01	8.70E+00	0.00E+00	
	Eu-154	6.17E-02	N/A	U	3.49E-01	1.49E-01	8.00E+00	7.71E-03	
	Eu-155	4.16E-02	N/A	U	2.09E-01	9.54E-02	2.80E+02	1.49E-04	
	Am-241	0.00E+00	N/A	U	3.53E-01	1.68E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-024	Co-60	3.36E-02	N/A	U	1.41E-01	5.94E-02	3.80E+00	8.84E-03	0.0511
	Cs-134	0.00E+00	N/A	U	1.76E-01	1.19E-01	5.70E+00	0.00E+00	
	Cs-137	1.34E-01	3.58E-02		1.17E-01	4.99E-02	1.10E+01	1.22E-02	
	Eu-152	0.00E+00	N/A	U	3.36E-01	1.57E-01	8.70E+00	0.00E+00	
	Eu-154	1.55E-01	N/A	U	5.46E-01	2.42E-01	8.00E+00	1.94E-02	
	Eu-155	1.59E-01	N/A	U	4.73E-01	2.26E-01	2.80E+02	5.68E-04	
	Am-241	2.12E-02	N/A	U	4.15E-01	1.98E-01	2.10E+00	1.01E-02	
8200X-3-CR-GSSX-025	Co-60	4.19E-02	N/A	U	1.52E-01	6.62E-02	3.80E+00	1.10E-02	0.0204
	Cs-134	0.00E+00	N/A	U	1.77E-01	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	4.94E-02	N/A	U	1.62E-01	7.31E-02	1.10E+01	4.49E-03	
	Eu-152	0.00E+00	N/A	U	3.59E-01	1.69E-01	8.70E+00	0.00E+00	
	Eu-154	3.83E-02	N/A	U	4.16E-01	1.80E-01	8.00E+00	4.79E-03	
	Eu-155	3.51E-02	N/A	U	2.23E-01	1.02E-01	2.80E+02	1.25E-04	
	Am-241	0.00E+00	N/A	U	3.91E-01	1.86E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-026	Co-60	0.00E+00	N/A	U	1.06E-01	4.45E-02	3.80E+00	0.00E+00	0.0347
	Cs-134	0.00E+00	N/A	U	1.32E-01	9.54E-02	5.70E+00	0.00E+00	
	Cs-137	2.11E-01	3.52E-02		9.42E-02	4.03E-02	1.10E+01	1.92E-02	
	Eu-152	0.00E+00	N/A	U	2.85E-01	1.34E-01	8.70E+00	0.00E+00	
	Eu-154	1.12E-01	N/A	U	3.84E-01	1.68E-01	8.00E+00	1.40E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-155	4.39E-01	N/A	U	4.15E-01	2.11E-01	2.80E+02	1.57E-03	
	Am-241	0.00E+00	N/A	U	3.21E-01	1.52E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-027	Co-60	7.17E-02	N/A	U	1.47E-01	8.22E-02	3.80E+00	1.89E-02	0.0307
	Cs-134	0.00E+00	N/A	U	1.82E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	8.64E-02	3.55E-02		1.32E-01	5.77E-02	1.10E+01	7.85E-03	
	Eu-152	0.00E+00	N/A	U	3.29E-01	1.54E-01	8.70E+00	0.00E+00	
	Eu-154	2.42E-02	N/A	U	4.86E-01	2.13E-01	8.00E+00	3.03E-03	
	Eu-155	2.66E-01	N/A	U	3.91E-01	1.86E-01	2.80E+02	9.50E-04	
	Am-241	0.00E+00	N/A	U	3.60E-01	1.70E-01	2.10E+00	0.00E+00	
8200X-3-CR-GSSX-028	Co-60	2.29E-03	N/A	U	7.97E-02	3.11E-02	3.80E+00	6.03E-04	0.0267
	Cs-134	0.00E+00	N/A	U	1.34E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.30E-01	5.78E-02	1.10E+01	0.00E+00	
	Eu-152	8.68E-02	N/A	U	3.14E-01	1.48E-01	8.70E+00	9.98E-03	
	Eu-154	1.82E-02	N/A	U	3.57E-01	1.54E-01	8.00E+00	2.28E-03	
	Eu-155	4.97E-02	N/A	U	3.95E-01	1.89E-01	2.80E+02	1.78E-04	
	Am-241	2.88E-02	N/A	U	3.58E-01	1.70E-01	2.10E+00	1.37E-02	

Table 2-53 8200 Gamma Spectroscopy Results for QC Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8200X-3-CR-GSSB-014	Co-60	0.00E+00	N/A	U	9.52E-02	5.50E-02	3.80E+00	0.00E+00	0.0135
	Cs-134	0.00E+00	N/A	U	1.45E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	1.47E-01	N/A	U	1.90E-01	8.77E-02	1.10E+01	1.34E-02	
	Eu-152	0.00E+00	N/A	U	3.13E-01	1.47E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.80E-01	1.64E-01	8.00E+00	0.00E+00	
	Eu-155	4.35E-02	N/A	U	4.54E-01	2.18E-01	2.80E+02	1.55E-04	
	Am-241	0.00E+00	N/A	U	3.73E-01	1.77E-01	2.10E+00	0.00E+00	
8200X-3-CQ-GSSX-017	Co-60	1.74E-02	N/A	U	1.29E-01	5.59E-02	3.80E+00	4.58E-03	0.0217
	Cs-134	0.00E+00	N/A	U	1.32E-01	1.09E-01	5.70E+00	0.00E+00	
	Cs-137	2.91E-02	N/A	U	1.39E-01	6.26E-02	1.10E+01	2.65E-03	
	Eu-152	8.62E-02	N/A	U	3.63E-01	1.72E-01	8.70E+00	9.91E-03	
	Eu-154	3.03E-02	N/A	U	3.69E-01	1.60E-01	8.00E+00	3.79E-03	
	Eu-155	2.06E-01	N/A	U	4.77E-01	2.30E-01	2.80E+02	7.36E-04	
	Am-241	0.00E+00	N/A	U	3.97E-01	1.89E-01	2.10E+00	0.00E+00	
8200X-3-CQ-GSSX-021	Co-60	9.07E-02	N/A	U	1.57E-01	6.76E-02	3.80E+00	2.39E-02	0.0677
	Cs-134	0.00E+00	N/A	U	1.76E-01	1.20E-01	5.70E+00	0.00E+00	
	Cs-137	2.44E-01	4.56E-02		1.34E-01	5.83E-02	1.10E+01	2.22E-02	
	Eu-152	0.00E+00	N/A	U	3.56E-01	1.67E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.48E-01	1.93E-01	8.00E+00	0.00E+00	
	Eu-155	3.11E-01	N/A	U	5.10E-01	2.53E-01	2.80E+02	1.11E-03	
	Am-241	4.32E-02	N/A	U	4.47E-01	2.13E-01	2.10E+00	2.06E-02	
8200X-3-CJ-GSSB-030	Co-60	3.10E-02	N/A	U	1.09E-01	4.56E-02	3.80E+00	8.16E-03	0.0379
	Cs-134	0.00E+00	N/A	U	1.55E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	1.58E-01	3.64E-02		1.17E-01	5.15E-02	1.10E+01	1.44E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.29E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	1.17E-01	N/A	U	3.86E-01	1.68E-01	8.00E+00	1.46E-02	
	Eu-155	2.09E-01	N/A	U	4.72E-01	2.27E-01	2.80E+02	7.46E-04	
	Am-241	0.00E+00	N/A	U	3.59E-01	1.71E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSB-034	Co-60	3.51E-02	N/A	U	1.23E-01	5.44E-02	3.80E+00	9.24E-03	0.0249
	Cs-134	0.00E+00	N/A	U	1.38E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	1.60E-01	3.72E-02		1.22E-01	5.41E-02	1.10E+01	1.45E-02	
	Eu-152	0.00E+00	N/A	U	3.30E-01	1.56E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.66E-01	1.59E-01	8.00E+00	0.00E+00	
	Eu-155	3.07E-01	N/A	U	4.47E-01	2.15E-01	2.80E+02	1.10E-03	
	Am-241	0.00E+00	N/A	U	3.68E-01	1.75E-01	2.10E+00	0.00E+00	
8200X-3-CQ-GSSX-048	Co-60	0.00E+00	N/A	U	1.33E-01	6.67E-02	3.80E+00	0.00E+00	0.0184
	Cs-134	0.00E+00	N/A	U	1.74E-01	1.02E-01	5.70E+00	0.00E+00	
	Cs-137	1.17E-01	N/A	U	1.64E-01	7.45E-02	1.10E+01	1.06E-02	
	Eu-152	0.00E+00	N/A	U	3.30E-01	1.56E-01	8.70E+00	0.00E+00	
	Eu-154	5.24E-02	N/A	U	4.36E-01	1.92E-01	8.00E+00	6.55E-03	
	Eu-155	3.27E-01	N/A	U	4.73E-01	2.28E-01	2.80E+02	1.17E-03	
	Am-241	0.00E+00	N/A	U	3.69E-01	1.76E-01	2.10E+00	0.00E+00	

Table 2-54 8200 Gamma Spectroscopy Results for Judgmental Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8200X-3-CJ-GSSX-029	Co-60	1.43E-02	N/A	U	1.25E-01	5.56E-02	3.80E+00	3.76E-03	0.0209
	Cs-134	0.00E+00	N/A	U	1.47E-01	9.83E-02	5.70E+00	0.00E+00	
	Cs-137	1.87E-01	3.98E-02		1.29E-01	5.74E-02	1.10E+01	1.70E-02	
	Eu-152	0.00E+00	N/A	U	3.14E-01	1.48E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.80E-01	1.66E-01	8.00E+00	0.00E+00	
	Eu-155	4.76E-02	N/A	U	3.21E-01	1.52E-01	2.80E+02	1.70E-04	
	Am-241	0.00E+00	N/A	U	3.65E-01	1.74E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSA-030	Co-60	0.00E+00	N/A	U	1.12E-01	5.09E-02	3.80E+00	0.00E+00	0.0183
	Cs-134	8.06E-03	N/A	U	1.37E-01	1.01E-01	5.70E+00	1.41E-03	
	Cs-137	1.39E-01	N/A	U	1.87E-01	8.70E-02	1.10E+01	1.26E-02	
	Eu-152	3.54E-02	N/A	U	3.02E-01	1.42E-01	8.70E+00	4.07E-03	
	Eu-154	0.00E+00	N/A	U	3.42E-01	1.48E-01	8.00E+00	0.00E+00	
	Eu-155	6.36E-02	N/A	U	3.21E-01	1.52E-01	2.80E+02	2.27E-04	
	Am-241	0.00E+00	N/A	U	3.73E-01	1.78E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-031	Co-60	0.00E+00	N/A	U	1.10E-01	6.26E-02	3.80E+00	0.00E+00	0.0142
	Cs-134	0.00E+00	N/A	U	1.39E-01	9.97E-02	5.70E+00	0.00E+00	
	Cs-137	1.39E-01	N/A	U	1.78E-01	8.23E-02	1.10E+01	1.26E-02	
	Eu-152	0.00E+00	N/A	U	3.08E-01	1.45E-01	8.70E+00	0.00E+00	
	Eu-154	9.88E-03	N/A	U	3.82E-01	1.68E-01	8.00E+00	1.24E-03	
	Eu-155	7.87E-02	N/A	U	4.48E-01	2.16E-01	2.80E+02	2.81E-04	
	Am-241	0.00E+00	N/A	U	3.71E-01	1.77E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-032	Co-60	1.33E-02	N/A	U	1.24E-01	5.26E-02	3.80E+00	3.50E-03	0.0232
	Cs-134	0.00E+00	N/A	U	1.57E-01	1.09E-01	5.70E+00	0.00E+00	
	Cs-137	2.03E-01	N/A	U	2.16E-01	1.01E-01	1.10E+01	1.85E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.38E-01	1.60E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.64E-01	1.56E-01	8.00E+00	0.00E+00	
	Eu-155	3.37E-01	N/A	U	4.72E-01	2.33E-01	2.80E+02	1.20E-03	
	Am-241	0.00E+00	N/A	U	4.04E-01	1.93E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-033	Co-60	6.34E-02	N/A	U	1.33E-01	7.06E-02	3.80E+00	1.67E-02	0.0487
	Cs-134	0.00E+00	N/A	U	1.52E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.82E-01	3.62E-02		1.08E-01	4.67E-02	1.10E+01	1.65E-02	
	Eu-152	4.05E-02	N/A	U	3.62E-01	1.72E-01	8.70E+00	4.66E-03	
	Eu-154	7.89E-02	N/A	U	4.79E-01	2.14E-01	8.00E+00	9.86E-03	
	Eu-155	2.59E-01	N/A	U	4.85E-01	2.33E-01	2.80E+02	9.25E-04	
	Am-241	0.00E+00	N/A	U	4.30E-01	2.06E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSA-034	Co-60	0.00E+00	N/A	U	1.20E-01	6.47E-02	3.80E+00	0.00E+00	0.0447
	Cs-134	0.00E+00	N/A	U	1.80E-01	1.20E-01	5.70E+00	0.00E+00	
	Cs-137	2.23E-01	N/A	U	2.30E-01	1.07E-01	1.10E+01	2.03E-02	
	Eu-152	7.64E-02	N/A	U	3.77E-01	1.78E-01	8.70E+00	8.78E-03	
	Eu-154	0.00E+00	N/A	U	4.07E-01	1.76E-01	8.00E+00	0.00E+00	
	Eu-155	2.27E-01	N/A	U	3.75E-01	1.78E-01	2.80E+02	8.11E-04	
	Am-241	3.12E-02	N/A	U	4.50E-01	2.15E-01	2.10E+00	1.49E-02	
8200X-3-CJ-GSSX-035	Co-60	0.00E+00	N/A	U	1.17E-01	6.99E-02	3.80E+00	0.00E+00	0.0591
	Cs-134	0.00E+00	N/A	U	1.60E-01	1.12E-01	5.70E+00	0.00E+00	
	Cs-137	1.79E-01	3.74E-02		1.17E-01	5.13E-02	1.10E+01	1.63E-02	
	Eu-152	2.10E-01	N/A	U	3.59E-01	1.84E-01	8.70E+00	2.41E-02	
	Eu-154	0.00E+00	N/A	U	4.07E-01	1.79E-01	8.00E+00	0.00E+00	
	Eu-155	2.90E-01	N/A	U	4.78E-01	2.47E-01	2.80E+02	1.04E-03	
	Am-241	3.71E-02	N/A	U	4.56E-01	2.19E-01	2.10E+00	1.77E-02	
8200X-3-CJ-GSSX-036	Co-60	0.00E+00	N/A	U	1.26E-01	5.63E-02	3.80E+00	0.00E+00	0.0136

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.48E-01	1.18E-01	5.70E+00	0.00E+00	
	Cs-137	1.37E-01	3.19E-02		9.72E-02	4.08E-02	1.10E+01	1.25E-02	
	Eu-152	0.00E+00	N/A	U	3.80E-01	1.80E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.09E-01	1.77E-01	8.00E+00	0.00E+00	
	Eu-155	3.23E-01	N/A	U	4.10E-01	1.95E-01	2.80E+02	1.15E-03	
	Am-241	0.00E+00	N/A	U	4.34E-01	2.07E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-037	Co-60	0.00E+00	N/A	U	1.28E-01	5.56E-02	3.80E+00	0.00E+00	0.0298
	Cs-134	0.00E+00	N/A	U	1.70E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.46E-01	3.68E-02		1.22E-01	5.37E-02	1.10E+01	1.33E-02	
	Eu-152	1.87E-02	N/A	U	3.54E-01	1.67E-01	8.70E+00	2.15E-03	
	Eu-154	4.24E-02	N/A	U	4.18E-01	1.82E-01	8.00E+00	5.30E-03	
	Eu-155	6.38E-02	N/A	U	3.22E-01	1.52E-01	2.80E+02	2.28E-04	
	Am-241	1.86E-02	N/A	U	4.31E-01	2.06E-01	2.10E+00	8.86E-03	
8200X-3-CJ-GSSX-038	Co-60	1.01E-02	N/A	U	1.34E-01	5.75E-02	3.80E+00	2.66E-03	0.0151
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	1.28E-01	N/A	U	1.97E-01	9.06E-02	1.10E+01	1.16E-02	
	Eu-152	0.00E+00	N/A	U	3.59E-01	1.69E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.34E-01	1.90E-01	8.00E+00	0.00E+00	
	Eu-155	2.25E-01	N/A	U	3.22E-01	1.52E-01	2.80E+02	8.04E-04	
	Am-241	0.00E+00	N/A	U	4.26E-01	2.03E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-039	Co-60	1.95E-02	N/A	U	1.10E-01	5.70E-02	3.80E+00	5.13E-03	0.0305
	Cs-134	0.00E+00	N/A	U	1.66E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	1.54E-01	N/A	U	1.78E-01	8.12E-02	1.10E+01	1.40E-02	
	Eu-152	0.00E+00	N/A	U	3.35E-01	1.58E-01	8.70E+00	0.00E+00	
	Eu-154	8.14E-02	N/A	U	4.63E-01	2.05E-01	8.00E+00	1.02E-02	
	Eu-155	3.28E-01	N/A	U	4.89E-01	2.35E-01	2.80E+02	1.17E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	3.94E-01	1.87E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-040	Co-60	6.80E-03	N/A	U	1.16E-01	4.93E-02	3.80E+00	1.79E-03	0.0246
	Cs-134	0.00E+00	N/A	U	1.53E-01	9.99E-02	5.70E+00	0.00E+00	
	Cs-137	2.25E-01	N/A	U	1.96E-01	9.11E-02	1.10E+01	2.05E-02	
	Eu-152	1.27E-02	N/A	U	3.42E-01	1.62E-01	8.70E+00	1.46E-03	
	Eu-154	3.94E-03	N/A	U	4.11E-01	1.81E-01	8.00E+00	4.93E-04	
	Eu-155	1.04E-01	N/A	U	4.42E-01	2.13E-01	2.80E+02	3.71E-04	
	Am-241	0.00E+00	N/A	U	3.80E-01	1.81E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-041	Co-60	9.15E-04	N/A	U	1.33E-01	6.48E-02	3.80E+00	2.41E-04	0.0128
	Cs-134	0.00E+00	N/A	U	1.55E-01	1.13E-01	5.70E+00	0.00E+00	
	Cs-137	1.33E-01	3.36E-02		1.10E-01	4.79E-02	1.10E+01	1.21E-02	
	Eu-152	0.00E+00	N/A	U	3.34E-01	1.58E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.69E-01	1.59E-01	8.00E+00	0.00E+00	
	Eu-155	1.30E-01	N/A	U	4.68E-01	2.25E-01	2.80E+02	4.64E-04	
	Am-241	0.00E+00	N/A	U	4.06E-01	1.94E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-042	Co-60	2.90E-02	N/A	U	1.32E-01	5.67E-02	3.80E+00	7.63E-03	0.0370
	Cs-134	0.00E+00	N/A	U	1.47E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	1.58E-01	N/A	U	1.95E-01	9.03E-02	1.10E+01	1.44E-02	
	Eu-152	0.00E+00	N/A	U	3.44E-01	1.63E-01	8.70E+00	0.00E+00	
	Eu-154	1.09E-01	N/A	U	4.38E-01	1.93E-01	8.00E+00	1.36E-02	
	Eu-155	3.99E-01	N/A	U	4.66E-01	2.24E-01	2.80E+02	1.43E-03	
	Am-241	0.00E+00	N/A	U	3.64E-01	1.73E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-043	Co-60	4.95E-02	N/A	U	1.23E-01	5.36E-02	3.80E+00	1.30E-02	0.0277
	Cs-134	0.00E+00	N/A	U	1.34E-01	9.87E-02	5.70E+00	0.00E+00	
	Cs-137	4.12E-02	N/A	U	1.49E-01	6.83E-02	1.10E+01	3.75E-03	
	Eu-152	0.00E+00	N/A	U	3.36E-01	1.60E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	3.50E-01	1.53E-01	8.00E+00	0.00E+00	
	Eu-155	2.34E-01	N/A	U	4.26E-01	2.05E-01	2.80E+02	8.36E-04	
	Am-241	2.11E-02	N/A	U	3.75E-01	1.79E-01	2.10E+00	1.00E-02	
8200X-3-CJ-GSSX-044	Co-60	4.21E-03	N/A	U	1.10E-01	4.67E-02	3.80E+00	1.11E-03	0.0546
	Cs-134	1.36E-02	N/A	U	1.40E-01	9.87E-02	5.70E+00	2.39E-03	
	Cs-137	2.15E-02	N/A	U	1.31E-01	5.90E-02	1.10E+01	1.95E-03	
	Eu-152	1.39E-01	N/A	U	3.38E-01	1.61E-01	8.70E+00	1.60E-02	
	Eu-154	4.06E-02	N/A	U	3.77E-01	1.66E-01	8.00E+00	5.08E-03	
	Eu-155	2.08E-01	N/A	U	4.36E-01	2.10E-01	2.80E+02	7.43E-04	
	Am-241	5.75E-02	N/A	U	3.93E-01	1.88E-01	2.10E+00	2.74E-02	
8200X-3-CJ-GSSX-045	Co-60	7.74E-03	N/A	U	1.06E-01	4.46E-02	3.80E+00	2.04E-03	0.0024
	Cs-134	0.00E+00	N/A	U	1.61E-01	1.01E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.32E-01	5.94E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.27E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.36E-01	1.45E-01	8.00E+00	0.00E+00	
	Eu-155	1.13E-01	N/A	U	4.60E-01	2.22E-01	2.80E+02	4.04E-04	
	Am-241	0.00E+00	N/A	U	3.61E-01	1.72E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-046	Co-60	2.61E-03	N/A	U	1.32E-01	5.77E-02	3.80E+00	6.87E-04	0.0174
	Cs-134	0.00E+00	N/A	U	1.41E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	9.11E-02	N/A	U	1.56E-01	7.10E-02	1.10E+01	8.28E-03	
	Eu-152	0.00E+00	N/A	U	3.21E-01	1.52E-01	8.70E+00	0.00E+00	
	Eu-154	6.00E-02	N/A	U	4.56E-01	2.03E-01	8.00E+00	7.50E-03	
	Eu-155	2.54E-01	N/A	U	4.77E-01	2.30E-01	2.80E+02	9.07E-04	
	Am-241	0.00E+00	N/A	U	3.81E-01	1.82E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-047	Co-60	2.50E-02	N/A	U	1.13E-01	6.24E-02	3.80E+00	6.58E-03	0.0346
	Cs-134	0.00E+00	N/A	U	1.53E-01	1.02E-01	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	2.66E-02	N/A	U	1.51E-01	6.86E-02	1.10E+01	2.42E-03	
	Eu-152	1.24E-01	N/A	U	3.57E-01	1.70E-01	8.70E+00	1.43E-02	
	Eu-154	8.67E-02	N/A	U	4.33E-01	1.93E-01	8.00E+00	1.08E-02	
	Eu-155	1.30E-01	N/A	U	4.48E-01	2.15E-01	2.80E+02	4.64E-04	
	Am-241	0.00E+00	N/A	U	4.08E-01	1.95E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-048	Co-60	7.32E-02	N/A	U	1.24E-01	7.06E-02	3.80E+00	1.93E-02	0.0247
	Cs-134	0.00E+00	N/A	U	1.48E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	5.29E-02	N/A	U	1.62E-01	7.35E-02	1.10E+01	4.81E-03	
	Eu-152	0.00E+00	N/A	U	3.16E-01	1.48E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.08E-01	1.78E-01	8.00E+00	0.00E+00	
	Eu-155	1.65E-01	N/A	U	4.52E-01	2.17E-01	2.80E+02	5.89E-04	
	Am-241	0.00E+00	N/A	U	3.46E-01	1.64E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-049	Co-60	1.07E-02	N/A	U	1.03E-01	6.20E-02	3.80E+00	2.82E-03	0.0146
	Cs-134	0.00E+00	N/A	U	1.48E-01	1.01E-01	5.70E+00	0.00E+00	
	Cs-137	1.26E-01	N/A	U	1.72E-01	7.90E-02	1.10E+01	1.15E-02	
	Eu-152	0.00E+00	N/A	U	3.17E-01	1.50E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.84E-01	1.68E-01	8.00E+00	0.00E+00	
	Eu-155	8.87E-02	N/A	U	3.35E-01	1.59E-01	2.80E+02	3.17E-04	
	Am-241	0.00E+00	N/A	U	3.64E-01	1.74E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-050	Co-60	4.87E-02	N/A	U	1.03E-01	7.47E-02	3.80E+00	1.28E-02	0.0350
	Cs-134	1.12E-02	N/A	U	1.77E-01	1.18E-01	5.70E+00	1.96E-03	
	Cs-137	4.19E-02	N/A	U	1.68E-01	7.55E-02	1.10E+01	3.81E-03	
	Eu-152	0.00E+00	N/A	U	3.18E-01	1.49E-01	8.70E+00	0.00E+00	
	Eu-154	1.24E-01	N/A	U	4.44E-01	1.92E-01	8.00E+00	1.55E-02	
	Eu-155	2.54E-01	N/A	U	4.67E-01	2.24E-01	2.80E+02	9.07E-04	
	Am-241	0.00E+00	N/A	U	3.59E-01	1.70E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8200X-3-CJ-GSSX-051	Co-60	6.16E-02	N/A	U	1.41E-01	8.38E-02	3.80E+00	1.62E-02	0.0396
	Cs-134	0.00E+00	N/A	U	1.75E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.35E-01	N/A	U	1.92E-01	8.72E-02	1.10E+01	1.23E-02	
	Eu-152	3.13E-02	N/A	U	3.40E-01	1.59E-01	8.70E+00	3.60E-03	
	Eu-154	5.71E-02	N/A	U	4.47E-01	1.93E-01	8.00E+00	7.14E-03	
	Eu-155	9.43E-02	N/A	U	4.58E-01	2.21E-01	2.80E+02	3.37E-04	
	Am-241	0.00E+00	N/A	U	3.52E-01	1.66E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-052	Co-60	2.98E-02	N/A	U	1.33E-01	5.56E-02	3.80E+00	7.84E-03	0.0172
	Cs-134	0.00E+00	N/A	U	1.76E-01	1.28E-01	5.70E+00	0.00E+00	
	Cs-137	9.25E-02	N/A	U	1.80E-01	8.14E-02	1.10E+01	8.41E-03	
	Eu-152	0.00E+00	N/A	U	3.41E-01	1.60E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	5.08E-01	2.23E-01	8.00E+00	0.00E+00	
	Eu-155	2.52E-01	N/A	U	4.60E-01	2.20E-01	2.80E+02	9.00E-04	
	Am-241	0.00E+00	N/A	U	3.86E-01	1.83E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-053	Co-60	1.52E-02	N/A	U	1.27E-01	6.53E-02	3.80E+00	4.00E-03	0.0380
	Cs-134	0.00E+00	N/A	U	1.59E-01	1.12E-01	5.70E+00	0.00E+00	
	Cs-137	5.30E-02	N/A	U	1.51E-01	6.70E-02	1.10E+01	4.82E-03	
	Eu-152	4.09E-02	N/A	U	3.17E-01	1.48E-01	8.70E+00	4.70E-03	
	Eu-154	1.93E-01	N/A	U	4.85E-01	2.13E-01	8.00E+00	2.41E-02	
	Eu-155	9.59E-02	N/A	U	3.11E-01	1.46E-01	2.80E+02	3.43E-04	
	Am-241	0.00E+00	N/A	U	3.78E-01	1.79E-01	2.10E+00	0.00E+00	
8200X-3-CJ-GSSX-054	Co-60	1.12E-02	N/A	U	1.31E-01	5.60E-02	3.80E+00	2.95E-03	0.0162
	Cs-134	0.00E+00	N/A	U	1.63E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	6.07E-02	N/A	U	1.61E-01	7.30E-02	1.10E+01	5.52E-03	
	Eu-152	0.00E+00	N/A	U	3.16E-01	1.48E-01	8.70E+00	0.00E+00	
	Eu-154	5.98E-02	N/A	U	3.90E-01	1.69E-01	8.00E+00	7.48E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-155	5.98E-02	N/A	U	3.90E-01	1.69E-01	2.80E+02	2.14E-04	
	Am-241	0.00E+00	N/A	U	3.82E-01	1.82E-01	2.10E+00	0.00E+00	

Table 2-55 8200 Summary Statistics

Combined					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	9.07E-02	2.31E-02	1.48E-02	2.43E-02
Cs-134	0.00E+00	1.36E-02	5.48E-04	0.00E+00	2.46E-03
Cs-137	0.00E+00	2.44E-01	1.04E-01	1.07E-01	6.27E-02
Eu-152	0.00E+00	2.10E-01	2.43E-02	0.00E+00	4.70E-02
Eu-154	0.00E+00	1.93E-01	4.19E-02	6.91E-03	5.53E-02
Eu-155	1.93E-02	4.39E-01	1.98E-01	2.05E-01	1.10E-01
Am-241	0.00E+00	9.14E-02	7.77E-03	0.00E+00	1.89E-02
Random					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	7.42E-02	2.56E-02	2.28E-02	2.45E-02
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	0.00E+00	2.11E-01	8.19E-02	8.74E-02	5.29E-02
Eu-152	0.00E+00	1.70E-01	2.31E-02	0.00E+00	4.42E-02
Eu-154	0.00E+00	1.88E-01	4.89E-02	1.69E-02	6.17E-02
Eu-155	1.93E-02	4.39E-01	2.02E-01	1.96E-01	1.19E-01
Am-241	0.00E+00	9.14E-02	9.20E-03	0.00E+00	2.28E-02
Judgmental					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	7.32E-02	1.91E-02	1.10E-02	2.22E-02
Cs-134	0.00E+00	1.36E-02	1.26E-03	0.00E+00	3.65E-03
Cs-137	0.00E+00	2.25E-01	1.18E-01	1.34E-01	6.46E-02
Eu-152	0.00E+00	2.10E-01	2.80E-02	0.00E+00	5.30E-02
Eu-154	0.00E+00	1.93E-01	3.64E-02	1.97E-03	5.08E-02
Eu-155	4.76E-02	3.99E-01	1.86E-01	1.87E-01	1.03E-01
Am-241	0.00E+00	5.75E-02	6.37E-03	0.00E+00	1.47E-02

Total Number of Samples	
Random	28
Judgmental	26
QC	6

Random	
SOF >0.5	0
Maximum SOF	0.0594
Minimum SOF	0.0041

Judgmental	
SOF >0.5	0
Maximum SOF	0.0591
Minimum SOF	0.0024

Figure 2-30 Survey Unit 8300 Random Sample and Scan Locations

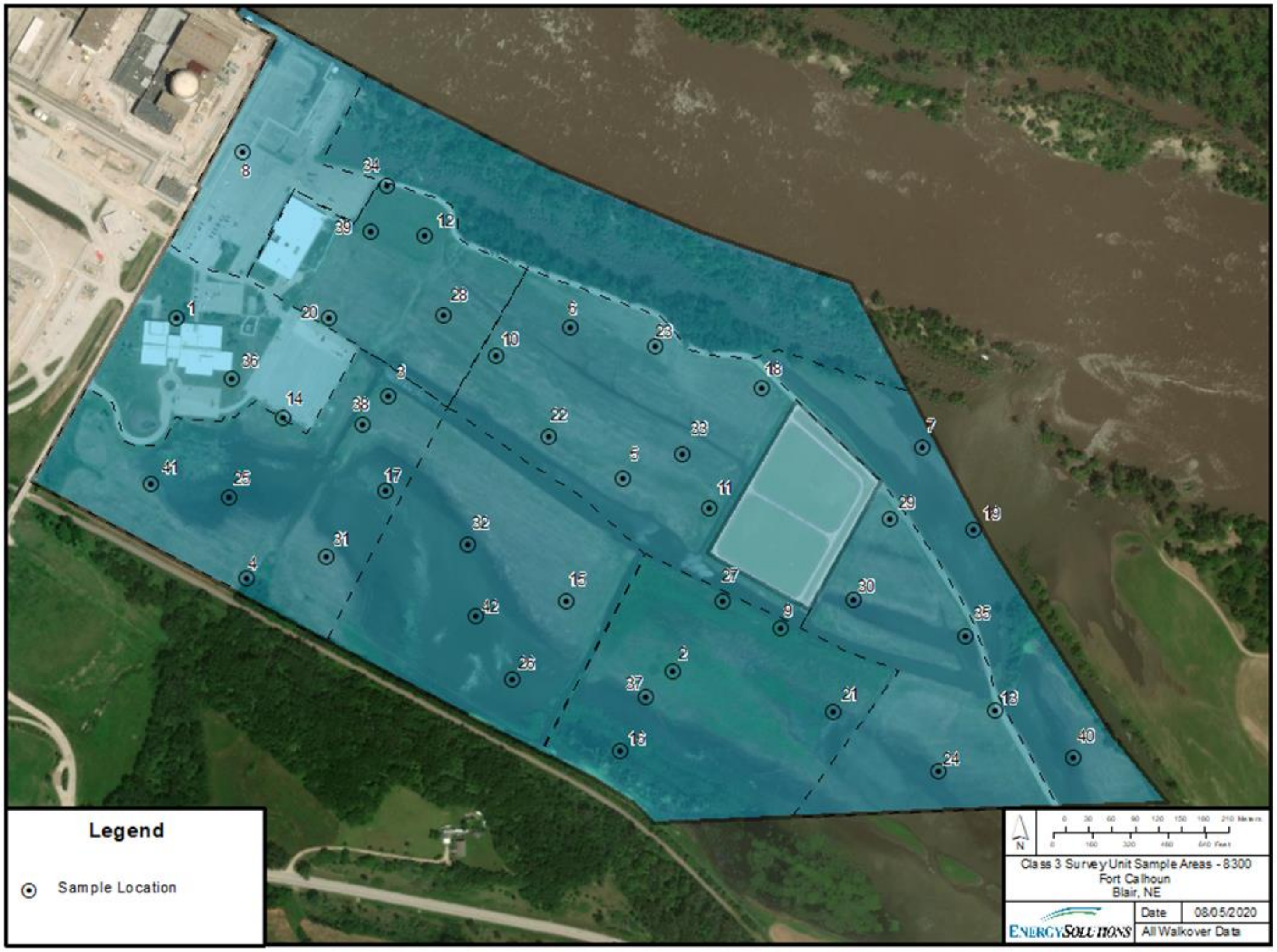


Figure 2-31 Survey Unit 8300 Judgmental Sample and Scan Locations

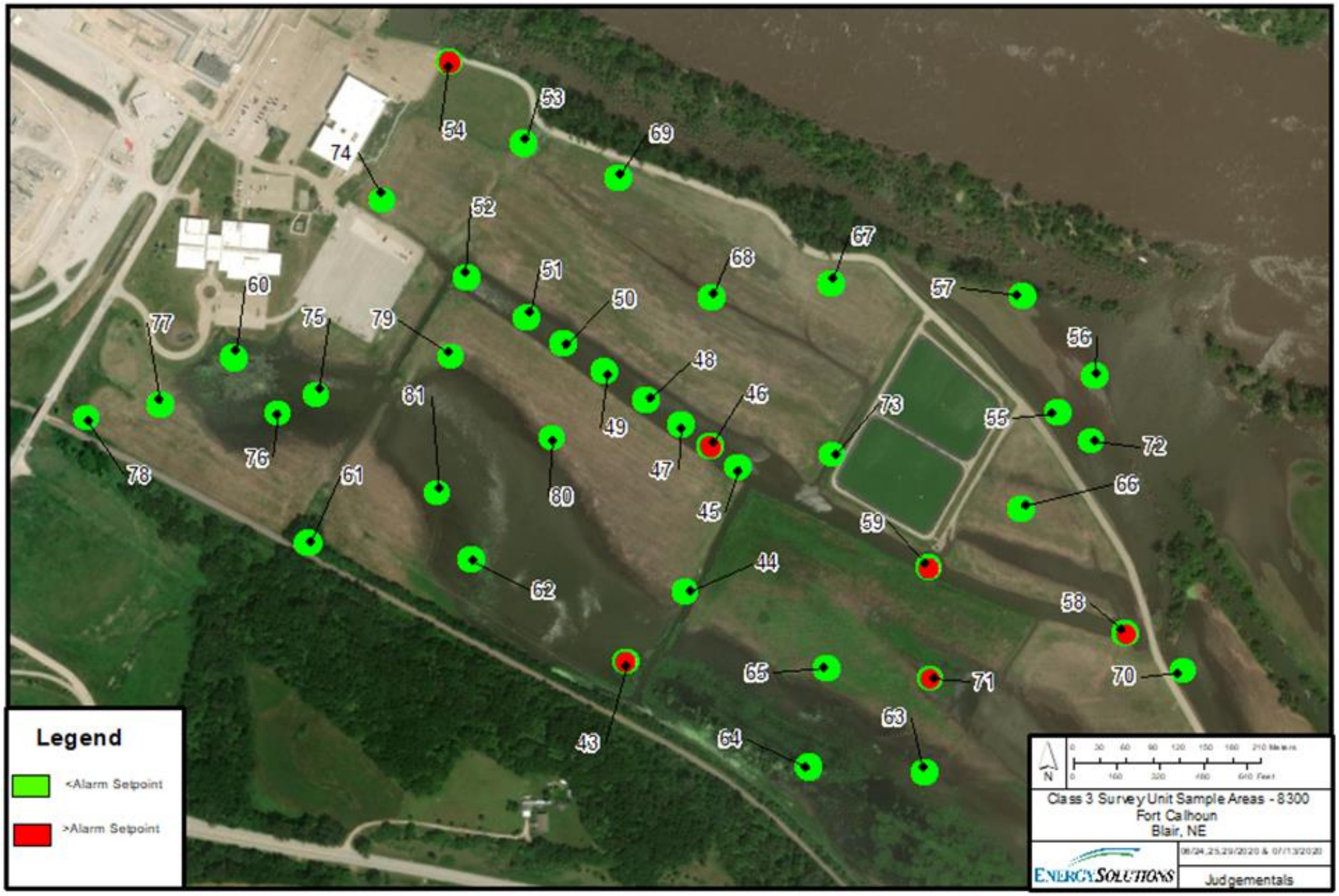


Table 2-56 8300 Gamma Spectroscopy Results for Random Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8300X-3-CR-GSSX-001	Co-60	0.00E+00	N/A	U	1.02E-01	4.13E-02	3.80E+00	0.00E+00	0.0165
	Cs-134	0.00E+00	N/A	U	1.38E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	7.74E-02	N/A	U	1.52E-01	6.82E-02	1.10E+01	7.04E-03	
	Eu-152	1.76E-02	N/A	U	3.09E-01	1.45E-01	8.70E+00	2.02E-03	
	Eu-154	5.35E-02	N/A	U	3.79E-01	1.63E-01	8.00E+00	6.69E-03	
	Eu-155	2.02E-01	N/A	U	3.04E-01	1.43E-01	2.80E+02	7.21E-04	
	Am-241	0.00E+00	N/A	U	3.42E-01	1.61E-01	-1.94E+03	0.00E+00	
8300X-3-CR-GSSX-002	Co-60	6.16E-03	N/A	U	1.05E-01	6.14E-02	3.80E+00	1.62E-03	0.0344
	Cs-134	0.00E+00	N/A	U	1.38E-01	9.88E-02	5.70E+00	0.00E+00	
	Cs-137	1.76E-01	3.43E-02		1.04E-01	4.53E-02	1.10E+01	1.60E-02	
	Eu-152	0.00E+00	N/A	U	3.01E-01	1.42E-01	8.70E+00	0.00E+00	
	Eu-154	1.25E-01	N/A	U	4.24E-01	1.89E-01	8.00E+00	1.56E-02	
	Eu-155	3.11E-01	N/A	U	4.24E-01	2.04E-01	2.80E+02	1.11E-03	
	Am-241	0.00E+00	N/A	U	3.50E-01	1.67E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-003	Co-60	1.27E-02	N/A	U	1.01E-01	4.22E-02	3.80E+00	3.34E-03	0.0241
	Cs-134	0.00E+00	N/A	U	1.31E-01	9.29E-02	5.70E+00	0.00E+00	
	Cs-137	7.87E-02	2.71E-02		9.72E-02	4.22E-02	1.10E+01	7.15E-03	
	Eu-152	1.43E-02	N/A	U	2.99E-01	1.41E-01	8.70E+00	1.64E-03	
	Eu-154	8.85E-02	N/A	U	3.72E-01	1.63E-01	8.00E+00	1.11E-02	
	Eu-155	2.65E-01	N/A	U	3.82E-01	1.83E-01	2.80E+02	9.46E-04	
	Am-241	0.00E+00	N/A	U	3.10E-01	1.47E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-004	Co-60	5.93E-02	N/A	U	1.28E-01	6.37E-02	3.80E+00	1.56E-02	0.0351
	Cs-134	0.00E+00	N/A	U	1.38E-01	9.75E-02	5.70E+00	0.00E+00	
	Cs-137	1.27E-01	3.07E-02		9.76E-02	4.19E-02	1.10E+01	1.15E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	6.37E-02	N/A	U	3.22E-01	1.52E-01	8.70E+00	7.32E-03	
	Eu-154	0.00E+00	N/A	U	3.95E-01	1.73E-01	8.00E+00	0.00E+00	
	Eu-155	1.87E-01	N/A	U	2.50E-01	1.17E-01	2.80E+02	6.68E-04	
	Am-241	0.00E+00	N/A	U	3.52E-01	1.67E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-005	Co-60	2.76E-02	N/A	U	1.05E-01	6.03E-02	3.80E+00	7.26E-03	0.0288
	Cs-134	0.00E+00	N/A	U	1.32E-01	9.58E-02	5.70E+00	0.00E+00	
	Cs-137	7.88E-02	2.29E-02		7.45E-02	3.08E-02	1.10E+01	7.16E-03	
	Eu-152	1.15E-01	N/A	U	3.12E-01	1.48E-01	8.70E+00	1.32E-02	
	Eu-154	0.00E+00	N/A	U	4.17E-01	1.86E-01	8.00E+00	0.00E+00	
	Eu-155	3.33E-01	N/A	U	4.19E-01	2.03E-01	2.80E+02	1.19E-03	
	Am-241	0.00E+00	N/A	U	3.13E-01	1.48E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-006	Co-60	2.62E-01	N/A	U	1.29E-01	5.54E-02	3.80E+00	6.89E-02	0.0735
	Cs-134	0.00E+00	N/A	U	1.51E-01	1.00E-01	5.70E+00	0.00E+00	
	Cs-137	1.08E-02	2.23E-02		9.85E-02	4.19E-02	1.10E+01	9.82E-04	
	Eu-152	2.95E-02	N/A	U	3.25E-01	1.53E-01	8.70E+00	3.39E-03	
	Eu-154	0.00E+00	N/A	U	3.86E-01	1.67E-01	8.00E+00	0.00E+00	
	Eu-155	5.67E-02	N/A	U	2.73E-01	1.28E-01	2.80E+02	2.03E-04	
	Am-241	0.00E+00	N/A	U	3.67E-01	1.74E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-007	Co-60	4.88E-02	N/A	U	1.15E-01	6.67E-02	3.80E+00	1.28E-02	0.0282
	Cs-134	0.00E+00	N/A	U	1.44E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	9.93E-02	N/A	U	1.70E-01	7.82E-02	1.10E+01	9.03E-03	
	Eu-152	3.56E-02	N/A	U	3.18E-01	1.50E-01	8.70E+00	4.09E-03	
	Eu-154	1.27E-02	N/A	U	3.71E-01	1.61E-01	8.00E+00	1.59E-03	
	Eu-155	1.91E-01	N/A	U	2.68E-01	1.26E-01	2.80E+02	6.82E-04	
	Am-241	0.00E+00	N/A	U	3.49E-01	1.66E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-008	Co-60	4.66E-02	N/A	U	1.16E-01	6.51E-02	3.80E+00	1.23E-02	0.0130

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.25E-01	7.08E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.10E-01	4.87E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.71E-01	1.27E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.51E-01	1.53E-01	8.00E+00	0.00E+00	
	Eu-155	2.03E-01	N/A	U	3.66E-01	1.75E-01	2.80E+02	7.25E-04	
	Am-241	0.00E+00	N/A	U	3.01E-01	1.42E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-009	Co-60	1.43E-02	N/A	U	1.26E-01	6.09E-02	3.80E+00	3.76E-03	0.0127
	Cs-134	0.00E+00	N/A	U	1.59E-01	9.86E-02	5.70E+00	0.00E+00	
	Cs-137	9.20E-02	2.80E-02		9.60E-02	4.10E-02	1.10E+01	8.36E-03	
	Eu-152	0.00E+00	N/A	U	2.99E-01	1.41E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.89E-01	1.70E-01	8.00E+00	0.00E+00	
	Eu-155	1.56E-01	N/A	U	3.38E-01	1.61E-01	2.80E+02	5.57E-04	
	Am-241	0.00E+00	N/A	U	3.54E-01	1.68E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-010	Co-60	5.01E-02	N/A	U	1.04E-01	5.42E-02	3.80E+00	1.32E-02	0.0393
	Cs-134	0.00E+00	N/A	U	1.32E-01	8.85E-02	5.70E+00	0.00E+00	
	Cs-137	1.35E-01	3.06E-02		9.74E-02	4.24E-02	1.10E+01	1.23E-02	
	Eu-152	1.13E-01	N/A	U	2.86E-01	1.35E-01	8.70E+00	1.30E-02	
	Eu-154	0.00E+00	N/A	U	3.28E-01	1.42E-01	8.00E+00	0.00E+00	
	Eu-155	2.44E-01	N/A	U	2.62E-01	1.23E-01	2.80E+02	8.71E-04	
	Am-241	0.00E+00	N/A	U	3.05E-01	1.45E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-011	Co-60	6.74E-03	N/A	U	1.00E-01	4.16E-02	3.80E+00	1.77E-03	0.0079
	Cs-134	0.00E+00	N/A	U	1.28E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	5.55E-02	2.32E-02		8.53E-02	3.58E-02	1.10E+01	5.05E-03	
	Eu-152	0.00E+00	N/A	U	2.69E-01	1.26E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.44E-01	1.48E-01	8.00E+00	0.00E+00	
	Eu-155	2.92E-01	N/A	U	4.17E-01	2.01E-01	2.80E+02	1.04E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	3.52E-01	1.68E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-012	Co-60	0.00E+00	N/A	U	1.39E-01	7.19E-02	3.80E+00	0.00E+00	0.0222
	Cs-134	0.00E+00	N/A	U	1.68E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	5.23E-02	N/A	U	1.71E-01	7.77E-02	1.10E+01	4.75E-03	
	Eu-152	6.67E-02	N/A	U	3.55E-01	1.68E-01	8.70E+00	7.67E-03	
	Eu-154	7.05E-02	N/A	U	4.45E-01	1.95E-01	8.00E+00	8.81E-03	
	Eu-155	2.73E-01	N/A	U	2.89E-01	1.35E-01	2.80E+02	9.75E-04	
	Am-241	0.00E+00	N/A	U	3.57E-01	1.69E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-013	Co-60	7.71E-02	N/A	U	1.34E-01	6.65E-02	3.80E+00	2.03E-02	0.0620
	Cs-134	0.00E+00	N/A	U	1.29E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.53E-01	3.32E-02		1.04E-01	4.52E-02	1.10E+01	1.39E-02	
	Eu-152	0.00E+00	N/A	U	3.01E-01	1.42E-01	8.70E+00	0.00E+00	
	Eu-154	2.20E-01	N/A	U	4.14E-01	1.83E-01	8.00E+00	2.75E-02	
	Eu-155	9.48E-02	N/A	U	3.02E-01	1.43E-01	2.80E+02	3.39E-04	
	Am-241	0.00E+00	N/A	U	3.73E-01	1.78E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-014	Co-60	4.97E-02	N/A	U	9.41E-02	5.78E-02	3.80E+00	1.31E-02	0.0190
	Cs-134	0.00E+00	N/A	U	1.35E-01	1.01E-01	5.70E+00	0.00E+00	
	Cs-137	5.19E-02	N/A	U	1.35E-01	6.11E-02	1.10E+01	4.72E-03	
	Eu-152	0.00E+00	N/A	U	2.79E-01	1.31E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.45E-01	1.51E-01	8.00E+00	0.00E+00	
	Eu-155	3.29E-01	N/A	U	3.99E-01	1.92E-01	2.80E+02	1.18E-03	
	Am-241	0.00E+00	N/A	U	3.47E-01	1.66E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-015	Co-60	2.48E-03	N/A	U	1.10E-01	6.65E-02	3.80E+00	6.53E-04	0.0222
	Cs-134	0.00E+00	N/A	U	1.77E-01	1.34E-01	5.70E+00	0.00E+00	
	Cs-137	1.30E-01	4.01E-02		1.42E-01	6.29E-02	1.10E+01	1.18E-02	
	Eu-152	7.70E-02	N/A	U	3.87E-01	1.83E-01	8.70E+00	8.85E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	4.62E-01	2.03E-01	8.00E+00	0.00E+00	
	Eu-155	2.47E-01	N/A	U	5.08E-01	2.55E-01	2.80E+02	8.82E-04	
	Am-241	0.00E+00	N/A	U	4.04E-01	1.92E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-016	Co-60	4.88E-02	N/A	U	1.27E-01	7.44E-02	3.80E+00	1.28E-02	0.0479
	Cs-134	0.00E+00	N/A	U	1.74E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	2.38E-01	4.29E-02		1.27E-01	5.57E-02	1.10E+01	2.16E-02	
	Eu-152	0.00E+00	N/A	U	3.57E-01	1.69E-01	8.70E+00	0.00E+00	
	Eu-154	9.93E-02	N/A	U	4.77E-01	2.12E-01	8.00E+00	1.24E-02	
	Eu-155	2.95E-01	N/A	U	2.90E-01	1.37E-01	2.80E+02	1.05E-03	
	Am-241	0.00E+00	N/A	U	3.85E-01	1.83E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-017	Co-60	5.88E-02	N/A	U	1.39E-01	7.98E-02	3.80E+00	1.55E-02	0.0325
	Cs-134	0.00E+00	N/A	U	1.85E-01	1.25E-01	5.70E+00	0.00E+00	
	Cs-137	1.20E-01	4.08E-02		1.46E-01	6.40E-02	1.10E+01	1.09E-02	
	Eu-152	0.00E+00	N/A	U	3.66E-01	1.71E-01	8.70E+00	0.00E+00	
	Eu-154	4.50E-02	N/A	U	5.41E-01	2.38E-01	8.00E+00	5.63E-03	
	Eu-155	1.35E-01	N/A	U	4.05E-01	1.91E-01	2.80E+02	4.82E-04	
	Am-241	0.00E+00	N/A	U	4.22E-01	1.99E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-018	Co-60	2.04E-02	N/A	U	1.21E-01	5.16E-02	3.80E+00	5.37E-03	0.0160
	Cs-134	0.00E+00	N/A	U	1.62E-01	1.16E-01	5.70E+00	0.00E+00	
	Cs-137	1.06E-01	2.88E-02		9.53E-02	4.08E-01	1.10E+01	9.64E-03	
	Eu-152	0.00E+00	N/A	U	3.04E-01	1.43E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.68E-01	1.60E-01	8.00E+00	0.00E+00	
	Eu-155	2.71E-01	N/A	U	2.58E-01	1.20E-01	2.80E+02	9.68E-04	
	Am-241	0.00E+00	N/A	U	3.52E-01	1.67E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-019	Co-60	5.61E-02	N/A	U	1.19E-01	6.74E-02	3.80E+00	1.48E-02	0.0299
	Cs-134	0.00E+00	N/A	U	1.54E-01	1.07E-01	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	9.01E-02	2.21E-02		8.73E-02	3.60E-02	1.10E+01	8.19E-03	
	Eu-152	0.00E+00	N/A	U	3.30E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	4.95E-02	N/A	U	4.25E-01	1.85E-01	8.00E+00	6.19E-03	
	Eu-155	1.99E-01	N/A	U	2.69E-01	1.25E-01	2.80E+02	7.11E-04	
	Am-241	0.00E+00	N/A	U	3.86E-01	1.84E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-020	Co-60	0.00E+00	N/A	U	1.16E-01	5.47E-02	3.80E+00	0.00E+00	0.0338
	Cs-134	0.00E+00	N/A	U	1.32E-01	9.65E-02	5.70E+00	0.00E+00	
	Cs-137	4.77E-02	2.20E-02		8.23E-02	3.39E-02	1.10E+01	4.34E-03	
	Eu-152	1.18E-01	N/A	U	3.02E-01	1.42E-01	8.70E+00	1.36E-02	
	Eu-154	1.24E-01	N/A	U	3.94E-01	1.71E-01	8.00E+00	1.55E-02	
	Eu-155	1.17E-01	N/A	U	3.97E-01	2.00E-01	2.80E+02	4.18E-04	
	Am-241	0.00E+00	N/A	U	3.53E-01	1.67E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-021	Co-60	4.69E-02	N/A	U	1.20E-01	5.72E-02	3.80E+00	1.23E-02	0.0427
	Cs-134	0.00E+00	N/A	U	1.58E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	2.64E-01	3.98E-02		1.01E-01	4.31E-02	1.10E+01	2.40E-02	
	Eu-152	0.00E+00	N/A	U	3.18E-01	1.50E-01	8.70E+00	0.00E+00	
	Eu-154	4.49E-02	N/A	U	4.08E-01	1.79E-01	8.00E+00	5.61E-03	
	Eu-155	2.10E-01	N/A	U	3.15E-01	1.49E-01	2.80E+02	0.00075	
	Am-241	0.00E+00	N/A	U	3.61E-01	1.72E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-022	Co-60	3.68E-02	N/A	U	7.41E-02	5.92E-02	3.80E+00	9.68E-03	0.0238
	Cs-134	2.96E-02	N/A	U	1.40E-01	9.66E-02	5.70E+00	5.19E-03	
	Cs-137	8.48E-02	N/A	U	1.50E-01	6.87E-02	1.10E+01	7.71E-03	
	Eu-152	0.00E+00	N/A	U	2.63E-01	1.23E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.32E-01	1.43E-01	8.00E+00	0.00E+00	
	Eu-155	3.45E-01	N/A	U	3.84E-01	1.84E-01	2.80E+02	1.23E-03	
	Am-241	0.00E+00	N/A	U	3.22E-01	1.53E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8300X-3-CR-GSSX-023	Co-60	2.37E-03	N/A	U	1.17E-01	5.74E-02	3.80E+00	6.24E-04	0.0552
	Cs-134	0.00E+00	N/A	U	1.46E-01	1.02E-01	5.70E+00	0.00E+00	
	Cs-137	6.66E-02	3.86E-02		1.52E-01	6.91E-02	1.10E+01	6.05E-03	
	Eu-152	0.00E+00	N/A	U	3.03E-01	1.43E-01	8.70E+00	0.00E+00	
	Eu-154	9.28E-02	N/A	U	4.34E-01	1.92E-01	8.00E+00	1.16E-02	
	Eu-155	1.69E-01	N/A	U	2.74E-01	1.28E-01	2.80E+02	6.04E-04	
	Am-241	7.62E-02	N/A	U	3.88E-01	1.85E-01	2.10E+00	3.63E-02	
8300X-3-CR-GSSX-024	Co-60	8.49E-02	N/A	U	1.33E-01	5.71E-02	3.80E+00	2.23E-02	0.0401
	Cs-134	0.00E+00	N/A	U	1.56E-01	1.23E-01	5.70E+00	0.00E+00	
	Cs-137	1.75E-01	3.57E-02		1.06E-01	4.54E-02	1.10E+01	1.59E-02	
	Eu-152	0.00E+00	N/A	U	3.44E-01	1.62E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.54E-01	2.00E-01	8.00E+00	0.00E+00	
	Eu-155	5.12E-01	N/A	U	3.23E-01	1.52E-01	2.80E+02	1.83E-03	
	Am-241	0.00E+00	N/A	U	3.74E-01	1.78E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSA-025	Co-60	0.00E+00	N/A	U	1.41E-01	7.57E-02	3.80E+00	0.00E+00	0.0138
	Cs-134	0.00E+00	N/A	U	1.69E-01	1.32E-01	5.70E+00	0.00E+00	
	Cs-137	1.26E-01	3.93E-02		1.37E-01	5.93E-02	1.10E+01	1.15E-02	
	Eu-152	6.84E-03	N/A	U	4.12E-01	1.94E-01	8.70E+00	7.86E-04	
	Eu-154	0.00E+00	N/A	U	3.47E-01	1.41E-01	8.00E+00	0.00E+00	
	Eu-155	4.38E-01	N/A	U	3.63E-01	1.70E-01	2.80E+02	1.56E-03	
	Am-241	0.00E+00	N/A	U	4.57E-01	2.17E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-026	Co-60	0.00E+00	N/A	U	1.01E-01	4.20E-02	3.80E+00	0.00E+00	0.0232
	Cs-134	0.00E+00	N/A	U	1.37E-01	9.35E-02	5.70E+00	0.00E+00	
	Cs-137	1.01E-01	N/A	U	1.68E-01	7.72E-02	1.10E+01	9.18E-03	
	Eu-152	0.00E+00	N/A	U	2.90E-01	1.37E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.63E-01	1.58E-01	8.00E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-155	1.25E-01	N/A	U	4.05E-01	1.94E-01	2.80E+02	4.46E-04	
	Am-241	2.84E-02	N/A	U	3.28E-01	1.56E-01	2.10E+00	1.35E-02	
8300X-3-CR-GSSX-027	Co-60	5.51E-03	N/A	U	1.02E-01	4.22E-02	3.80E+00	1.45E-03	0.0477
	Cs-134	0.00E+00	N/A	U	1.44E-01	1.13E-01	5.70E+00	0.00E+00	
	Cs-137	6.69E-02	N/A	U	1.68E-01	7.72E-02	1.10E+01	6.08E-03	
	Eu-152	0.00E+00	N/A	U	2.87E-01	1.35E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.31E-01	1.41E-01	8.00E+00	0.00E+00	
	Eu-155	1.02E-01	N/A	U	2.70E-01	1.26E-01	2.80E+02	3.64E-04	
	Am-241	8.36E-02	N/A	U	3.76E-01	1.79E-01	2.10E+00	3.98E-02	
8300X-3-CR-GSSX-028	Co-60	0.00E+00	N/A	U	1.02E-01	4.33E-02	3.80E+00	0.00E+00	0.0302
	Cs-134	0.00E+00	N/A	U	1.33E-01	9.30E-02	5.70E+00	0.00E+00	
	Cs-137	9.23E-02	3.00E-02		1.08E-01	4.77E-02	1.10E+01	8.39E-03	
	Eu-152	8.49E-02	N/A	U	3.03E-01	1.44E-01	8.70E+00	9.76E-03	
	Eu-154	8.99E-02	N/A	U	3.42E-01	1.49E-01	8.00E+00	1.12E-02	
	Eu-155	2.31E-01	N/A	U	3.87E-01	1.87E-01	2.80E+02	8.25E-04	
	Am-241	0.00E+00	N/A	U	3.22E-01	1.53E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-029	Co-60	4.74E-02	N/A	U	1.45E-01	6.28E-02	3.80E+00	1.25E-02	0.0333
	Cs-134	0.00E+00	N/A	U	1.82E-01	1.18E-01	5.70E+00	0.00E+00	
	Cs-137	9.63E-02	2.80E-02		9.20E-02	3.83E-02	1.10E+01	8.75E-03	
	Eu-152	0.00E+00	N/A	U	3.14E-01	1.47E-01	8.70E+00	0.00E+00	
	Eu-154	8.46E-02	N/A	U	3.67E-01	1.56E-01	8.00E+00	1.06E-02	
	Eu-155	4.16E-01	N/A	U	4.46E-01	2.35E-01	2.80E+02	1.49E-03	
	Am-241	0.00E+00	N/A	U	3.94E-01	1.87E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-030	Co-60	5.30E-02	N/A	U	1.14E-01	6.48E-02	3.80E+00	1.39E-02	0.0402
	Cs-134	1.40E-02	N/A	U	1.58E-01	1.04E-01	5.70E+00	2.46E-03	
	Cs-137	1.67E-01	3.24E-02		9.44E-02	4.06E-02	1.10E+01	1.52E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	6.41E-02	N/A	U	3.17E-01	1.50E-01	8.70E+00	7.37E-03	
	Eu-154	1.46E-03	N/A	U	3.83E-01	1.68E-01	8.00E+00	1.83E-04	
	Eu-155	3.10E-01	N/A	U	2.56E-01	1.20E-01	2.80E+02	1.11E-03	
	Am-241	0.00E+00	N/A	U	3.41E-01	1.62E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-031	Co-60	0.00E+00	N/A	U	1.19E-01	5.15E-02	3.80E+00	0.00E+00	0.0146
	Cs-134	0.00E+00	N/A	U	1.47E-01	8.97E-02	5.70E+00	0.00E+00	
	Cs-137	1.59E-01	3.40E-02		1.05E-01	4.57E-02	1.10E+01	1.45E-02	
	Eu-152	0.00E+00	N/A	U	2.68E-01	1.36E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.65E-01	1.58E-01	8.00E+00	0.00E+00	
	Eu-155	4.82E-02	N/A	U	3.73E-01	1.78E-01	2.80E+02	1.72E-04	
	Am-241	0.00E+00	N/A	U	3.38E-01	1.61E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-032	Co-60	4.90E-02	N/A	U	1.19E-01	5.12E-02	3.80E+00	1.29E-02	0.0275
	Cs-134	0.00E+00	N/A	U	1.41E-01	9.27E-02	5.70E+00	0.00E+00	
	Cs-137	1.52E-01	3.30E-02		1.02E-01	4.42E-02	1.10E+01	1.38E-02	
	Eu-152	0.00E+00	N/A	U	3.07E-01	1.45E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.33E-01	1.42E-01	8.00E+00	0.00E+00	
	Eu-155	2.29E-01	N/A	U	3.23E-01	1.53E-01	2.80E+02	8.18E-04	
	Am-241	0.00E+00	N/A	U	3.54E-01	1.69E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-033	Co-60	5.75E-02	N/A	U	1.28E-01	5.50E-02	3.80E+00	1.51E-02	0.0258
	Cs-134	0.00E+00	N/A	U	1.59E-01	1.02E-01	5.70E+00	0.00E+00	
	Cs-137	1.03E-01	N/A	U	1.66E-01	7.61E-02	1.10E+01	9.36E-03	
	Eu-152	0.00E+00	N/A	U	3.03E-01	1.42E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.93E-01	1.72E-01	8.00E+00	0.00E+00	
	Eu-155	3.70E-01	N/A	U	4.27E-01	2.05E-01	2.80E+02	1.32E-03	
	Am-241	0.00E+00	N/A	U	3.58E-01	1.71E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-034	Co-60	0.00E+00	N/A	U	1.33E-01	7.79E-02	3.80E+00	0.00E+00	0.0042

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.65E-01	1.16E-01	5.70E+00	0.00E+00	
	Cs-137	3.77E-02	2.36E-02		9.43E-02	3.92E-02	1.10E+01	3.43E-03	
	Eu-152	0.00E+00	N/A	U	3.37E-01	1.58E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.36E-01	1.40E-01	8.00E+00	0.00E+00	
	Eu-155	2.21E-01	N/A	U	3.07E-01	1.44E-01	2.80E+02	7.89E-04	
	Am-241	0.00E+00	N/A	U	4.00E-01	1.90E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-035	Co-60	4.20E-03	N/A	U	1.14E-01	4.74E-02	3.80E+00	1.11E-03	0.0980
	Cs-134	0.00E+00	N/A	U	1.72E-01	1.19E-01	5.70E+00	0.00E+00	
	Cs-137	1.53E-01	3.45E-02		1.08E-01	4.63E-02	1.10E+01	1.39E-02	
	Eu-152	0.00E+00	N/A	U	3.35E-01	1.58E-01	8.70E+00	0.00E+00	
	Eu-154	1.10E-01	N/A	U	4.18E-01	1.83E-01	8.00E+00	1.38E-02	
	Eu-155	1.88E-01	N/A	U	2.78E-01	1.30E-01	2.80E+02	6.71E-04	
	Am-241	1.44E-01	N/A	U	4.03E-01	1.92E-01	2.10E+00	6.86E-02	
8300X-3-CR-GSSX-036	Co-60	0.00E+00	N/A	U	1.03E-01	4.28E-02	3.80E+00	0.00E+00	0.0064
	Cs-134	0.00E+00	N/A	U	1.46E-01	9.42E-02	5.70E+00	0.00E+00	
	Cs-137	6.14E-02	N/A	U	1.34E-01	6.04E-02	1.10E+01	5.58E-03	
	Eu-152	0.00E+00	N/A	U	2.77E-01	1.30E-01	8.70E+00	0.00E+00	
	Eu-154	6.20E-04	N/A	U	3.55E-01	1.54E-01	8.00E+00	7.75E-05	
	Eu-155	2.01E-01	N/A	U	2.87E-01	1.35E-01	2.80E+02	7.18E-04	
	Am-241	0.00E+00	N/A	U	3.24E-01	1.54E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-037	Co-60	4.03E-02	N/A	U	1.47E-01	7.05E-02	3.80E+00	1.06E-02	0.0277
	Cs-134	0.00E+00	N/A	U	1.63E-01	9.82E-02	5.70E+00	0.00E+00	
	Cs-137	1.83E-01	3.88E-02		1.22E-01	5.37E-02	1.10E+01	1.66E-02	
	Eu-152	0.00E+00	N/A	U	3.06E-01	1.44E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.93E-01	1.71E-01	8.00E+00	0.00E+00	
	Eu-155	1.15E-01	N/A	U	3.38E-01	1.60E-01	2.80E+02	4.11E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	3.77E-01	1.79E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-038	Co-60	1.75E-02	N/A	U	9.65E-02	4.02E-02	3.80E+00	4.61E-03	0.0111
	Cs-134	0.00E+00	N/A	U	1.36E-01	8.11E-02	5.70E+00	0.00E+00	
	Cs-137	6.32E-02	2.48E-02		9.10E-02	3.91E-02	1.10E+01	5.75E-03	
	Eu-152	0.00E+00	N/A	U	2.82E-01	1.33E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.75E-01	1.65E-01	8.00E+00	0.00E+00	
	Eu-155	2.18E-01	N/A	U	4.02E-01	1.93E-01	2.80E+02	7.79E-04	
	Am-241	0.00E+00	N/A	U	2.89E-01	1.36E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-039	Co-60	2.75E-02	N/A	U	1.08E-01	4.49E-02	3.80E+00	7.24E-03	0.0211
	Cs-134	0.00E+00	N/A	U	1.46E-01	9.87E-02	5.70E+00	0.00E+00	
	Cs-137	1.11E-01	N/A	U	1.63E-01	7.46E-02	1.10E+01	1.01E-02	
	Eu-152	2.88E-02	N/A	U	3.14E-01	1.48E-01	8.70E+00	3.31E-03	
	Eu-154	0.00E+00	N/A	U	3.56E-01	1.53E-01	8.00E+00	0.00E+00	
	Eu-155	1.21E-01	N/A	U	2.90E-01	1.37E-01	2.80E+02	4.32E-04	
	Am-241	0.00E+00	N/A	U	3.41E-01	1.62E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-040	Co-60	8.90E-03	N/A	U	1.25E-01	5.25E-02	3.80E+00	2.34E-03	0.0331
	Cs-134	0.00E+00	N/A	U	1.49E-01	1.18E-01	5.70E+00	0.00E+00	
	Cs-137	1.88E-01	4.05E-02		1.28E-01	5.65E-02	1.10E+01	1.71E-02	
	Eu-152	0.00E+00	N/A	U	3.38E-01	1.59E-01	8.70E+00	0.00E+00	
	Eu-154	1.09E-01	N/A	U	3.86E-01	1.66E-01	8.00E+00	1.36E-02	
	Eu-155	2.39E-02	N/A	U	2.80E-01	1.31E-01	2.80E+02	8.54E-05	
	Am-241	0.00E+00	N/A	U	3.99E-01	1.90E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSA-041	Co-60	0.00E+00	N/A	U	1.04E-01	5.46E-02	3.80E+00	0.00E+00	0.0045
	Cs-134	0.00E+00	N/A	U	1.45E-01	9.99E-02	5.70E+00	0.00E+00	
	Cs-137	4.63E-02	2.20E-02		8.34E-02	3.53E-02	1.10E+01	4.21E-03	
	Eu-152	0.00E+00	N/A	U	2.89E-01	1.36E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	2.97E-01	1.26E-01	8.00E+00	0.00E+00	
	Eu-155	8.41E-02	N/A	U	2.49E-01	1.17E-01	2.80E+02	3.00E-04	
	Am-241	0.00E+00	N/A	U	3.34E-01	1.59E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSX-042	Co-60	0.00E+00	N/A	U	9.39E-02	3.81E-02	3.80E+00	0.00E+00	0.0250
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	1.84E-01	3.45E-02		9.90E-02	4.25E-02	1.10E+01	1.67E-02	
	Eu-152	6.65E-02	N/A	U	3.38E-01	1.60E-01	8.70E+00	7.64E-03	
	Eu-154	0.00E+00	N/A	U	3.73E-01	1.62E-01	8.00E+00	0.00E+00	
	Eu-155	1.81E-01	N/A	U	2.84E-01	1.34E-01	2.80E+02	6.46E-04	
	Am-241	0.00E+00	N/A	U	3.56E-01	1.69E-01	2.10E+00	0.00E+00	

Table 2-57 8300 Gamma Spectroscopy Results for QC Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8300X-3-CQ-GSSX-004	Co-60	1.09E-02	N/A	U	1.14E-01	6.03E-02	3.80E+00	2.87E-03	0.0191
	Cs-134	0.00E+00	N/A	U	1.59E-01	9.81E-02	5.70E+00	0.00E+00	
	Cs-137	1.47E-01	3.21E-02		9.87E-02	4.25E-02	1.10E+01	1.34E-02	
	Eu-152	1.83E-02	N/A	U	3.06E-01	1.44E-01	8.70E+00	2.10E-03	
	Eu-154	0.00E+00	N/A	U	3.42E-01	1.47E-01	8.00E+00	0.00E+00	
	Eu-155	2.08E-01	N/A	U	3.99E-01	1.91E-01	2.80E+02	7.43E-04	
	Am-241	0.00E+00	N/A	U	3.50E-01	1.66E-01	2.10E+00	0.00E+00	
8300X-3-CQ-GSSX-014	Co-60	0.00E+00	N/A	U	1.06E-01	4.91E-02	3.80E+00	0.00E+00	0.0018
	Cs-134	0.00E+00	N/A	U	1.40E-01	9.00E-02	5.70E+00	0.00E+00	
	Cs-137	8.73E-03	N/A	U	1.28E-01	5.77E-02	1.10E+01	7.94E-04	
	Eu-152	0.00E+00	N/A	U	2.84E-01	1.34E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.42E-01	1.50E-01	8.00E+00	0.00E+00	
	Eu-155	2.93E-01	N/A	U	3.78E-01	1.89E-01	2.80E+02	1.05E-03	
	Am-241	0.00E+00	N/A	U	3.06E-01	1.46E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSB-025	Co-60	2.26E-02	N/A	U	1.22E-01	7.35E-02	3.80E+00	5.95E-03	0.0206
	Cs-134	0.00E+00	N/A	U	1.38E-01	1.02E-01	5.70E+00	0.00E+00	
	Cs-137	1.55E-01	3.16E-02		9.14E-02	3.86E-02	1.10E+01	1.41E-02	
	Eu-152	0.00E+00	N/A	U	2.98E-01	1.40E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.95E-01	1.72E-01	8.00E+00	0.00E+00	
	Eu-155	1.69E-01	N/A	U	4.32E-01	2.10E-01	2.80E+02	6.04E-04	
	Am-241	0.00E+00	N/A	U	3.42E-01	1.62E-01	2.10E+00	0.00E+00	
8300X-3-CQ-GSSX-026	Co-60	7.45E-03	N/A	U	1.11E-01	4.72E-02	3.80E+00	1.96E-03	0.0329
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	1.95E-01	2.96E-02		7.99E-02	3.33E-02	1.10E+01	1.77E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.13E-01	1.48E-01	8.70E+00	0.00E+00	
	Eu-154	9.70E-02	N/A	U	4.19E-01	1.86E-01	8.00E+00	1.21E-02	
	Eu-155	2.96E-01	N/A	U	4.33E-01	2.08E-01	2.80E+02	1.06E-03	
	Am-241	0.00E+00	N/A	U	3.39E-01	1.61E-01	2.10E+00	0.00E+00	
8300X-3-CR-GSSB-041	Co-60	4.22E-02	N/A	U	1.26E-01	5.59E-02	3.80E+00	1.11E-02	0.0226
	Cs-134	0.00E+00	N/A	U	1.61E-01	1.17E-01	5.70E+00	0.00E+00	
	Cs-137	1.17E-01	N/A	U	1.55E-01	7.06E-02	1.10E+01	1.06E-02	
	Eu-152	0.00E+00	N/A	U	3.22E-01	1.52E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.91E-01	1.71E-01	8.00E+00	0.00E+00	
	Eu-155	2.28E-01	N/A	U	2.94E-01	1.39E-01	2.80E+02	8.14E-04	
	Am-241	0.00E+00	N/A	U	3.62E-01	1.72E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSB-055	Co-60	7.34E-03	N/A	U	1.35E-01	8.02E-02	3.80E+00	1.93E-03	0.0243
	Cs-134	0.00E+00	N/A	U	2.00E-01	1.20E-01	5.70E+00	0.00E+00	
	Cs-137	8.32E-02	N/A	U	2.12E-01	9.70E-02	1.10E+01	7.56E-03	
	Eu-152	9.90E-02	N/A	U	4.03E-01	1.91E-01	8.70E+00	1.14E-02	
	Eu-154	2.06E-02	N/A	U	4.12E-01	1.75E-01	8.00E+00	2.58E-03	
	Eu-155	2.46E-01	N/A	U	3.64E-01	1.72E-01	2.80E+02	8.79E-04	
	Am-241	0.00E+00	N/A	U	4.21E-01	2.00E-01	2.10E+00	0.00E+00	
8300X-3-CQ-GSSX-068	Co-60	1.75E-02	N/A	U	1.51E-01	7.27E-02	3.80E+00	4.61E-03	0.0362
	Cs-134	0.00E+00	N/A	U	1.59E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	1.56E-01	3.62E-02		1.15E-01	4.98E-02	1.10E+01	1.42E-02	
	Eu-152	0.00E+00	N/A	U	3.06E-01	1.43E-01	8.70E+00	0.00E+00	
	Eu-154	1.37E-01	N/A	U	4.44E-01	1.95E-01	8.00E+00	1.71E-02	
	Eu-155	8.47E-02	N/A	U	4.40E-01	2.11E-01	2.80E+02	3.03E-04	
	Am-241	0.00E+00	N/A	U	3.55E-01	1.68E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSB-071	Co-60	0.00E+00	N/A	U	1.44E-01	5.98E-02	3.80E+00	0.00E+00	0.0334

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.82E-01	1.43E-01	5.70E+00	0.00E+00	
	Cs-137	9.56E-02	N/A	U	2.08E-01	9.46E-02	1.10E+01	8.69E-03	
	Eu-152	0.00E+00	N/A	U	3.92E-01	1.85E-01	8.70E+00	0.00E+00	
	Eu-154	1.90E-01	N/A	U	4.51E-01	2.69E-01	8.00E+00	2.38E-02	
	Eu-155	2.63E-01	N/A	U	5.24E-01	2.69E-01	2.80E+02	9.39E-04	
	Am-241	0.00E+00	N/A	U	4.20E-01	1.99E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSB-079	Co-60	7.59E-04	N/A	U	1.27E-01	5.67E-02	3.80E+00	2.00E-04	0.0136
	Cs-134	0.00E+00	N/A	U	1.53E-01	9.37E-02	5.70E+00	0.00E+00	
	Cs-137	1.25E-01	3.50E-02		1.18E-01	5.09E-02	1.10E+01	1.14E-02	
	Eu-152	0.00E+00	N/A	U	3.34E-01	1.57E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.96E-01	1.69E-01	8.00E+00	0.00E+00	
	Eu-155	5.57E-01	N/A	U	4.68E-01	2.26E-01	2.80E+02	1.99E-03	
	Am-241	0.00E+00	N/A	U	3.55E-01	1.68E-01	2.10E+00	0.00E+00	
8300X-3-CQ-GSSX-081	Co-60	7.04E-04	N/A	U	9.12E-02	3.61E-02	3.80E+00	1.85E-04	0.0230
	Cs-134	0.00E+00	N/A	U	1.69E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	1.11E-01	N/A	U	1.82E-01	8.35E-02	1.10E+01	1.01E-02	
	Eu-152	1.19E-02	N/A	U	3.28E-01	1.55E-01	8.70E+00	1.37E-03	
	Eu-154	8.63E-02	N/A	U	3.84E-01	1.66E-01	8.00E+00	1.08E-02	
	Eu-155	1.52E-01	N/A	U	4.46E-01	2.14E-01	2.80E+02	5.43E-04	
	Am-241	0.00E+00	N/A	U	3.59E-01	1.70E-01	2.10E+00	0.00E+00	

Table 2-58 8300 Gamma Spectroscopy Results for Judgmental Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8300X-3-CI-GSSX-03A	Co-60	1.01E-02	N/A	U	1.69E-01	9.30E-02	3.80E+00	2.66E-03	0.0227
	Cs-134	0.00E+00	N/A	U	1.95E-01	1.48E-01	5.70E+00	0.00E+00	
	Cs-137	2.74E-02	N/A	U	2.20E-01	9.99E-02	1.10E+01	2.49E-03	
	Eu-152	1.50E-01	N/A	U	4.16E-01	1.95E-01	8.70E+00	1.72E-02	
	Eu-154	0.00E+00	N/A	U	4.29E-01	1.79E-01	8.00E+00	0.00E+00	
	Eu-155	9.34E-02	N/A	U	3.83E-01	1.80E-01	2.80E+02	3.34E-04	
	Am-241	0.00E+00	N/A	U	4.59E-01	2.18E-01	2.10E+00	0.00E+00	
8300X-3-CI-GSSX-03D	Co-60	3.55E-02	N/A	U	1.34E-01	7.51E-02	3.80E+00	9.34E-03	0.0218
	Cs-134	0.00E+00	N/A	U	1.70E-01	1.12E-01	5.70E+00	0.00E+00	
	Cs-137	3.87E-02	N/A	U	1.72E-01	7.79E-02	1.10E+01	3.52E-03	
	Eu-152	7.37E-02	N/A	U	3.39E-01	1.59E-01	8.70E+00	8.47E-03	
	Eu-154	0.00E+00	N/A	U	4.58E-01	2.00E-01	8.00E+00	0.00E+00	
	Eu-155	1.36E-01	N/A	U	4.65E-01	2.23E-01	2.80E+02	4.86E-04	
	Am-241	0.00E+00	N/A	U	3.72E-01	1.76E-01	2.10E+00	0.00E+00	
8300X-3-CI-GSSX-09A	Co-60	4.38E-02	N/A	U	1.22E-01	7.48E-02	3.80E+00	1.15E-02	0.0472
	Cs-134	0.00E+00	N/A	U	1.72E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	6.69E-02	N/A	U	1.76E-01	8.00E-02	1.10E+01	6.08E-03	
	Eu-152	6.43E-02	N/A	U	3.24E-01	1.52E-01	8.70E+00	7.39E-03	
	Eu-154	1.69E-01	N/A	U	4.12E-01	1.78E-01	8.00E+00	2.11E-02	
	Eu-155	3.04E-01	N/A	U	4.54E-01	2.26E-01	2.80E+02	1.09E-03	
	Am-241	0.00E+00	N/A	U	3.61E-01	1.71E-01	2.10E+00	0.00E+00	
8300X-3-CI-GSSX-09C	Co-60	0.00E+00	N/A	U	1.39E-01	5.95E-02	3.80E+00	0.00E+00	0.0153
	Cs-134	0.00E+00	N/A	U	1.63E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	1.61E-01	N/A	U	2.02E-01	9.29E-02	1.10E+01	1.46E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.30E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.88E-01	1.66E-01	8.00E+00	0.00E+00	
	Eu-155	1.84E-01	N/A	U	2.86E-01	1.33E-01	2.80E+02	6.57E-04	
	Am-241	0.00E+00	N/A	U	3.76E-01	1.78E-01	2.10E+00	0.00E+00	
8300X-3-CI-GSSX-015	Co-60	1.84E-02	N/A	U	1.18E-01	4.96E-02	3.80E+00	4.84E-03	0.0361
	Cs-134	0.00E+00	N/A	U	1.60E-01	1.00E-01	5.70E+00	0.00E+00	
	Cs-137	1.51E-01	N/A	U	2.11E-01	9.79E-02	1.10E+01	1.37E-02	
	Eu-152	5.51E-02	N/A	U	3.22E-01	1.51E-01	8.70E+00	6.33E-03	
	Eu-154	8.38E-02	N/A	U	3.69E-01	1.58E-01	8.00E+00	1.05E-02	
	Eu-155	2.12E-01	N/A	U	4.56E-01	2.19E-01	2.80E+02	7.57E-04	
	Am-241	0.00E+00	N/A	U	3.72E-01	1.77E-01	2.10E+00	0.00E+00	
8300X-3-CI-GSSX-15A	Co-60	5.33E-02	N/A	U	1.37E-01	6.51E-02	3.80E+00	1.40E-02	0.0607
	Cs-134	0.00E+00	N/A	U	1.35E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	8.44E-02	N/A	U	2.00E-01	9.21E-02	1.10E+01	7.67E-03	
	Eu-152	5.34E-02	N/A	U	3.37E-01	1.59E-01	8.70E+00	6.14E-03	
	Eu-154	0.00E+00	N/A	U	4.59E-01	2.02E-01	8.00E+00	0.00E+00	
	Eu-155	5.00E-02	N/A	U	4.52E-01	2.17E-01	2.80E+02	1.79E-04	
	Am-241	6.87E-02	N/A	U	4.09E-01	1.95E-01	2.10E+00	3.27E-02	
8300X-3-CI-GSSX-017	Co-60	6.92E-02	N/A	U	1.41E-01	6.72E-02	3.80E+00	1.82E-02	0.0432
	Cs-134	0.00E+00	N/A	U	2.13E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	2.62E-01	N/A	U	2.17E-01	1.00E-01	1.10E+01	2.38E-02	
	Eu-152	0.00E+00	N/A	U	3.20E-01	1.50E-01	8.70E+00	0.00E+00	
	Eu-154	1.77E-04	N/A	U	4.03E-01	1.72E-01	8.00E+00	2.21E-05	
	Eu-155	3.23E-01	N/A	U	4.72E-01	2.26E-01	2.80E+02	1.15E-03	
	Am-241	0.00E+00	N/A	U	3.71E-01	1.76E-01	2.10E+00	0.00E+00	
8300X-3-CI-GSSX-019	Co-60	0.00E+00	N/A	U	1.44E-01	5.97E-02	3.80E+00	0.00E+00	0.0299

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.27E-01	5.70E+00	0.00E+00	
	Cs-137	1.86E-01	N/A	U	2.28E-01	1.04E-01	1.10E+01	1.69E-02	
	Eu-152	8.68E-02	N/A	U	4.05E-01	1.90E-01	8.70E+00	9.98E-03	
	Eu-154	1.73E-02	N/A	U	4.94E-01	2.13E-01	8.00E+00	2.16E-03	
	Eu-155	2.40E-01	N/A	U	5.36E-01	2.65E-01	2.80E+02	8.57E-04	
	Am-241	0.00E+00	N/A	U	4.38E-01	2.08E-01	2.10E+00	0.00E+00	
8300X-3-CI-GSSX-26A	Co-60	3.80E-02	N/A	U	1.30E-01	5.58E-02	3.80E+00	1.00E-02	0.0404
	Cs-134	0.00E+00	N/A	U	1.59E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	9.13E-02	3.39E-02		1.25E-01	5.52E-02	1.10E+01	8.30E-03	
	Eu-152	2.92E-02	N/A	U	3.44E-01	1.63E-01	8.70E+00	3.36E-03	
	Eu-154	0.00E+00	N/A	U	3.61E-01	1.55E-01	8.00E+00	0.00E+00	
	Eu-155	3.25E-01	N/A	U	4.30E-01	2.06E-01	2.80E+02	1.16E-03	
	Am-241	3.70E-02	N/A	U	3.65E-01	1.74E-01	2.10E+00	1.76E-02	
8300X-3-CI-GSSX-33A	Co-60	5.70E-02	N/A	U	9.82E-02	6.50E-02	3.80E+00	1.50E-02	0.0263
	Cs-134	0.00E+00	N/A	U	1.61E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	8.96E-02	2.76E-02		9.30E-02	3.89E-02	1.10E+01	8.15E-03	
	Eu-152	1.76E-02	N/A	U	3.22E-01	1.51E-01	8.70E+00	2.02E-03	
	Eu-154	0.00E+00	N/A	U	4.42E-01	1.94E-01	8.00E+00	0.00E+00	
	Eu-155	3.04E-01	N/A	U	4.39E-01	2.22E-01	2.80E+02	1.09E-03	
	Am-241	0.00E+00	N/A	U	3.61E-01	1.71E-01	2.10E+00	0.00E+00	
8300X-3-CI-GSSX-40B	Co-60	0.00E+00	N/A	U	1.24E-01	5.16E-02	3.80E+00	0.00E+00	0.0169
	Cs-134	0.00E+00	N/A	U	1.58E-01	1.12E-01	5.70E+00	0.00E+00	
	Cs-137	1.35E-01	3.56E-02		1.17E-01	5.04E-02	1.10E+01	1.23E-02	
	Eu-152	3.37E-02	N/A	U	3.54E-01	1.67E-01	8.70E+00	3.87E-03	
	Eu-154	0.00E+00	N/A	U	4.46E-01	1.94E-01	8.00E+00	0.00E+00	
	Eu-155	2.14E-01	N/A	U	4.49E-01	2.30E-01	2.80E+02	7.64E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	4.00E-01	1.90E-01	2.10E+00	0.00E+00	
8300X-3-CI-GSSX-42A	Co-60	1.50E-02	N/A	U	1.30E-01	7.25E-02	3.80E+00	3.95E-03	0.0275
	Cs-134	0.00E+00	N/A	U	1.59E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	2.48E-01	4.61E-02		1.41E-01	6.28E-02	1.10E+01	2.25E-02	
	Eu-152	0.00E+00	N/A	U	3.41E-01	1.61E-01	8.70E+00	0.00E+00	
	Eu-154	4.27E-03	N/A	U	4.25E-01	1.85E-01	8.00E+00	5.34E-04	
	Eu-155	1.38E-01	N/A	U	4.53E-01	2.17E-01	2.80E+02	4.93E-04	
	Am-241	0.00E+00	N/A	U	3.71E-01	1.76E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-043	Co-60	1.19E-02	N/A	U	1.61E-01	7.76E-02	3.80E+00	3.13E-03	0.0499
	Cs-134	0.00E+00	N/A	U	2.04E-01	1.37E-01	5.70E+00	0.00E+00	
	Cs-137	1.51E-01	5.43E-02		2.01E-01	9.06E-02	1.10E+01	1.37E-02	
	Eu-152	1.51E-01	N/A	U	4.39E-01	2.07E-01	8.70E+00	1.74E-02	
	Eu-154	1.17E-01	N/A	U	5.75E-01	2.53E-01	8.00E+00	1.46E-02	
	Eu-155	3.00E-01	N/A	U	6.00E-01	2.88E-01	2.80E+02	1.07E-03	
	Am-241	0.00E+00	N/A	U	4.90E-01	2.33E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-044	Co-60	5.65E-02	N/A	U	1.50E-01	6.53E-02	3.80E+00	1.49E-02	0.0331
	Cs-134	0.00E+00	N/A	U	1.70E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	1.27E-01	3.11E-02		9.68E-02	4.06E-02	1.10E+01	1.15E-02	
	Eu-152	0.00E+00	N/A	U	3.56E-01	1.68E-01	8.70E+00	0.00E+00	
	Eu-154	5.34E-02	N/A	U	4.51E-01	1.98E-01	8.00E+00	6.68E-03	
	Eu-155	0.00E+00	N/A	U	4.98E-01	2.39E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	4.07E-01	1.94E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-045	Co-60	3.25E-02	N/A	U	1.13E-01	6.71E-02	3.80E+00	8.55E-03	0.0164
	Cs-134	0.00E+00	N/A	U	1.51E-01	9.77E-02	5.70E+00	0.00E+00	
	Cs-137	3.30E-02	2.43E-02		9.88E-02	4.20E-02	1.10E+01	3.00E-03	
	Eu-152	3.32E-02	N/A	U	3.01E-01	1.41E-01	8.70E+00	3.82E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	3.73E-01	1.60E-01	8.00E+00	0.00E+00	
	Eu-155	2.76E-01	N/A	U	4.38E-01	2.18E-01	2.80E+02	9.86E-04	
	Am-241	0.00E+00	N/A	U	3.43E-01	1.62E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-046	Co-60	3.20E-02	N/A	U	1.17E-01	4.86E-02	3.80E+00	8.42E-03	0.0424
	Cs-134	0.00E+00	N/A	U	1.64E-01	9.67E-02	5.70E+00	0.00E+00	
	Cs-137	8.05E-02	N/A	U	1.75E-01	7.97E-02	1.10E+01	7.32E-03	
	Eu-152	0.00E+00	N/A	U	2.99E-01	1.40E-01	8.70E+00	0.00E+00	
	Eu-154	3.82E-02	N/A	U	4.11E-01	1.79E-01	8.00E+00	4.78E-03	
	Eu-155	1.77E-01	N/A	U	3.01E-01	1.41E-01	2.80E+02	6.32E-04	
	Am-241	4.47E-02	N/A	U	3.74E-01	1.78E-01	2.10E+00	2.13E-02	
8300X-3-CJ-GSSX-047	Co-60	5.53E-04	N/A	U	1.19E-01	5.85E-02	3.80E+00	1.46E-04	0.0056
	Cs-134	0.00E+00	N/A	U	1.71E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	5.07E-02	N/A	U	1.67E-01	7.52E-02	1.10E+01	4.61E-03	
	Eu-152	0.00E+00	N/A	U	3.07E-01	1.43E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.06E-01	1.74E-01	8.00E+00	0.00E+00	
	Eu-155	2.25E-01	N/A	U	4.33E-01	2.18E-01	2.80E+02	8.04E-04	
	Am-241	0.00E+00	N/A	U	3.53E-01	1.67E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-048	Co-60	1.58E-02	N/A	U	1.10E-01	6.67E-02	3.80E+00	4.16E-03	0.0110
	Cs-134	0.00E+00	N/A	U	1.44E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	3.10E-02	N/A	U	1.63E-01	7.37E-02	1.10E+01	2.82E-03	
	Eu-152	2.71E-02	N/A	U	3.26E-01	1.53E-01	8.70E+00	3.11E-03	
	Eu-154	0.00E+00	N/A	U	4.11E-01	1.78E-01	8.00E+00	0.00E+00	
	Eu-155	2.63E-01	N/A	U	4.32E-01	2.07E-01	2.80E+02	9.39E-04	
	Am-241	0.00E+00	N/A	U	3.66E-01	1.74E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-049	Co-60	8.30E-02	N/A	U	1.12E-01	7.61E-02	3.80E+00	2.18E-02	0.0304
	Cs-134	0.00E+00	N/A	U	1.63E-01	1.17E-01	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	3.62E-02	N/A	U	1.74E-01	7.90E-02	1.10E+01	3.29E-03	
	Eu-152	0.00E+00	N/A	U	2.85E-01	1.32E-01	8.70E+00	0.00E+00	
	Eu-154	3.88E-02	N/A	U	4.36E-01	1.89E-01	8.00E+00	4.85E-03	
	Eu-155	1.30E-01	N/A	U	4.33E-01	2.07E-01	2.80E+02	4.64E-04	
	Am-241	0.00E+00	N/A	U	3.51E-01	1.66E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-050	Co-60	0.00E+00	N/A	U	1.41E-01	5.90E-02	3.80E+00	0.00E+00	0.0118
	Cs-134	1.93E-02	N/A	U	1.84E-01	1.20E-01	5.70E+00	3.39E-03	
	Cs-137	9.06E-02	N/A	U	1.90E-01	8.60E-02	1.10E+01	8.24E-03	
	Eu-152	0.00E+00	N/A	U	3.16E-01	1.47E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.79E-01	2.08E-01	8.00E+00	0.00E+00	
	Eu-155	4.47E-02	N/A	U	4.52E-01	2.16E-01	2.80E+02	1.60E-04	
	Am-241	0.00E+00	N/A	U	4.16E-01	1.98E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-051	Co-60	2.30E-02	N/A	U	1.19E-01	7.07E-02	3.80E+00	6.05E-03	0.0433
	Cs-134	0.00E+00	N/A	U	1.55E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	7.92E-02	N/A	U	1.67E-01	7.56E-02	1.10E+01	7.20E-03	
	Eu-152	0.00E+00	N/A	U	2.86E-01	1.33E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.67E-01	1.55E-01	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	4.27E-01	2.04E-01	2.80E+02	0.00E+00	
	Am-241	6.30E-02	N/A	U	3.83E-01	1.82E-01	2.10E+00	3.00E-02	
8300X-3-CJ-GSSX-052	Co-60	6.73E-02	N/A	U	1.41E-01	8.24E-02	3.80E+00	1.77E-02	0.0306
	Cs-134	0.00E+00	N/A	U	1.83E-01	1.22E-01	5.70E+00	0.00E+00	
	Cs-137	1.28E-01	N/A	U	1.96E-01	8.95E-02	1.10E+01	1.16E-02	
	Eu-152	0.00E+00	N/A	U	3.44E-01	1.62E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.22E-01	1.81E-01	8.00E+00	0.00E+00	
	Eu-155	3.46E-01	N/A	U	4.79E-01	2.29E-01	2.80E+02	1.24E-03	
	Am-241	0.00E+00	N/A	U	3.57E-01	1.69E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8300X-3-CJ-GSSX-053	Co-60	7.18E-02	N/A	U	1.66E-01	7.16E-02	3.80E+00	1.89E-02	0.0242
	Cs-134	0.00E+00	N/A	U	1.84E-01	1.22E-01	5.70E+00	0.00E+00	
	Cs-137	5.79E-02	N/A	U	1.85E-01	8.36E-02	1.10E+01	5.26E-03	
	Eu-152	0.00E+00	N/A	U	4.02E-01	1.90E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.29E-01	1.83E-01	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	5.19E-01	2.49E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	4.41E-01	2.10E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-054	Co-60	5.75E-02	N/A	U	1.10E-01	7.32E-02	3.80E+00	1.51E-02	0.0252
	Cs-134	0.00E+00	N/A	U	1.63E-01	1.13E-01	5.70E+00	0.00E+00	
	Cs-137	9.82E-02	N/A	U	1.89E-01	8.63E-02	1.10E+01	8.93E-03	
	Eu-152	0.00E+00	N/A	U	3.39E-01	1.59E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.07E-01	1.75E-01	8.00E+00	0.00E+00	
	Eu-155	3.17E-01	N/A	U	4.76E-01	2.31E-01	2.80E+02	1.13E-03	
	Am-241	0.00E+00	N/A	U	3.93E-01	1.87E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSA-055	Co-60	4.21E-02	N/A	U	1.36E-01	5.66E-02	3.80E+00	1.11E-02	0.0486
	Cs-134	0.00E+00	N/A	U	1.89E-01	1.20E-01	5.70E+00	0.00E+00	
	Cs-137	2.29E-01	4.66E-02		1.43E-01	6.28E-02	1.10E+01	2.08E-02	
	Eu-152	4.37E-02	N/A	U	3.77E-01	1.78E-01	8.70E+00	5.02E-03	
	Eu-154	8.87E-02	N/A	U	4.69E-01	2.03E-01	8.00E+00	1.11E-02	
	Eu-155	1.61E-01	N/A	U	5.07E-01	2.49E-01	2.80E+02	5.75E-04	
	Am-241	0.00E+00	N/A	U	4.05E-01	1.92E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-056	Co-60	4.82E-02	N/A	U	1.41E-01	7.61E-02	3.80E+00	1.27E-02	0.0318
	Cs-134	0.00E+00	N/A	U	1.79E-01	1.16E-01	5.70E+00	0.00E+00	
	Cs-137	3.93E-02	N/A	U	1.68E-01	7.58E-02	1.10E+01	3.57E-03	
	Eu-152	1.33E-01	N/A	U	3.60E-01	1.69E-01	8.70E+00	1.53E-02	
	Eu-154	0.00E+00	N/A	U	5.05E-01	2.23E-01	8.00E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-155	6.28E-02	N/A	U	4.79E-01	2.29E-01	2.80E+02	2.24E-04	
	Am-241	0.00E+00	N/A	U	3.95E-01	1.87E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-057	Co-60	3.87E-02	N/A	U	1.60E-01	6.90E-02	3.80E+00	1.02E-02	0.0152
	Cs-134	0.00E+00	N/A	U	2.07E-01	1.16E-01	5.70E+00	0.00E+00	
	Cs-137	4.59E-02	N/A	U	1.73E-01	7.74E-02	1.10E+01	4.17E-03	
	Eu-152	0.00E+00	N/A	U	3.43E-01	1.61E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.76E-01	2.07E-01	8.00E+00	0.00E+00	
	Eu-155	2.49E-01	N/A	U	4.93E-01	2.36E-01	2.80E+02	8.89E-04	
	Am-241	0.00E+00	N/A	U	3.96E-01	1.88E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-058	Co-60	5.66E-02	N/A	U	1.44E-01	7.26E-02	3.80E+00	1.49E-02	0.0365
	Cs-134	0.00E+00	N/A	U	1.83E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	1.05E-01	N/A	U	1.94E-01	8.81E-02	1.10E+01	9.55E-03	
	Eu-152	1.00E-01	N/A	U	3.79E-01	1.79E-01	8.70E+00	1.15E-02	
	Eu-154	0.00E+00	N/A	U	4.95E-01	2.16E-01	8.00E+00	0.00E+00	
	Eu-155	1.68E-01	N/A	U	4.97E-01	2.38E-01	2.80E+02	6.00E-04	
	Am-241	0.00E+00	N/A	U	4.25E-01	2.02E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-059	Co-60	0.00E+00	N/A	U	1.49E-01	8.88E-02	3.80E+00	0.00E+00	0.0818
	Cs-134	0.00E+00	N/A	U	1.82E-01	1.24E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.80E-01	8.06E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.99E-01	1.88E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.39E-01	1.87E-01	8.00E+00	0.00E+00	
	Eu-155	2.29E+01	N/A	U	5.79E-01	2.82E-01	2.80E+02	8.18E-02	
	Am-241	0.00E+00	N/A	U	4.49E-01	2.13E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-060	Co-60	0.00E+00	N/A	U	2.04E-01	9.09E-02	3.80E+00	0.00E+00	0.0150
	Cs-134	0.00E+00	N/A	U	2.72E-01	1.66E-01	5.70E+00	0.00E+00	
	Cs-137	7.03E-02	N/A	U	2.30E-01	1.02E-01	1.10E+01	6.39E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	6.18E-02	N/A	U	4.64E-01	2.16E-01	8.70E+00	7.10E-03	
	Eu-154	0.00E+00	N/A	U	5.66E-01	2.37E-01	8.00E+00	0.00E+00	
	Eu-155	4.23E-01	N/A	U	6.61E-01	3.16E-01	2.80E+02	1.51E-03	
	Am-241	0.00E+00	N/A	U	5.43E-01	2.58E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-061	Co-60	2.11E-02	N/A	U	1.38E-01	6.32E-02	3.80E+00	5.55E-03	0.0213
	Cs-134	0.00E+00	N/A	U	1.71E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	1.06E-01	N/A	U	1.85E-01	8.41E-02	1.10E+01	9.64E-03	
	Eu-152	5.00E-02	N/A	U	3.54E-01	1.66E-01	8.70E+00	5.75E-03	
	Eu-154	0.00E+00	N/A	U	4.76E-01	2.08E-01	8.00E+00	0.00E+00	
	Eu-155	1.06E-01	N/A	U	3.14E-01	1.47E-01	2.80E+02	3.79E-04	
	Am-241	0.00E+00	N/A	U	4.11E-01	1.96E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-062	Co-60	0.00E+00	N/A	U	1.48E-01	6.37E-02	3.80E+00	0.00E+00	0.0007
	Cs-134	0.00E+00	N/A	U	1.72E-01	1.16E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.71E-01	7.72E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.34E-01	1.56E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.61E-01	2.00E-01	8.00E+00	0.00E+00	
	Eu-155	2.02E-01	N/A	U	4.74E-01	2.27E-01	2.80E+02	7.21E-04	
	Am-241	0.00E+00	N/A	U	4.00E-01	1.90E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-063	Co-60	8.00E-02	N/A	U	1.23E-01	9.41E-02	3.80E+00	2.11E-02	0.0527
	Cs-134	0.00E+00	N/A	U	1.78E-01	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	2.49E-01	4.58E-02		1.33E-01	5.78E-02	1.10E+01	2.26E-02	
	Eu-152	0.00E+00	N/A	U	3.65E-01	1.71E-01	8.70E+00	0.00E+00	
	Eu-154	6.50E-02	N/A	U	4.97E-01	2.18E-01	8.00E+00	8.13E-03	
	Eu-155	2.48E-01	N/A	U	5.28E-01	2.54E-01	2.80E+02	8.86E-04	
	Am-241	0.00E+00	N/A	U	4.29E-01	2.04E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-064	Co-60	0.00E+00	N/A	U	1.77E-01	8.85E-02	3.80E+00	0.00E+00	0.0057

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	2.51E-01	1.60E-01	5.70E+00	0.00E+00	
	Cs-137	6.30E-02	N/A	U	2.80E-01	1.26E-01	1.10E+01	5.73E-03	
	Eu-152	0.00E+00	N/A	U	4.04E-01	1.85E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	5.90E-01	2.44E-01	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	5.86E-01	2.78E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	4.99E-01	2.35E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-065	Co-60	3.02E-02	N/A	U	1.41E-01	7.85E-02	3.80E+00	7.95E-03	0.0352
	Cs-134	0.00E+00	N/A	U	1.84E-01	1.30E-01	5.70E+00	0.00E+00	
	Cs-137	2.60E-01	4.55E-02		1.21E-01	5.07E-02	1.10E+01	2.36E-02	
	Eu-152	0.00E+00	N/A	U	3.91E-01	1.84E-01	8.70E+00	0.00E+00	
	Eu-154	1.77E-02	N/A	U	5.16E-01	2.24E-01	8.00E+00	2.21E-03	
	Eu-155	3.92E-01	N/A	U	5.16E-01	2.47E-01	2.80E+02	1.40E-03	
	Am-241	0.00E+00	N/A	U	4.13E-01	1.96E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-066	Co-60	0.00E+00	N/A	U	1.46E-01	6.46E-02	3.80E+00	0.00E+00	0.0117
	Cs-134	0.00E+00	N/A	U	1.80E-01	1.18E-01	5.70E+00	0.00E+00	
	Cs-137	1.19E-01	3.78E-02		1.34E-01	5.87E-02	1.10E+01	1.08E-02	
	Eu-152	0.00E+00	N/A	U	3.50E-01	1.65E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.16E-01	1.79E-01	8.00E+00	0.00E+00	
	Eu-155	2.34E-01	N/A	U	4.56E-01	2.18E-01	2.80E+02	8.36E-04	
	Am-241	0.00E+00	N/A	U	3.93E-01	1.87E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-067	Co-60	5.06E-02	N/A	U	1.52E-01	6.80E-02	3.80E+00	1.33E-02	0.0183
	Cs-134	0.00E+00	N/A	U	1.92E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	8.73E-03	N/A	U	1.58E-01	7.11E-02	1.10E+01	7.94E-04	
	Eu-152	2.90E-02	N/A	U	3.39E-01	1.59E-01	8.70E+00	3.33E-03	
	Eu-154	0.00E+00	N/A	U	5.25E-01	2.34E-01	8.00E+00	0.00E+00	
	Eu-155	2.40E-01	N/A	U	4.77E-01	2.29E-01	2.80E+02	8.57E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	3.71E-01	1.76E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-068	Co-60	0.00E+00	N/A	U	1.24E-01	6.27E-02	3.80E+00	0.00E+00	0.0225
	Cs-134	0.00E+00	N/A	U	1.86E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.07E-01	3.41E-02		1.20E-01	5.25E-02	1.10E+01	9.73E-03	
	Eu-152	4.19E-02	N/A	U	3.24E-01	1.52E-01	8.70E+00	4.82E-03	
	Eu-154	5.64E-02	N/A	U	3.46E-01	1.46E-01	8.00E+00	7.05E-03	
	Eu-155	2.62E-01	N/A	U	4.64E-01	2.23E-01	2.80E+02	9.36E-04	
	Am-241	0.00E+00	N/A	U	3.73E-01	1.77E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-069	Co-60	1.72E-02	N/A	U	1.09E-01	4.57E-02	3.80E+00	4.53E-03	0.0128
	Cs-134	0.00E+00	N/A	U	1.48E-01	9.77E-02	5.70E+00	0.00E+00	
	Cs-137	8.31E-02	N/A	U	1.41E-01	6.28E-02	1.10E+01	7.55E-03	
	Eu-152	0.00E+00	N/A	U	3.06E-01	1.44E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.83E-01	1.65E-01	8.00E+00	0.00E+00	
	Eu-155	1.91E-01	N/A	U	4.10E-01	1.96E-01	2.80E+02	6.82E-04	
	Am-241	0.00E+00	N/A	U	3.47E-01	1.65E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-070	Co-60	4.43E-03	N/A	U	1.31E-01	5.55E-02	3.80E+00	1.17E-03	0.0589
	Cs-134	0.00E+00	N/A	U	1.83E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	3.27E-01	N/A	U	2.55E-01	1.19E-01	1.10E+01	2.97E-02	
	Eu-152	0.00E+00	N/A	U	3.36E-01	1.58E-01	8.70E+00	0.00E+00	
	Eu-154	2.14E-01	N/A	U	5.01E-01	2.22E-01	8.00E+00	2.68E-02	
	Eu-155	3.62E-01	N/A	U	4.57E-01	2.42E-01	2.80E+02	1.29E-03	
	Am-241	0.00E+00	N/A	U	3.98E-01	1.90E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSA-071	Co-60	1.11E-02	N/A	U	1.40E-01	5.94E-02	3.80E+00	2.92E-03	0.0100
	Cs-134	0.00E+00	N/A	U	1.86E-01	1.33E-01	5.70E+00	0.00E+00	
	Cs-137	5.49E-02	3.08E-02		1.21E-01	5.13E-02	1.10E+01	4.99E-03	
	Eu-152	0.00E+00	N/A	U	3.83E-01	1.80E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	1.04E-02	N/A	U	5.03E-01	2.20E-01	8.00E+00	1.30E-03	
	Eu-155	2.11E-01	N/A	U	5.34E-01	2.57E-01	2.80E+02	7.54E-04	
	Am-241	0.00E+00	N/A	U	4.10E-01	1.95E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-072	Co-60	7.69E-02	N/A	U	1.41E-01	6.76E-02	3.80E+00	2.02E-02	0.0546
	Cs-134	0.00E+00	N/A	U	1.75E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	2.01E-01	3.66E-02		1.01E-01	4.27E-02	1.10E+01	1.83E-02	
	Eu-152	1.16E-01	N/A	U	3.40E-01	1.60E-01	8.70E+00	1.33E-02	
	Eu-154	7.78E-04	N/A	U	3.83E-01	1.65E-01	8.00E+00	9.73E-05	
	Eu-155	2.07E-01	N/A	U	4.71E-01	2.26E-01	2.80E+02	7.39E-04	
	Am-241	3.97E-03	N/A	U	4.03E-01	1.92E-01	2.10E+00	1.89E-03	
8300X-3-CJ-GSSX-073	Co-60	4.09E-02	N/A	U	1.27E-01	5.14E-02	3.80E+00	1.08E-02	0.0137
	Cs-134	0.00E+00	N/A	U	1.80E-01	1.19E-01	5.70E+00	0.00E+00	
	Cs-137	1.27E-02	N/A	U	1.55E-01	6.84E-02	1.10E+01	1.15E-03	
	Eu-152	0.00E+00	N/A	U	3.44E-01	1.61E-01	8.70E+00	0.00E+00	
	Eu-154	5.16E-03	N/A	U	4.70E-01	2.02E-01	8.00E+00	6.45E-04	
	Eu-155	3.15E-01	N/A	U	5.16E-01	2.47E-01	2.80E+02	1.13E-03	
	Am-241	0.00E+00	N/A	U	4.08E-01	1.94E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-074	Co-60	0.00E+00	N/A	U	1.35E-01	6.43E-02	3.80E+00	0.00E+00	0.0088
	Cs-134	0.00E+00	N/A	U	1.54E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	1.32E-02	N/A	U	1.57E-01	7.09E-02	1.10E+01	1.20E-03	
	Eu-152	0.00E+00	N/A	U	3.23E-01	1.51E-01	8.70E+00	0.00E+00	
	Eu-154	5.03E-02	N/A	U	3.95E-01	1.70E-01	8.00E+00	6.29E-03	
	Eu-155	3.68E-01	N/A	U	4.68E-01	2.25E-01	2.80E+02	1.31E-03	
	Am-241	0.00E+00	N/A	U	3.79E-01	1.80E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-075	Co-60	1.81E-02	N/A	U	1.33E-01	6.42E-02	3.80E+00	4.76E-03	0.0216
	Cs-134	0.00E+00	N/A	U	1.71E-01	1.09E-01	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	1.78E-01	N/A	U	2.19E-01	1.01E-01	1.10E+01	1.62E-02	
	Eu-152	0.00E+00	N/A	U	3.40E-01	1.60E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.22E-01	1.82E-01	8.00E+00	0.00E+00	
	Eu-155	1.72E-01	N/A	U	4.70E-01	2.25E-01	2.80E+02	6.14E-04	
	Am-241	0.00E+00	N/A	U	3.67E-01	1.73E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-076	Co-60	4.69E-02	N/A	U	1.60E-01	7.08E-02	3.80E+00	1.23E-02	0.0472
	Cs-134	0.00E+00	N/A	U	1.66E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	5.53E-02	N/A	U	2.01E-01	9.20E-02	1.10E+01	5.03E-03	
	Eu-152	6.16E-02	N/A	U	3.58E-01	1.68E-01	8.70E+00	7.08E-03	
	Eu-154	0.00E+00	N/A	U	3.94E-01	1.67E-01	8.00E+00	0.00E+00	
	Eu-155	1.38E-02	N/A	U	4.66E-01	2.23E-01	2.80E+02	4.93E-05	
	Am-241	4.76E-02	N/A	U	3.93E-01	1.87E-01	2.10E+00	2.27E-02	
8300X-3-CJ-GSSX-077	Co-60	3.48E-02	N/A	U	1.18E-01	7.78E-02	3.80E+00	9.16E-03	0.0474
	Cs-134	0.00E+00	N/A	U	1.75E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.67E-01	7.55E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.44E-01	1.62E-01	8.70E+00	0.00E+00	
	Eu-154	2.99E-01	N/A	U	4.51E-01	1.97E-01	8.00E+00	3.74E-02	
	Eu-155	2.42E-01	N/A	U	2.74E-01	1.27E-01	2.80E+02	8.64E-04	
	Am-241	0.00E+00	N/A	U	3.52E-01	1.66E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-078	Co-60	3.72E-02	N/A	U	1.28E-01	7.46E-02	3.80E+00	9.79E-03	0.0293
	Cs-134	0.00E+00	N/A	U	1.80E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	6.57E-02	N/A	U	1.59E-01	7.09E-02	1.10E+01	5.97E-03	
	Eu-152	1.12E-01	N/A	U	3.52E-01	1.66E-01	8.70E+00	1.29E-02	
	Eu-154	0.00E+00	N/A	U	3.56E-01	1.48E-01	8.00E+00	0.00E+00	
	Eu-155	1.74E-01	N/A	U	4.64E-01	2.22E-01	2.80E+02	6.21E-04	
	Am-241	0.00E+00	N/A	U	4.01E-01	1.90E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8300X-3-CJ-GSSA-079	Co-60	5.23E-02	N/A	U	1.10E-01	7.59E-02	3.80E+00	1.38E-02	0.0242
	Cs-134	0.00E+00	N/A	U	1.87E-01	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	8.85E-02	3.21E-02		1.15E-01	4.91E-02	1.10E+01	8.05E-03	
	Eu-152	9.27E-03	N/A	U	3.57E-01	1.68E-01	8.70E+00	1.07E-03	
	Eu-154	0.00E+00	N/A	U	4.09E-01	1.74E-01	8.00E+00	0.00E+00	
	Eu-155	3.84E-01	N/A	U	5.02E-01	2.41E-01	2.80E+02	1.37E-03	
	Am-241	0.00E+00	N/A	U	3.91E-01	1.85E-01	2.10E+00	0.00E+00	
8300X-3-CJ-GSSX-080	Co-60	3.75E-02	N/A	U	1.14E-01	4.67E-02	3.80E+00	9.87E-03	0.0617
	Cs-134	0.00E+00	N/A	U	2.09E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	1.07E-01	3.47E-02		1.23E-01	5.33E-02	1.10E+01	9.73E-03	
	Eu-152	0.00E+00	N/A	U	3.47E-01	1.64E-01	8.70E+00	0.00E+00	
	Eu-154	1.81E-01	N/A	U	4.25E-01	1.84E-01	8.00E+00	2.26E-02	
	Eu-155	8.22E-02	N/A	U	4.64E-01	2.23E-01	2.80E+02	2.94E-04	
	Am-241	4.03E-02	N/A	U	3.90E-01	1.86E-01	2.10E+00	1.92E-02	
8300X-3-CJ-GSSX-081	Co-60	0.00E+00	N/A	U	1.03E-01	7.04E-02	3.80E+00	0.00E+00	0.0097
	Cs-134	0.00E+00	N/A	U	1.52E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	9.18E-02	3.06E-02		1.08E-01	4.69E-02	1.10E+01	8.35E-03	
	Eu-152	4.62E-03	N/A	U	3.44E-01	1.63E-01	8.70E+00	5.31E-04	
	Eu-154	0.00E+00	N/A	U	4.00E-01	1.74E-01	8.00E+00	0.00E+00	
	Eu-155	2.37E-01	N/A	U	4.19E-01	2.31E-01	2.80E+02	8.46E-04	
	Am-241	0.00E+00	N/A	U	3.73E-01	1.77E-01	2.10E+00	0.00E+00	

Table 2-59 8300 Summary Statistics

Combined					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	2.62E-01	2.89E-02	2.04E-02	3.40E-02
Cs-134	0.00E+00	2.96E-02	6.11E-04	0.00E+00	3.71E-03
Cs-137	0.00E+00	3.27E-01	1.07E-01	9.56E-02	6.72E-02
Eu-152	0.00E+00	1.51E-01	2.49E-02	0.00E+00	3.97E-02
Eu-154	0.00E+00	2.99E-01	3.36E-02	0.00E+00	5.91E-02
Eu-155	0.00E+00	2.29E+01	4.37E-01	2.14E-01	2.24E+00
Am-241	0.00E+00	1.44E-01	6.19E-03	0.00E+00	2.13E-02
Random					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	2.62E-01	3.17E-02	1.90E-02	4.43E-02
Cs-134	0.00E+00	2.96E-02	1.04E-03	0.00E+00	5.00E-03
Cs-137	0.00E+00	2.64E-01	1.10E-01	1.00E-01	5.77E-02
Eu-152	0.00E+00	1.18E-01	2.15E-02	0.00E+00	3.64E-02
Eu-154	0.00E+00	2.20E-01	3.38E-02	0.00E+00	5.21E-02
Eu-155	2.39E-02	5.12E-01	2.20E-01	2.07E-01	1.07E-01
Am-241	0.00E+00	1.44E-01	7.91E-03	0.00E+00	2.78E-02
Judgmental					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	8.30E-02	3.01E-02	3.20E-02	2.54E-02
Cs-134	0.00E+00	1.93E-02	3.78E-04	0.00E+00	2.70E-03
Cs-137	0.00E+00	3.27E-01	1.02E-01	8.85E-02	7.71E-02
Eu-152	0.00E+00	1.51E-01	3.02E-02	0.00E+00	4.35E-02
Eu-154	0.00E+00	2.99E-01	2.96E-02	0.00E+00	6.28E-02
Eu-155	0.00E+00	2.29E+01	6.51E-01	2.14E-01	3.18E+00
Am-241	0.00E+00	6.87E-02	5.99E-03	0.00E+00	1.68E-02

Total Number of Samples	
Random	42
Judgmental	51
QC	10

Random	
SOF >0.5	0
Maximum SOF	0.0980
Minimum SOF	0.0042

Judgmental	
SOF >0.5	0
Maximum SOF	0.0818
Minimum SOF	0.0007

Figure 2-32 Survey Unit 8400 Random Sample Locations

Figure 2-33 Survey Unit 8400 Judgmental Sample and Scan Locations

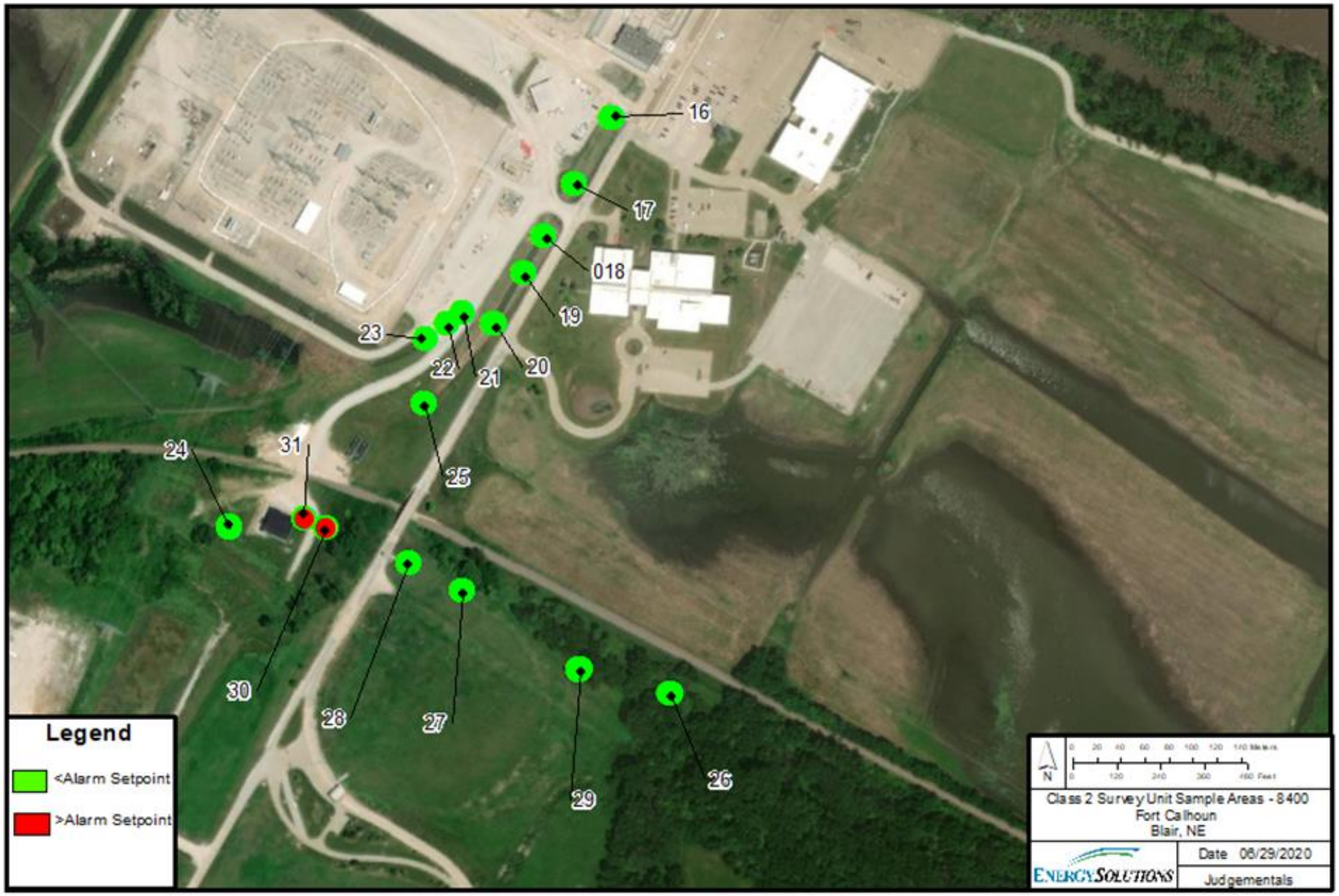


Table 2-60 8400 Gamma Spectroscopy Results for Random Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8400X-2-CR-GSSX-001	Co-60	2.67E-02	N/A	U	1.10E-01	5.51E-02	3.80E+00	7.03E-03	0.0291
	Cs-134	0.00E+00	N/A	U	1.34E-01	9.76E-02	5.70E+00	0.00E+00	
	Cs-137	5.97E-02	N/A	U	1.26E-01	5.61E-02	1.10E+01	5.43E-03	
	Eu-152	1.37E-01	N/A	U	3.34E-01	1.58E-01	8.70E+00	1.57E-02	
	Eu-154	0.00E+00	N/A	U	3.61E-01	1.57E-01	8.00E+00	0.00E+00	
	Eu-155	2.44E-01	N/A	U	4.32E-01	2.08E-01	2.80E+02	8.71E-04	
	Am-241	0.00E+00	N/A	U	3.54E-01	1.69E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSA-002	Co-60	1.18E-03	N/A	U	9.91E-02	5.18E-02	3.80E+00	3.11E-04	0.0003
	Cs-134	0.00E+00	N/A	U	1.32E-01	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.05E-01	4.66E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.11E-01	1.47E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.91E-01	1.25E-01	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	4.09E-01	1.96E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	3.54E-01	1.68E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSX-003	Co-60	2.72E-02	N/A	U	1.45E-01	6.19E-02	3.80E+00	7.16E-03	0.0216
	Cs-134	0.00E+00	N/A	U	1.75E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	1.37E-01	N/A	U	1.85E-01	8.43E-02	1.10E+01	1.25E-02	
	Eu-152	0.00E+00	N/A	U	3.46E-01	1.63E-01	8.70E+00	0.00E+00	
	Eu-154	1.13E-02	N/A	U	4.22E-01	1.82E-01	8.00E+00	1.41E-03	
	Eu-155	1.59E-01	N/A	U	4.70E-01	2.25E-01	2.80E+02	5.68E-04	
	Am-241	0.00E+00	N/A	U	3.83E-01	1.82E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSX-004	Co-60	1.06E-02	N/A	U	1.35E-01	5.71E-02	3.80E+00	2.79E-03	0.0416
	Cs-134	0.00E+00	N/A	U	1.60E-01	9.89E-02	5.70E+00	0.00E+00	
	Cs-137	2.49E-01	4.70E-02		1.44E-01	6.42E-02	1.10E+01	2.26E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	2.89E-03	N/A	U	3.53E-01	1.67E-01	8.70E+00	3.32E-04	
	Eu-154	0.00E+00	N/A	U	5.06E-01	2.25E-01	8.00E+00	0.00E+00	
	Eu-155	1.52E-01	N/A	U	4.94E-01	2.38E-01	2.80E+02	5.43E-04	
	Am-241	3.22E-02	N/A	U	4.04E-01	1.93E-01	2.10E+00	1.53E-02	
8400X-2-CR-GSSX-005	Co-60	2.95E-02	N/A	U	9.76E-02	6.24E-02	3.80E+00	7.76E-03	0.0152
	Cs-134	0.00E+00	N/A	U	1.79E-01	1.29E-01	5.70E+00	0.00E+00	
	Cs-137	6.91E-02	N/A	U	1.73E-01	7.83E-02	1.10E+01	6.28E-03	
	Eu-152	0.00E+00	N/A	U	3.39E-01	1.60E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.46E-01	1.95E-01	8.00E+00	0.00E+00	
	Eu-155	3.20E-01	N/A	U	4.84E-01	2.37E-01	2.80E+02	1.14E-03	
	Am-241	0.00E+00	N/A	U	4.06E-01	1.93E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSX-006	Co-60	1.07E-02	N/A	U	1.12E-01	7.67E-02	3.80E+00	2.82E-03	0.0457
	Cs-134	0.00E+00	N/A	U	1.59E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	1.03E-01	3.08E-02		1.02E-01	4.24E-02	1.10E+01	9.36E-03	
	Eu-152	1.04E-01	N/A	U	4.01E-01	1.89E-01	8.70E+00	1.20E-02	
	Eu-154	1.59E-01	N/A	U	4.35E-01	1.87E-01	8.00E+00	1.99E-02	
	Eu-155	4.87E-01	N/A	U	5.20E-01	2.58E-01	2.80E+02	1.74E-03	
	Am-241	0.00E+00	N/A	U	4.05E-01	1.92E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSA-007	Co-60	3.60E-02	N/A	U	1.33E-01	5.50E-02	3.80E+00	9.47E-03	0.0108
	Cs-134	0.00E+00	N/A	U	1.92E-01	1.19E-01	5.70E+00	0.00E+00	
	Cs-137	9.05E-03	N/A	U	1.58E-01	6.99E-02	1.10E+01	8.23E-04	
	Eu-152	0.00E+00	N/A	U	3.43E-01	1.61E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.21E-01	1.79E-01	8.00E+00	0.00E+00	
	Eu-155	1.49E-01	N/A	U	4.91E-01	2.35E-01	2.80E+02	5.32E-04	
	Am-241	0.00E+00	N/A	U	3.97E-01	1.88E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSX-008	Co-60	2.29E-02	N/A	U	7.77E-02	3.75E-02	3.80E+00	6.03E-03	0.0180

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	9.02E-02	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	3.53E-02	N/A	U	1.00E-01	4.47E-02	1.10E+01	3.21E-03	
	Eu-152	1.80E-02	N/A	U	3.04E-01	1.44E-01	8.70E+00	2.07E-03	
	Eu-154	5.34E-02	N/A	U	2.61E-01	1.12E-01	8.00E+00	6.68E-03	
	Eu-155	0.00E+00	N/A	U	3.59E-01	1.72E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	2.97E-01	1.41E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSX-009	Co-60	1.89E-02	N/A	U	1.00E-01	4.32E-02	3.80E+00	4.97E-03	0.0460
	Cs-134	9.04E-02	N/A	U	1.31E-01	7.03E-02	5.70E+00	1.59E-02	
	Cs-137	5.28E-02	N/A	U	1.19E-01	5.38E-02	1.10E+01	4.80E-03	
	Eu-152	8.80E-02	N/A	U	3.00E-01	1.42E-01	8.70E+00	1.01E-02	
	Eu-154	0.00E+00	N/A	U	3.79E-01	1.70E-01	8.00E+00	0.00E+00	
	Eu-155	1.16E-01	N/A	U	3.68E-01	1.76E-01	2.80E+02	4.14E-04	
	Am-241	2.07E-02	N/A	U	3.67E-01	1.75E-01	2.10E+00	9.86E-03	
8400X-2-CR-GSSX-010	Co-60	0.00E+00	N/A	U	1.15E-01	4.74E-02	3.80E+00	0.00E+00	0.0106
	Cs-134	0.00E+00	N/A	U	1.49E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	1.26E-02	N/A	U	1.71E-01	7.75E-02	1.10E+01	1.15E-03	
	Eu-152	7.42E-02	N/A	U	3.36E-01	1.58E-01	8.70E+00	8.53E-03	
	Eu-154	0.00E+00	N/A	U	3.93E-01	1.69E-01	8.00E+00	0.00E+00	
	Eu-155	2.46E-01	N/A	U	4.63E-01	2.30E-01	2.80E+02	8.79E-04	
	Am-241	0.00E+00	N/A	U	3.88E-01	1.85E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSX-011	Co-60	4.42E-02	N/A	U	1.32E-01	5.64E-02	3.80E+00	1.16E-02	0.0187
	Cs-134	0.00E+00	N/A	U	1.60E-01	1.28E-01	5.70E+00	0.00E+00	
	Cs-137	7.77E-02	N/A	U	1.62E-01	7.26E-02	1.10E+01	7.06E-03	
	Eu-152	0.00E+00	N/A	U	3.35E-01	1.57E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.15E-01	1.78E-01	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	4.74E-01	2.27E-01	2.80E+02	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	4.10E-01	1.95E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSX-012	Co-60	2.76E-02	N/A	U	1.21E-01	6.50E-02	3.80E+00	7.26E-03	0.0458
	Cs-134	0.00E+00	N/A	U	1.48E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.31E-01	5.83E-02	1.10E+01	0.00E+00	
	Eu-152	7.44E-02	N/A	U	3.45E-01	1.63E-01	8.70E+00	8.55E-03	
	Eu-154	1.29E-01	N/A	U	4.37E-01	1.94E-01	8.00E+00	1.61E-02	
	Eu-155	2.32E-01	N/A	U	4.56E-01	2.19E-01	2.80E+02	8.29E-04	
	Am-241	2.74E-02	N/A	U	4.25E-01	2.03E-01	2.10E+00	1.30E-02	
8400X-2-CR-GSSX-013	Co-60	4.99E-04	N/A	U	1.37E-01	5.87E-02	3.80E+00	1.31E-04	0.0211
	Cs-134	0.00E+00	N/A	U	1.59E-01	9.75E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.58E-01	7.17E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.99E-01	1.40E-01	8.70E+00	0.00E+00	
	Eu-154	1.62E-01	N/A	U	4.68E-01	2.07E-01	8.00E+00	2.03E-02	
	Eu-155	2.05E-01	N/A	U	4.50E-01	2.16E-01	2.80E+02	7.32E-04	
	Am-241	0.00E+00	N/A	U	3.61E-01	1.71E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSX-014	Co-60	3.63E-02	N/A	U	9.59E-02	4.09E-02	3.80E+00	9.55E-03	0.0334
	Cs-134	0.00E+00	N/A	U	1.15E-01	9.78E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.12E-01	5.05E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.76E-01	1.30E-01	8.70E+00	0.00E+00	
	Eu-154	2.92E-02	N/A	U	2.94E-01	1.27E-01	8.00E+00	3.65E-03	
	Eu-155	7.69E-02	N/A	U	3.78E-01	1.82E-01	2.80E+02	2.75E-04	
	Am-241	4.19E-02	N/A	U	3.57E-01	1.71E-01	2.10E+00	2.00E-02	
8400X-2-CR-GSSX-015	Co-60	3.44E-02	N/A	U	1.44E-01	6.84E-02	3.80E+00	9.05E-03	0.0288
	Cs-134	0.00E+00	N/A	U	1.78E-01	1.17E-01	5.70E+00	0.00E+00	
	Cs-137	1.56E-02	2.33E-02		1.02E-01	4.26E-02	1.10E+01	1.42E-03	
	Eu-152	7.74E-02	N/A	U	3.55E-01	1.67E-01	8.70E+00	8.90E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	6.75E-02	N/A	U	4.51E-01	4.03E-01	8.00E+00	8.44E-03	
	Eu-155	2.68E-01	N/A	U	3.35E-01	1.57E-01	2.80E+02	9.57E-04	
	Am-241	0.00E+00	N/A	U	4.08E-01	1.94E-01	2.10E+00	0.00E+00	

Table 2-61 8400 Gamma Spectroscopy Results for QC Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8400X-2-CR-GSSB-002	Co-60	1.77E-02	N/A	U	8.96E-02	4.32E-02	3.80E+00	4.66E-03	0.0092
	Cs-134	0.00E+00	N/A	U	1.16E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	4.79E-02	N/A	U	1.21E-01	5.45E-02	1.10E+01	4.35E-03	
	Eu-152	0.00E+00	N/A	U	2.97E-01	1.41E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.95E-01	1.27E-01	8.00E+00	0.00E+00	
	Eu-155	6.53E-02	N/A	U	4.04E-01	1.94E-01	2.80E+02	2.33E-04	
	Am-241	0.00E+00	N/A	U	3.60E-01	1.72E-01	2.10E+00	0.00E+00	
8400X-2-CQ-GSSX-003	Co-60	0.00E+00	N/A	U	9.71E-02	6.36E-02	3.80E+00	0.00E+00	0.0095
	Cs-134	0.00E+00	N/A	U	1.80E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	4.38E-02	3.30E-02		1.34E-01	5.90E-02	1.10E+01	3.98E-03	
	Eu-152	4.05E-02	N/A	U	3.53E-01	1.67E-01	8.70E+00	4.66E-03	
	Eu-154	0.00E+00	N/A	U	4.27E-01	1.85E-01	8.00E+00	0.00E+00	
	Eu-155	2.47E-01	N/A	U	4.71E-01	2.26E-01	2.80E+02	8.82E-04	
	Am-241	0.00E+00	N/A	U	3.90E-01	1.86E-01	2.10E+00	0.00E+00	
8400X-2-CR-GSSB-007	Co-60	1.67E-02	N/A	U	9.38E-02	7.62E-02	3.80E+00	4.39E-03	0.0399
	Cs-134	0.00E+00	N/A	U	2.07E-01	1.26E-01	5.70E+00	0.00E+00	
	Cs-137	8.95E-03	N/A	U	1.67E-01	7.49E-02	1.10E+01	8.14E-04	
	Eu-152	7.64E-02	N/A	U	3.46E-01	1.62E-01	8.70E+00	8.78E-03	
	Eu-154	3.71E-02	N/A	U	4.60E-01	7.77E-01	8.00E+00	4.64E-03	
	Eu-155	2.49E-01	N/A	U	5.14E-01	2.47E-01	2.80E+02	8.89E-04	
	Am-241	4.28E-02	N/A	U	4.47E-01	2.13E-01	2.10E+00	2.04E-02	
8400X-2-CQ-GSSX-011	Co-60	6.66E-02	N/A	U	1.31E-01	7.52E-02	3.80E+00	1.75E-02	0.0190
	Cs-134	0.00E+00	N/A	U	2.09E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	9.99E-03	N/A	U	1.60E-01	7.19E-02	1.10E+01	9.08E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.20E-01	1.50E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.98E-01	1.70E-01	8.00E+00	0.00E+00	
	Eu-155	1.72E-01	N/A	U	4.71E-01	2.26E-01	2.80E+02	6.14E-04	
	Am-241	0.00E+00	N/A	U	3.74E-01	1.77E-01	2.10E+00	0.00E+00	

Table 2-62 8400 Gamma Spectroscopy Results for Judgmental Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8400X-2-CJ-GSSX-016	Co-60	4.59E-03	N/A	U	1.16E-01	4.96E-02	3.80E+00	1.21E-03	0.0092
	Cs-134	0.00E+00	N/A	U	1.33E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	1.91E-02	N/A	U	1.30E-01	5.83E-02	1.10E+01	1.74E-03	
	Eu-152	4.89E-02	N/A	U	3.32E-01	1.57E-01	8.70E+00	5.62E-03	
	Eu-154	0.00E+00	N/A	U	3.56E-01	1.54E-01	8.00E+00	0.00E+00	
	Eu-155	1.82E-01	N/A	U	4.29E-01	2.06E-01	2.80E+02	6.50E-04	
	Am-241	0.00E+00	N/A	U	3.63E-01	1.73E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-017	Co-60	2.25E-02	N/A	U	1.00E-01	5.07E-02	3.80E+00	5.92E-03	0.0171
	Cs-134	0.00E+00	N/A	U	1.34E-01	9.90E-02	5.70E+00	0.00E+00	
	Cs-137	6.19E-02	N/A	U	1.27E-01	5.71E-02	1.10E+01	5.63E-03	
	Eu-152	4.44E-02	N/A	U	2.98E-01	1.41E-01	8.70E+00	5.10E-03	
	Eu-154	0.00E+00	N/A	U	3.39E-01	1.48E-01	8.00E+00	0.00E+00	
	Eu-155	1.38E-01	N/A	U	3.73E-01	1.79E-01	2.80E+02	4.93E-04	
	Am-241	0.00E+00	N/A	U	3.32E-01	1.58E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-018	Co-60	2.28E-02	N/A	U	9.94E-02	4.17E-02	3.80E+00	6.00E-03	0.0313
	Cs-134	0.00E+00	N/A	U	1.33E-01	9.97E-02	5.70E+00	0.00E+00	
	Cs-137	2.37E-02	N/A	U	1.33E-01	6.02E-02	1.10E+01	2.15E-03	
	Eu-152	7.60E-02	N/A	U	3.15E-01	1.49E-01	8.70E+00	8.74E-03	
	Eu-154	1.08E-01	N/A	U	3.25E-01	1.40E-01	8.00E+00	1.35E-02	
	Eu-155	2.60E-01	N/A	U	3.81E-01	1.83E-01	2.80E+02	9.29E-04	
	Am-241	0.00E+00	N/A	U	3.37E-01	1.60E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-019	Co-60	1.68E-02	N/A	U	1.16E-01	7.19E-02	3.80E+00	4.42E-03	0.0296
	Cs-134	0.00E+00	N/A	U	1.53E-01	1.19E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.61E-01	7.17E-02	1.10E+01	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.16E-01	1.48E-01	8.70E+00	0.00E+00	
	Eu-154	1.94E-01	N/A	U	4.42E-01	1.91E-01	8.00E+00	2.43E-02	
	Eu-155	2.55E-01	N/A	U	4.47E-01	2.14E-01	2.80E+02	9.11E-04	
	Am-241	0.00E+00	N/A	U	3.53E-01	1.67E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-020	Co-60	6.71E-02	N/A	U	1.34E-01	8.24E-02	3.80E+00	1.77E-02	0.0323
	Cs-134	0.00E+00	N/A	U	1.65E-01	1.22E-01	5.70E+00	0.00E+00	
	Cs-137	1.59E-02	N/A	U	1.53E-01	6.79E-02	1.10E+01	1.45E-03	
	Eu-152	0.00E+00	N/A	U	3.61E-01	1.69E-01	8.70E+00	0.00E+00	
	Eu-154	1.02E-01	N/A	U	4.53E-01	1.96E-01	8.00E+00	1.28E-02	
	Eu-155	1.31E-01	N/A	U	4.75E-01	2.28E-01	2.80E+02	4.68E-04	
	Am-241	0.00E+00	N/A	U	3.75E-01	1.77E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-021	Co-60	8.52E-02	N/A	U	1.24E-01	7.22E-02	3.80E+00	2.24E-02	0.0253
	Cs-134	0.00E+00	N/A	U	1.69E-01	1.35E-01	5.70E+00	0.00E+00	
	Cs-137	3.14E-02	N/A	U	1.67E-01	7.54E-02	1.10E+01	2.85E-03	
	Eu-152	0.00E+00	N/A	U	3.40E-01	1.59E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.18E-01	1.80E-01	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	4.59E-01	2.20E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	4.03E-01	1.91E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-022	Co-60	9.92E-02	N/A	U	1.27E-01	6.74E-02	3.80E+00	2.61E-02	0.0296
	Cs-134	0.00E+00	N/A	U	1.34E-01	1.24E-01	5.70E+00	0.00E+00	
	Cs-137	2.73E-02	N/A	U	1.48E-01	6.68E-02	1.10E+01	2.48E-03	
	Eu-152	8.93E-03	N/A	U	3.19E-01	1.50E-01	8.70E+00	1.03E-03	
	Eu-154	0.00E+00	N/A	U	3.70E-01	1.60E-01	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	4.43E-01	2.13E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	3.57E-01	1.69E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-023	Co-60	5.73E-02	N/A	U	1.08E-01	6.49E-02	3.80E+00	1.51E-02	0.0175

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.59E-01	1.29E-01	5.70E+00	0.00E+00	
	Cs-137	2.17E-02	N/A	U	1.50E-01	6.73E-02	1.10E+01	1.97E-03	
	Eu-152	0.00E+00	N/A	U	3.43E-01	1.62E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.51E-01	1.98E-01	8.00E+00	0.00E+00	
	Eu-155	1.12E-01	N/A	U	4.23E-01	2.18E-01	2.80E+02	4.00E-04	
	Am-241	0.00E+00	N/A	U	3.41E-01	1.61E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-024	Co-60	9.03E-03	N/A	U	1.26E-01	7.00E-02	3.80E+00	2.38E-03	0.0240
	Cs-134	0.00E+00	N/A	U	1.60E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	5.00E-02	N/A	U	1.52E-01	6.82E-02	1.10E+01	4.55E-03	
	Eu-152	0.00E+00	N/A	U	3.30E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	1.35E-01	N/A	U	4.86E-01	2.16E-01	8.00E+00	1.69E-02	
	Eu-155	5.11E-02	N/A	U	2.92E-01	1.36E-01	2.80E+02	1.83E-04	
	Am-241	0.00E+00	N/A	U	3.95E-01	1.88E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-025	Co-60	3.55E-02	N/A	U	9.63E-02	7.81E-02	3.80E+00	9.34E-03	0.0119
	Cs-134	0.00E+00	N/A	U	1.77E-01	1.32E-01	5.70E+00	0.00E+00	
	Cs-137	2.14E-02	N/A	U	1.72E-01	7.73E-02	1.10E+01	1.95E-03	
	Eu-152	0.00E+00	N/A	U	3.41E-01	1.59E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.61E-01	1.99E-01	8.00E+00	0.00E+00	
	Eu-155	1.76E-01	N/A	U	4.94E-01	2.37E-01	2.80E+02	6.29E-04	
	Am-241	0.00E+00	N/A	U	4.02E-01	1.91E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-026	Co-60	0.00E+00	N/A	U	1.11E-01	6.05E-02	3.80E+00	0.00E+00	0.0068
	Cs-134	0.00E+00	N/A	U	1.43E-01	9.21E-02	5.70E+00	0.00E+00	
	Cs-137	6.42E-02	N/A	U	1.49E-01	6.73E-02	1.10E+01	5.84E-03	
	Eu-152	0.00E+00	N/A	U	2.90E-01	1.36E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.29E-01	1.40E-01	8.00E+00	0.00E+00	
	Eu-155	2.74E-01	N/A	U	3.99E-01	1.98E-01	2.80E+02	9.79E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	3.15E-01	1.48E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-027	Co-60	5.59E-03	N/A	U	1.24E-01	5.14E-02	3.80E+00	1.47E-03	0.0298
	Cs-134	2.96E-03	N/A	U	1.74E-01	1.16E-01	5.70E+00	5.19E-04	
	Cs-137	3.35E-02	2.85E-02		1.18E-01	5.04E-02	1.10E+01	3.05E-03	
	Eu-152	4.40E-02	N/A	U	3.50E-01	1.64E-01	8.70E+00	5.06E-03	
	Eu-154	1.54E-01	N/A	U	4.90E-01	2.15E-01	8.00E+00	1.93E-02	
	Eu-155	1.14E-01	N/A	U	4.38E-01	2.09E-01	2.80E+02	4.07E-04	
	Am-241	0.00E+00	N/A	U	3.49E-01	1.64E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-028	Co-60	1.31E-02	N/A	U	1.13E-01	5.59E-02	3.80E+00	3.45E-03	0.0191
	Cs-134	0.00E+00	N/A	U	1.57E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	7.75E-02	N/A	U	1.68E-01	7.63E-02	1.10E+01	7.05E-03	
	Eu-152	0.00E+00	N/A	U	3.18E-01	1.49E-01	8.70E+00	0.00E+00	
	Eu-154	6.26E-02	N/A	U	4.39E-01	1.92E-01	8.00E+00	7.83E-03	
	Eu-155	2.24E-01	N/A	U	4.42E-01	2.21E-01	2.80E+02	8.00E-04	
	Am-241	0.00E+00	N/A	U	3.81E-01	1.81E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-029	Co-60	3.42E-02	N/A	U	1.34E-01	5.97E-02	3.80E+00	9.00E-03	0.0188
	Cs-134	0.00E+00	N/A	U	1.79E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	6.02E-02	N/A	U	1.64E-01	7.40E-02	1.10E+01	5.47E-03	
	Eu-152	3.76E-02	N/A	U	3.23E-01	1.51E-01	8.70E+00	4.32E-03	
	Eu-154	0.00E+00	N/A	U	3.68E-01	1.55E-01	8.00E+00	0.00E+00	
	Eu-155	1.18E-02	N/A	U	4.07E-01	1.94E-01	2.80E+02	4.21E-05	
	Am-241	0.00E+00	N/A	U	3.65E-01	1.73E-01	2.10E+00	0.00E+00	
8400X-2-CJ-GSSX-030	Co-60	3.10E-02	N/A	U	1.29E-01	5.73E-02	3.80E+00	8.16E-03	0.0607
	Cs-134	0.00E+00	N/A	U	1.70E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	3.40E-02	N/A	U	1.43E-01	6.41E-02	1.10E+01	3.09E-03	
	Eu-152	0.00E+00	N/A	U	3.42E-01	1.61E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	3.99E-01	1.74E-01	8.00E+00	0.00E+00	
	Eu-155	2.53E-01	N/A	U	4.72E-01	2.27E-01	2.80E+02	9.04E-04	
	Am-241	1.02E-01	N/A	U	4.13E-01	1.97E-01	2.10E+00	4.86E-02	
8400X-2-CJ-GSSX-031	Co-60	3.12E-02	N/A	U	1.36E-01	7.30E-02	3.80E+00	8.21E-03	0.0090
	Cs-134	0.00E+00	N/A	U	1.71E-01	1.09E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.47E-01	6.58E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.34E-01	1.57E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.89E-01	1.67E-01	8.00E+00	0.00E+00	
	Eu-155	2.34E-01	N/A	U	4.46E-01	2.19E-01	2.80E+02	8.36E-04	
	Am-241	0.00E+00	N/A	U	3.69E-01	1.75E-01	2.10E+00	0.00E+00	
8400X-2-CI-GSSX-032	Co-60	4.49E-02	N/A	U	1.44E-01	6.48E-02	3.80E+00	1.18E-02	0.0826
	Cs-134	0.00E+00	N/A	U	1.43E-01	9.36E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.38E-01	6.18E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.30E-01	1.56E-01	8.70E+00	0.00E+00	
	Eu-154	1.21E-01	N/A	U	4.45E-01	1.97E-01	8.00E+00	1.51E-02	
	Eu-155	2.49E-01	N/A	U	4.60E-01	2.21E-01	2.80E+02	8.89E-04	
	Am-241	1.15E-01	N/A	U	3.94E-01	1.88E-01	2.10E+00	5.48E-02	
8400X-2-CI-GSSX-033	Co-60	0.00E+00	N/A	U	1.26E-01	5.65E-02	3.80E+00	0.00E+00	0.0143
	Cs-134	0.00E+00	N/A	U	1.78E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	4.52E-02	N/A	U	1.71E-01	7.75E-02	1.10E+01	4.11E-03	
	Eu-152	4.45E-02	N/A	U	3.58E-01	1.69E-01	8.70E+00	5.11E-03	
	Eu-154	0.00E+00	N/A	U	4.46E-01	1.94E-01	8.00E+00	0.00E+00	
	Eu-155	1.99E-01	N/A	U	3.41E-01	1.61E-01	2.80E+02	7.11E-04	
	Am-241	9.26E-03	N/A	U	3.95E-01	1.88E-01	2.10E+00	4.41E-03	

Table 2-63 8400 Summary Statistics

Combined					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	9.92E-02	2.72E-02	2.29E-02	2.38E-02
Cs-134	0.00E+00	9.04E-02	2.52E-03	0.00E+00	1.49E-02
Cs-137	0.00E+00	2.49E-01	4.10E-02	3.14E-02	4.73E-02
Eu-152	0.00E+00	1.37E-01	2.69E-02	0.00E+00	3.76E-02
Eu-154	0.00E+00	1.94E-01	4.12E-02	0.00E+00	6.13E-02
Eu-155	0.00E+00	4.87E-01	1.69E-01	1.76E-01	1.09E-01
Am-241	0.00E+00	1.15E-01	1.06E-02	0.00E+00	2.66E-02
Random					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	4.42E-02	2.18E-02	2.67E-02	1.42E-02
Cs-134	0.00E+00	9.04E-02	6.03E-03	0.00E+00	2.33E-02
Cs-137	0.00E+00	2.49E-01	5.47E-02	3.53E-02	6.83E-02
Eu-152	0.00E+00	1.37E-01	3.84E-02	2.89E-03	4.82E-02
Eu-154	0.00E+00	1.62E-01	4.08E-02	0.00E+00	6.07E-02
Eu-155	0.00E+00	4.87E-01	1.77E-01	1.59E-01	1.33E-01
Am-241	0.00E+00	4.19E-02	8.15E-03	0.00E+00	1.46E-02
Judgmental					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	9.92E-02	3.22E-02	2.69E-02	2.89E-02
Cs-134	0.00E+00	2.96E-03	1.64E-04	0.00E+00	6.98E-04
Cs-137	0.00E+00	7.75E-02	3.26E-02	2.94E-02	2.31E-02
Eu-152	0.00E+00	7.60E-02	1.69E-02	0.00E+00	2.47E-02
Eu-154	0.00E+00	1.94E-01	4.87E-02	0.00E+00	6.75E-02
Eu-155	0.00E+00	2.74E-01	1.59E-01	1.79E-01	9.43E-02
Am-241	0.00E+00	1.15E-01	1.26E-02	0.00E+00	3.50E-02

Total Number of Samples	
Random	15
Judgmental	18
QC	4

Random	
SOF >0.5	0
Maximum SOF	0.0460
Minimum SOF	0.0003

Judgmental	
SOF >0.5	0
Maximum SOF	0.0826
Minimum SOF	0.0068

Figure 2-34 Survey Unit 8700 Random Sample and Scan Locations

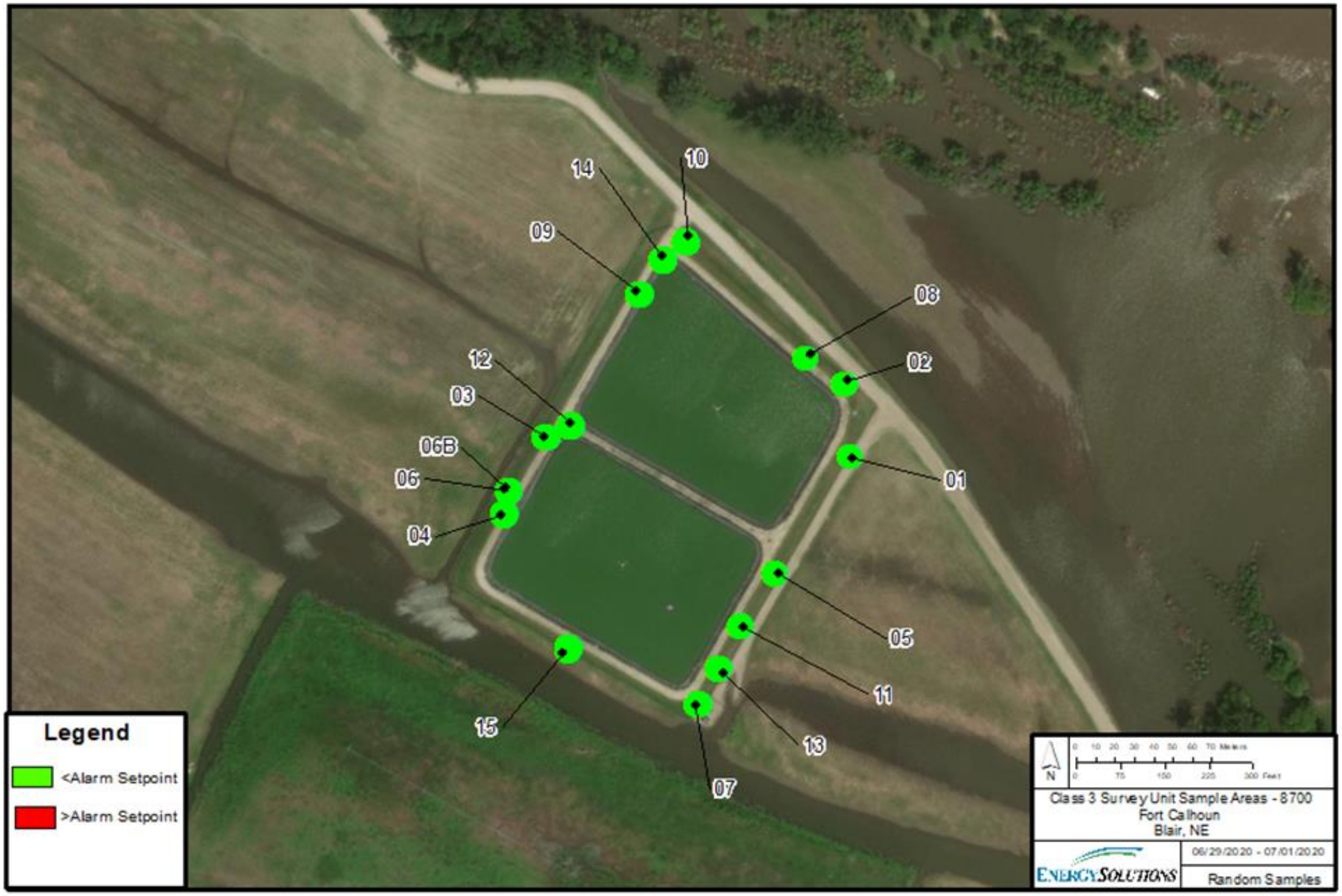


Figure 2-35 Survey Unit 8700 Judgmental Sample and Scan Locations

Table 2-64 8700 Gamma Spectroscopy Results for Random Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8700X-2-CR-GSSX-001	Co-60	4.67E-02	N/A	U	1.15E-01	7.88E-02	3.80E+00	1.23E-02	0.0175
	Cs-134	0.00E+00	N/A	U	2.04E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	5.53E-02	N/A	U	1.95E-01	8.84E-02	1.10E+01	5.03E-03	
	Eu-152	0.00E+00	N/A	U	3.55E-01	1.66E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.73E-01	2.03E-01	8.00E+00	0.00E+00	
	Eu-155	6.06E-02	N/A	U	4.90E-01	2.34E-01	2.80E+02	2.16E-04	
	Am-241	0.00E+00	N/A	U	4.08E-01	1.93E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-002	Co-60	4.39E-02	N/A	U	1.55E-01	6.64E-02	3.80E+00	1.16E-02	0.0162
	Cs-134	0.00E+00	N/A	U	1.98E-01	1.19E-01	5.70E+00	0.00E+00	
	Cs-137	2.15E-02	N/A	U	1.94E-01	8.72E-02	1.10E+01	1.95E-03	
	Eu-152	0.00E+00	N/A	U	3.73E-01	1.75E-01	8.70E+00	0.00E+00	
	Eu-154	1.36E-02	N/A	U	4.87E-01	2.09E-01	8.00E+00	1.70E-03	
	Eu-155	2.83E-01	N/A	U	4.94E-01	2.36E-01	2.80E+02	1.01E-03	
	Am-241	0.00E+00	N/A	U	4.12E-01	1.96E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-003	Co-60	0.00E+00	N/A	U	1.33E-01	5.69E-02	3.80E+00	0.00E+00	0.0187
	Cs-134	0.00E+00	N/A	U	1.46E-01	1.17E-01	5.70E+00	0.00E+00	
	Cs-137	8.75E-02	N/A	U	1.69E-01	7.70E-02	1.10E+01	7.95E-03	
	Eu-152	0.00E+00	N/A	U	3.22E-01	1.51E-01	8.70E+00	0.00E+00	
	Eu-154	8.12E-02	N/A	U	4.21E-01	1.84E-01	8.00E+00	1.02E-02	
	Eu-155	1.60E-01	N/A	U	4.29E-01	2.06E-01	2.80E+02	5.71E-04	
	Am-241	0.00E+00	N/A	U	3.39E-01	1.60E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-004	Co-60	4.83E-02	N/A	U	1.19E-01	5.23E-02	3.80E+00	1.27E-02	0.0229
	Cs-134	0.00E+00	N/A	U	1.72E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	3.66E-02	N/A	U	1.57E-01	7.07E-02	1.10E+01	3.33E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	8.58E-03	N/A	U	3.09E-01	1.45E-01	8.70E+00	9.86E-04	
	Eu-154	4.13E-02	N/A	U	4.19E-01	1.82E-01	8.00E+00	5.16E-03	
	Eu-155	1.87E-01	N/A	U	4.66E-01	2.24E-01	2.80E+02	6.68E-04	
	Am-241	0.00E+00	N/A	U	3.59E-01	1.70E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-005	Co-60	3.90E-02	N/A	U	9.07E-02	6.59E-02	3.80E+00	1.03E-02	0.0125
	Cs-134	0.00E+00	N/A	U	1.81E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	1.87E-02	N/A	U	1.48E-01	6.60E-02	1.10E+01	1.70E-03	
	Eu-152	0.00E+00	N/A	U	3.30E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.45E-01	1.45E-01	8.00E+00	0.00E+00	
	Eu-155	1.43E-01	N/A	U	4.45E-01	2.13E-01	2.80E+02	5.11E-04	
	Am-241	0.00E+00	N/A	U	3.67E-02	1.74E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSA-006	Co-60	1.84E-02	N/A	U	1.21E-01	5.12E-02	3.80E+00	4.84E-03	0.0152
	Cs-134	0.00E+00	N/A	U	1.81E-01	3.32E-01	5.70E+00	0.00E+00	
	Cs-137	3.31E-02	N/A	U	1.50E-01	6.80E-02	1.10E+01	3.01E-03	
	Eu-152	0.00E+00	N/A	U	3.28E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	5.46E-02	N/A	U	4.12E-01	1.80E-01	8.00E+00	6.83E-03	
	Eu-155	1.52E-01	N/A	U	3.34E-01	1.58E-01	2.80E+02	5.43E-04	
	Am-241	0.00E+00	N/A	U	4.05E-01	1.93E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-007	Co-60	1.12E-02	N/A	U	1.31E-01	5.55E-02	3.80E+00	2.95E-03	0.0177
	Cs-134	0.00E+00	N/A	U	1.76E-01	1.17E-01	5.70E+00	0.00E+00	
	Cs-137	1.57E-01	N/A	U	1.86E-01	8.53E-02	1.10E+01	1.43E-02	
	Eu-152	0.00E+00	N/A	U	3.49E-01	1.65E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.64E-01	1.54E-01	8.00E+00	0.00E+00	
	Eu-155	1.45E-01	N/A	U	3.30E-01	1.55E-01	2.80E+02	5.18E-04	
	Am-241	0.00E+00	N/A	U	3.79E-01	1.80E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-008	Co-60	3.55E-02	N/A	U	1.57E-01	7.11E-02	3.80E+00	9.34E-03	0.0187

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.68E-01	1.16E-01	5.70E+00	0.00E+00	
	Cs-137	5.19E-02	N/A	U	1.61E-01	7.23E-02	1.10E+01	4.72E-03	
	Eu-152	2.80E-02	N/A	U	3.47E-01	1.63E-01	8.70E+00	3.22E-03	
	Eu-154	0.00E+00	N/A	U	4.27E-01	1.84E-01	8.00E+00	0.00E+00	
	Eu-155	3.93E-01	N/A	U	4.92E-01	2.36E-01	2.80E+02	1.40E-03	
	Am-241	0.00E+00	N/A	U	3.92E-01	1.86E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-009	Co-60	3.04E-02	N/A	U	1.09E-01	4.60E-02	3.80E+00	8.00E-03	0.0148
	Cs-134	0.00E+00	N/A	U	1.48E-01	9.49E-02	5.70E+00	0.00E+00	
	Cs-137	1.17E-02	N/A	U	1.38E-01	6.20E-02	1.10E+01	1.06E-03	
	Eu-152	4.27E-02	N/A	U	3.15E-01	1.49E-01	8.70E+00	4.91E-03	
	Eu-154	0.00E+00	N/A	U	3.21E-01	1.37E-01	8.00E+00	0.00E+00	
	Eu-155	2.19E-01	N/A	U	4.27E-01	2.05E-01	2.80E+02	7.82E-04	
	Am-241	0.00E+00	N/A	U	3.27E-01	1.55E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-010	Co-60	0.00E+00	N/A	U	1.32E-01	5.60E-02	3.80E+00	0.00E+00	0.0058
	Cs-134	0.00E+00	N/A	U	1.89E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	4.16E-02	N/A	U	1.60E-01	7.25E-02	1.10E+01	3.78E-03	
	Eu-152	1.29E-02	N/A	U	3.48E-01	1.64E-01	8.70E+00	1.48E-03	
	Eu-154	0.00E+00	N/A	U	4.44E-01	1.95E-01	8.00E+00	0.00E+00	
	Eu-155	1.59E-01	N/A	U	4.69E-01	2.28E-01	2.80E+02	5.68E-04	
	Am-241	0.00E+00	N/A	U	3.93E-01	1.87E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSA-011	Co-60	0.00E+00	N/A	U	1.25E-01	8.24E-02	3.80E+00	0.00E+00	0.0062
	Cs-134	0.00E+00	N/A	U	1.80E-01	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	4.34E-02	N/A	U	1.84E-01	8.37E-02	1.10E+01	3.95E-03	
	Eu-152	0.00E+00	N/A	U	3.46E-01	1.62E-01	8.70E+00	0.00E+00	
	Eu-154	1.37E-02	N/A	U	3.95E-01	1.68E-01	8.00E+00	1.71E-03	
	Eu-155	1.43E-01	N/A	U	4.72E-01	2.26E-01	2.80E+02	5.11E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	4.02E-01	1.91E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-012	Co-60	5.72E-03	N/A	U	6.51E-02	2.64E-02	3.80E+00	1.51E-03	0.0051
	Cs-134	0.00E+00	N/A	U	8.61E-02	7.50E-02	5.70E+00	0.00E+00	
	Cs-137	3.77E-02	N/A	U	8.89E-02	3.95E-02	1.10E+01	3.43E-03	
	Eu-152	0.00E+00	N/A	U	2.23E-01	1.05E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	1.81E-01	7.37E-02	8.00E+00	0.00E+00	
	Eu-155	4.00E-02	N/A	U	2.75E-01	1.31E-01	2.80E+02	1.43E-04	
	Am-241	0.00E+00	N/A	U	2.47E-01	1.16E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-013	Co-60	1.11E-02	N/A	U	1.15E-01	4.66E-02	3.80E+00	2.92E-03	0.0141
	Cs-134	0.00E+00	N/A	U	1.93E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	3.98E-02	N/A	U	1.59E-01	7.10E-02	1.10E+01	3.62E-03	
	Eu-152	6.34E-02	N/A	U	3.55E-01	1.67E-01	8.70E+00	7.29E-03	
	Eu-154	0.00E+00	N/A	U	3.83E-01	1.61E-01	8.00E+00	0.00E+00	
	Eu-155	7.81E-02	N/A	U	3.43E-01	1.61E-01	2.80E+02	2.79E-04	
	Am-241	0.00E+00	N/A	U	3.58E-01	1.69E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-014	Co-60	4.02E-02	N/A	U	1.02E-01	4.32E-02	3.80E+00	1.06E-02	0.0173
	Cs-134	0.00E+00	N/A	U	1.28E-01	9.31E-02	5.70E+00	0.00E+00	
	Cs-137	6.57E-02	N/A	U	1.23E-01	5.55E-02	1.10E+01	5.97E-03	
	Eu-152	0.00E+00	N/A	U	2.83E-01	1.34E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.58E-01	1.07E-01	8.00E+00	0.00E+00	
	Eu-155	1.96E-01	N/A	U	3.64E-01	1.80E-01	2.80E+02	7.00E-04	
	Am-241	0.00E+00	N/A	U	3.04E-01	1.44E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSX-015	Co-60	0.00E+00	N/A	U	1.26E-01	6.12E-02	3.80E+00	0.00E+00	0.0358
	Cs-134	0.00E+00	N/A	U	1.62E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	9.99E-02	N/A	U	1.82E-01	8.33E-02	1.10E+01	9.08E-03	
	Eu-152	5.28E-02	N/A	U	3.20E-01	1.50E-01	8.70E+00	6.07E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	1.59E-01	N/A	U	4.12E-01	1.79E-01	8.00E+00	1.99E-02	
	Eu-155	2.07E-01	N/A	U	4.43E-01	2.12E-01	2.80E+02	7.39E-04	
	Am-241	0.00E+00	N/A	U	3.41E-01	1.61E-01	2.10E+00	0.00E+00	

Table 2-65 8700 Gamma Spectroscopy Results for QC Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8700X-2-CQ-GSSX-003	Co-60	3.26E-02	N/A	U	1.21E-01	5.06E-02	3.80E+00	8.58E-03	0.0269
	Cs-134	0.00E+00	N/A	U	1.69E-01	1.18E-01	5.70E+00	0.00E+00	
	Cs-137	3.28E-02	N/A	U	1.56E-01	7.04E-02	1.10E+01	2.98E-03	
	Eu-152	0.00E+00	N/A	U	3.72E-01	1.76E-01	8.70E+00	0.00E+00	
	Eu-154	1.19E-01	N/A	U	4.12E-01	1.79E-01	8.00E+00	1.49E-02	
	Eu-155	1.40E-01	N/A	U	4.65E-01	2.23E-01	2.80E+02	5.00E-04	
	Am-241	0.00E+00	N/A	U	4.01E-01	1.91E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSB-006	Co-60	5.48E-03	N/A	U	1.15E-01	7.75E-02	3.80E+00	1.44E-03	0.0433
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.20E-01	5.70E+00	0.00E+00	
	Cs-137	9.90E-03	N/A	U	1.35E-01	5.87E-02	1.10E+01	9.00E-04	
	Eu-152	5.68E-02	N/A	U	3.44E-01	1.61E-01	8.70E+00	6.53E-03	
	Eu-154	2.66E-01	N/A	U	5.16E-01	2.27E-01	8.00E+00	3.33E-02	
	Eu-155	3.38E-01	N/A	U	4.61E-01	2.20E-01	2.80E+02	1.21E-03	
	Am-241	0.00E+00	N/A	U	3.75E-01	1.77E-01	2.10E+00	0.00E+00	
8700X-2-CR-GSSB-011	Co-60	7.92E-02	N/A	U	1.54E-01	7.60E-02	3.80E+00	2.08E-02	0.0254
	Cs-134	0.00E+00	N/A	U	1.76E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	3.85E-02	N/A	U	1.63E-01	7.35E-02	1.10E+01	3.50E-03	
	Eu-152	0.00E+00	N/A	U	3.13E-01	1.46E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.82E-01	1.63E-01	8.00E+00	0.00E+00	
	Eu-155	3.07E-01	N/A	U	4.50E-01	2.16E-01	2.80E+02	1.10E-03	
	Am-241	0.00E+00	N/A	U	3.75E-01	1.78E-01	2.10E+00	0.00E+00	
8700X-2-CQ-GSSX-013	Co-60	0.00E+00	N/A	U	1.40E-01	6.42E-02	3.80E+00	0.00E+00	0.0203
	Cs-134	0.00E+00	N/A	U	1.74E-01	1.26E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.63E-01	7.29E-02	1.10E+01	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	1.75E-01	N/A	U	3.54E-01	1.66E-01	8.70E+00	2.01E-02	
	Eu-154	0.00E+00	N/A	U	4.35E-01	1.86E-01	8.00E+00	0.00E+00	
	Eu-155	5.94E-02	N/A	U	4.76E-01	2.28E-01	2.80E+02	2.12E-04	
	Am-241	0.00E+00	N/A	U	3.75E-01	1.77E-01	2.10E+00	0.00E+00	

Table 2-66 8700 Gamma Spectroscopy Results for Judgmental Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
8700X-2-CJ-GSMX-016	Co-60	2.85E-03	N/A	U	7.55E-03	4.50E-03	3.80E+00	7.50E-04	0.0012
	Cs-134	0.00E+00	N/A	U	9.15E-03	6.80E-03	5.70E+00	0.00E+00	
	Cs-137	5.22E-03	N/A	U	1.19E-02	5.39E-03	1.10E+01	4.75E-04	
	Eu-152	0.00E+00	N/A	U	2.80E-02	1.59E-02	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.40E-02	1.02E-02	8.00E+00	0.00E+00	
	Eu-155	6.49E-03	N/A	U	4.67E-02	2.21E-02	2.80E+02	2.32E-05	
	Am-241	0.00E+00	N/A	U	8.60E-02	4.08E-02	2.10E+00	0.00E+00	
8700X-2-CJ-GSMX-017	Co-60	2.28E-03	N/A	U	8.31E-03	3.53E-03	3.80E+00	6.00E-04	0.0025
	Cs-134	0.00E+00	N/A	U	9.35E-03	5.95E-03	5.70E+00	0.00E+00	
	Cs-137	1.42E-02	2.57E-03		7.14E-03	3.05E-03	1.10E+01	1.29E-03	
	Eu-152	2.60E-03	N/A	U	2.82E-02	1.66E-02	8.70E+00	2.99E-04	
	Eu-154	2.75E-03	N/A	U	2.19E-02	9.16E-03	8.00E+00	3.44E-04	
	Eu-155	0.00E+00	N/A	U	4.62E-02	2.19E-02	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	8.62E-02	4.10E-02	2.10E+00	0.00E+00	
8700X-2-CJ-GSSX-018	Co-60	6.23E-02	N/A	U	1.55E-01	7.19E-02	3.80E+00	1.64E-02	0.0331
	Cs-134	0.00E+00	N/A	U	1.82E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	6.28E-02	N/A	U	1.80E-01	8.13E-02	1.10E+01	5.71E-03	
	Eu-152	3.42E-02	N/A	U	3.32E-01	1.55E-01	8.70E+00	3.93E-03	
	Eu-154	5.55E-02	N/A	U	4.71E-01	2.05E-01	8.00E+00	6.94E-03	
	Eu-155	4.75E-02	N/A	U	4.53E-01	2.18E-01	2.80E+02	1.70E-04	
	Am-241	0.00E+00	N/A	U	3.73E-01	1.77E-01	2.10E+00	0.00E+00	
8700X-2-CJ-GSSX-019	Co-60	0.00E+00	N/A	U	1.21E-01	6.86E-02	3.80E+00	0.00E+00	0.0002
	Cs-134	0.00E+00	N/A	U	1.88E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.43E-01	6.28E-02	1.10E+01	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.40E-01	1.59E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.30E-01	1.85E-01	8.00E+00	0.00E+00	
	Eu-155	5.33E-02	N/A	U	3.44E-01	1.62E-01	2.80E+02	1.90E-04	
	Am-241	0.00E+00	N/A	U	3.80E-01	1.80E-01	2.10E+00	0.00E+00	
8700X-2-CJ-GSSX-020	Co-60	2.01E-02	N/A	U	1.30E-01	6.38E-02	3.80E+00	5.29E-03	0.0400
	Cs-134	0.00E+00	N/A	U	1.62E-01	1.01E-01	5.70E+00	0.00E+00	
	Cs-137	2.78E-02	N/A	U	1.56E-01	7.07E-02	1.10E+01	2.53E-03	
	Eu-152	0.00E+00	N/A	U	3.29E-01	1.55E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.43E-01	1.46E-01	8.00E+00	0.00E+00	
	Eu-155	1.95E-01	N/A	U	4.53E-01	2.18E-01	2.80E+02	6.96E-04	
	Am-241	6.61E-02	N/A	U	3.99E-01	1.90E-01	2.10E+00	3.15E-02	
8700X-2-CJ-GSSX-021	Co-60	4.58E-02	N/A	U	1.38E-01	7.10E-02	3.80E+00	1.21E-02	0.0226
	Cs-134	0.00E+00	N/A	U	1.65E-01	9.59E-02	5.70E+00	0.00E+00	
	Cs-137	2.84E-02	N/A	U	1.54E-01	6.90E-02	1.10E+01	2.58E-03	
	Eu-152	0.00E+00	N/A	U	3.46E-01	1.63E-01	8.70E+00	0.00E+00	
	Eu-154	5.88E-02	N/A	U	4.56E-01	2.00E-01	8.00E+00	7.35E-03	
	Eu-155	1.79E-01	N/A	U	4.77E-01	2.30E-01	2.80E+02	6.39E-04	
	Am-241	0.00E+00	N/A	U	3.55E-01	1.68E-01	2.10E+00	0.00E+00	
8700X-2-CJ-GSSX-022	Co-60	2.55E-02	N/A	U	1.46E-01	6.32E-02	3.80E+00	6.71E-03	0.0226
	Cs-134	0.00E+00	N/A	U	1.66E-01	1.02E-01	5.70E+00	0.00E+00	
	Cs-137	1.61E-01	N/A	U	1.90E-01	8.73E-02	1.10E+01	1.46E-02	
	Eu-152	0.00E+00	N/A	U	3.46E-01	1.63E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.06E-01	1.76E-01	8.00E+00	0.00E+00	
	Eu-155	3.62E-01	N/A	U	4.86E-01	2.34E-01	2.80E+02	1.29E-03	
	Am-241	0.00E+00	N/A	U	3.95E-01	1.88E-01	2.10E+00	0.00E+00	
8700X-2-CJ-GSSX-022	Co-60	0.00E+00	N/A	U	1.21E-01	7.34E-02	3.80E+00	0.00E+00	0.0011

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.45E-01	9.71E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.46E-01	6.57E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.03E-01	1.42E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.02E-01	1.74E-01	8.00E+00	0.00E+00	
	Eu-155	3.16E-01	N/A	U	4.44E-01	2.13E-01	2.80E+02	1.13E-03	
	Am-241	0.00E+00	N/A	U	3.79E-01	1.81E-01	2.10E+00	0.00E+00	
8700X-2-CJ-GSSX-023	Co-60	0.00E+00	N/A	U	1.21E-01	7.34E-02	3.80E+00	0.00E+00	0.0011
	Cs-134	0.00E+00	N/A	U	1.45E-01	9.71E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.46E-01	6.57E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.03E-01	1.42E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.02E-01	1.74E-01	8.00E+00	0.00E+00	
	Eu-155	3.16E-01	N/A	U	4.44E-01	2.13E-01	2.80E+02	1.13E-03	
	Am-241	0.00E+00	N/A	U	3.79E-01	1.81E-01	2.10E+00	0.00E+00	
8700X-2-CI-GSSX-024	Co-60	5.26E-02	N/A	U	1.02E-01	6.64E-02	3.80E+00	1.38E-02	0.0173
	Cs-134	0.00E+00	N/A	U	1.52E-01	9.63E-02	5.70E+00	0.00E+00	
	Cs-137	3.26E-02	N/A	U	1.41E-01	6.30E-02	1.10E+01	2.96E-03	
	Eu-152	0.00E+00	N/A	U	3.20E-01	1.51E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.04E-01	1.77E-01	8.00E+00	0.00E+00	
	Eu-155	1.26E-01	N/A	U	4.44E-01	2.13E-01	2.80E+02	4.50E-04	
	Am-241	0.00E+00	N/A	U	3.44E-01	1.63E-01	2.10E+00	0.00E+00	

Table 2-67 8700 Summary Statistics

Combined					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	7.92E-02	2.27E-02	1.84E-02	2.28E-02
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	0.00E+00	1.61E-01	4.19E-02	3.31E-02	4.09E-02
Eu-152	0.00E+00	1.75E-01	1.64E-02	0.00E+00	3.63E-02
Eu-154	0.00E+00	2.66E-01	2.98E-02	0.00E+00	6.05E-02
Eu-155	0.00E+00	3.93E-01	1.73E-01	1.59E-01	1.09E-01
Am-241	0.00E+00	6.61E-02	2.28E-03	0.00E+00	1.23E-02
Random					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	4.83E-02	2.20E-02	1.84E-02	1.91E-02
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	1.17E-02	1.57E-01	5.34E-02	4.16E-02	3.74E-02
Eu-152	0.00E+00	6.34E-02	1.39E-02	0.00E+00	2.20E-02
Eu-154	0.00E+00	1.59E-01	2.42E-02	0.00E+00	4.48E-02
Eu-155	4.00E-02	3.93E-01	1.71E-01	1.59E-01	8.75E-02
Am-241	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Judgmental					
Radionuclide	**Min (pCi/g)**	**Max (pCi/g)**	**Mean (pCi/g)**	**Median (pCi/g)**	**SD (pCi/g)**
Co-60	0.00E+00	6.23E-02	2.11E-02	1.15E-02	2.44E-02
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	0.00E+00	1.61E-01	3.32E-02	2.10E-02	4.91E-02
Eu-152	0.00E+00	3.42E-02	3.68E-03	0.00E+00	1.08E-02
Eu-154	0.00E+00	5.88E-02	1.17E-02	0.00E+00	2.40E-02
Eu-155	0.00E+00	3.62E-01	1.60E-01	1.53E-01	1.36E-01
Am-241	0.00E+00	6.61E-02	6.61E-03	0.00E+00	2.09E-02

Total Number of Samples	
Random	15
Judgmental	10
QC	4

Random	
SOF >0.5	0
Maximum SOF	0.0433
Minimum SOF	0.0002

Judgmental	
SOF >0.5	0
Maximum SOF	0.0400
Minimum SOF	0.0002

Figure 2-36 Survey Unit 7000 Judgmental Sample Locations

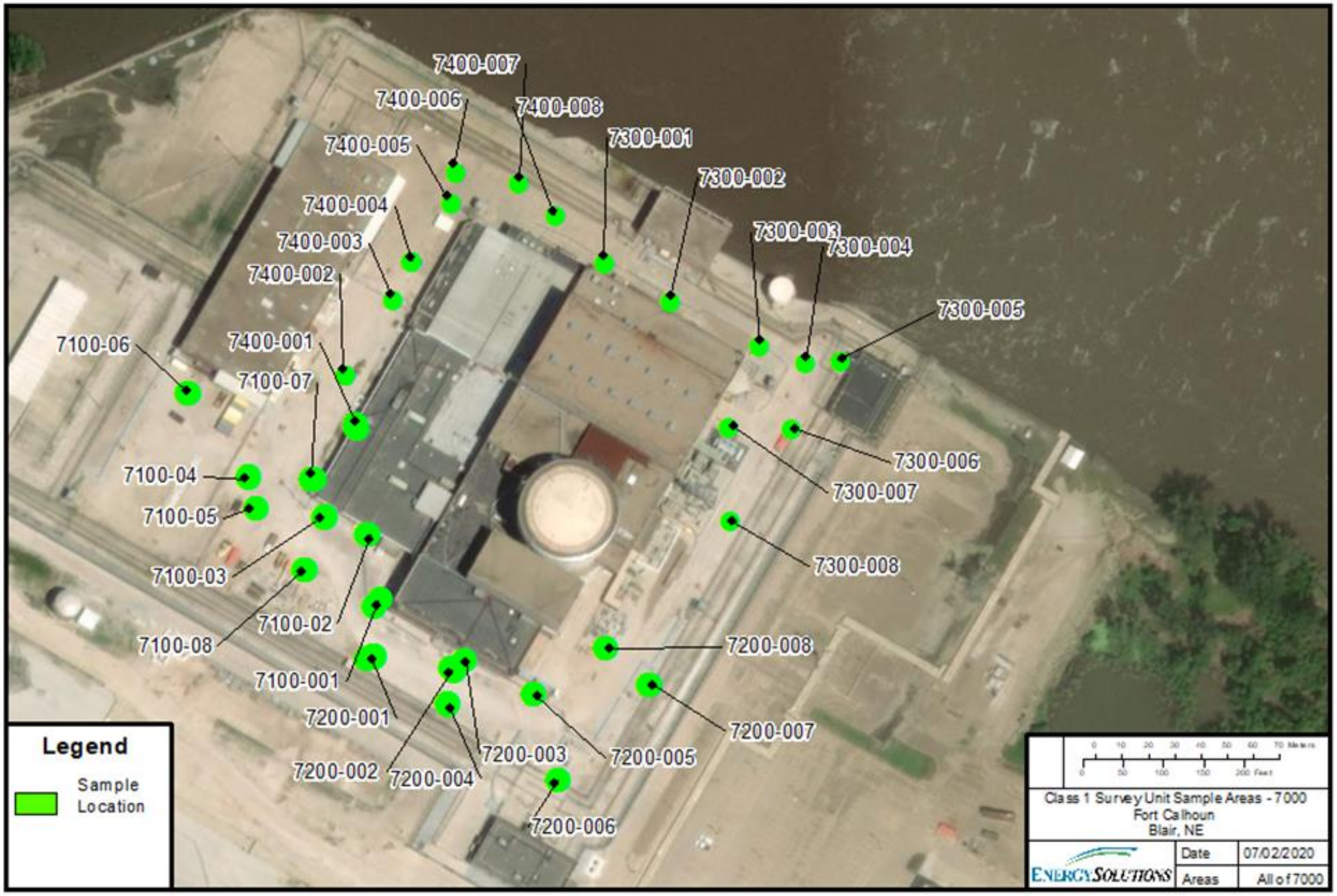


Table 2-68 7100 Gamma Spectroscopy Results for Judgmental Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7100X-1-CJ-GSSX-001	Co-60	1.22E-02	N/A	U	9.68E-02	5.20E-02	3.80E+00	3.21E-03	0.0177
	Cs-134	0.00E+00	N/A	U	1.25E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	5.33E-02	N/A	U	1.22E-01	5.49E-02	1.10E+01	4.85E-03	
	Eu-152	0.00E+00	N/A	U	3.09E-01	1.46E-01	8.70E+00	0.00E+00	
	Eu-154	7.74E-02	N/A	U	3.46E-01	1.52E-01	8.00E+00	9.68E-03	
	Eu-155	0.00E+00	N/A	U	3.61E-01	1.73E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	3.22E-01	1.53E-01	2.10E+00	0.00E+00	
7100X-1-CJ-GSSX-002	Co-60	6.55E-02	N/A	U	8.32E-02	4.80E-02	3.80E+00	1.72E-02	0.0311
	Cs-134	0.00E+00	N/A	U	1.02E-01	9.88E-02	5.70E+00	0.00E+00	
	Cs-137	7.77E-02	2.73E-02		9.98E-02	4.44E-02	1.10E+01	7.06E-03	
	Eu-152	0.00E+00	N/A	U	2.69E-01	1.27E-01	8.70E+00	0.00E+00	
	Eu-154	4.67E-02	N/A	U	3.18E-01	1.40E-01	8.00E+00	5.84E-03	
	Eu-155	2.66E-01	N/A	U	3.62E-01	1.74E-01	2.80E+02	9.50E-04	
	Am-241	0.00E+00	N/A	U	3.11E-01	1.48E-01	2.10E+00	0.00E+00	
7100X-1-CJ-GSSX-003	Co-60	1.92E-02	N/A	U	7.26E-02	3.39E-02	3.80E+00	5.05E-03	0.0158
	Cs-134	0.00E+00	N/A	U	9.13E-02	8.76E-02	5.70E+00	0.00E+00	
	Cs-137	5.32E-02	N/A	U	9.73E-02	4.38E-02	1.10E+01	4.84E-03	
	Eu-152	8.96E-03	N/A	U	2.59E-01	1.23E-01	8.70E+00	1.03E-03	
	Eu-154	3.69E-02	N/A	U	2.50E-01	1.09E-01	8.00E+00	4.61E-03	
	Eu-155	7.10E-02	N/A	U	3.13E-01	1.50E-01	2.80E+02	2.54E-04	
	Am-241	0.00E+00	N/A	U	2.64E-01	1.25E-01	2.10E+00	0.00E+00	
7100X-1-CJ-GSSX-004	Co-60	0.00E+00	N/A	U	7.67E-02	3.23E-02	3.80E+00	0.00E+00	0.0163
	Cs-134	0.00E+00	N/A	U	7.85E-02	9.34E-02	5.70E+00	0.00E+00	
	Cs-137	2.14E-02	N/A	U	1.01E-01	4.57E-02	1.10E+01	1.95E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	8.37E-02	N/A	U	2.67E-01	1.27E-01	8.70E+00	9.62E-03	
	Eu-154	2.95E-02	N/A	U	2.55E-01	1.11E-01	8.00E+00	3.69E-03	
	Eu-155	2.79E-01	N/A	U	3.49E-01	1.68E-01	2.80E+02	9.96E-04	
	Am-241	0.00E+00	N/A	U	3.02E-01	1.44E-01	2.10E+00	0.00E+00	
7100X-1-CJ-GSSX-005	Co-60	0.00E+00	N/A	U	6.23E-02	2.38E-02	3.80E+00	0.00E+00	0.0055
	Cs-134	0.00E+00	N/A	U	8.61E-02	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.07E-01	4.75E-02	1.10E+01	0.00E+00	
	Eu-152	4.79E-02	N/A	U	2.74E-01	1.29E-01	8.70E+00	5.51E-03	
	Eu-154	0.00E+00	N/A	U	3.06E-01	1.33E-01	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	3.48E-01	1.66E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	2.94E-01	1.39E-01	2.10E+00	0.00E+00	
7100X-1-CJ-GSSX-006	Co-60	4.69E-02	N/A	U	8.35E-02	4.84E-02	3.80E+00	1.23E-02	0.0131
	Cs-134	0.00E+00	N/A	U	8.38E-02	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	8.45E-02	3.69E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.64E-01	1.25E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.55E-01	1.09E-01	8.00E+00	0.00E+00	
	Eu-155	2.02E-01	N/A	U	3.48E-01	1.67E-01	2.80E+02	7.21E-04	
	Am-241	0.00E+00	N/A	U	2.83E-01	1.34E-01	2.10E+00	0.00E+00	
7100X-1-CJ-GSSA-007	Co-60	3.24E-02	N/A	U	8.25E-02	5.36E-02	3.80E+00	8.53E-03	0.0138
	Cs-134	0.00E+00	N/A	U	1.12E-01	1.13E-01	5.70E+00	0.00E+00	
	Cs-137	5.82E-02	N/A	U	1.41E-01	6.42E-02	1.10E+01	5.29E-03	
	Eu-152	0.00E+00	N/A	U	2.79E-01	1.31E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.30E-01	9.23E-02	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	4.05E-01	1.94E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	3.39E-01	1.61E-01	2.10E+00	0.00E+00	
7100X-1-CJ-GSSX-008	Co-60	1.06E-02	N/A	U	7.25E-02	4.21E-02	3.80E+00	2.79E-03	0.0209

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.02E-01	9.66E-02	5.70E+00	0.00E+00	
	Cs-137	4.34E-02	N/A	U	1.06E-01	4.77E-02	1.10E+01	3.95E-03	
	Eu-152	0.00E+00	N/A	U	2.72E-01	1.29E-01	8.70E+00	0.00E+00	
	Eu-154	1.12E-01	N/A	U	2.89E-01	1.25E-01	8.00E+00	1.40E-02	
	Eu-155	4.92E-02	N/A	U	3.48E-01	1.67E-01	2.80E+02	1.76E-04	
	Am-241	0.00E+00	N/A	U	3.14E-01	1.49E-01	2.10E+00	0.00E+00	
7100X-1-CJ-GSBX-009	Co-60	0.00E+00	N/A	U	1.10E-01	6.74E-02	3.80E+00	0.00E+00	0.0529
	Cs-134	0.00E+00	N/A	U	1.51E-01	9.30E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.24E-01	5.49E-02	1.10E+01	0.00E+00	
	Eu-152	1.35E-01	N/A	U	3.37E-01	1.59E-01	8.70E+00	1.55E-02	
	Eu-154	3.85E-02	N/A	U	4.26E-01	1.88E-01	8.00E+00	4.81E-03	
	Eu-155	0.00E+00	N/A	U	3.74E-01	1.78E-01	2.80E+02	0.00E+00	
	Am-241	6.85E-02	N/A	U	3.66E-01	1.74E-01	2.10E+00	3.26E-02	
7100X-1-CJ-GSBX-010	Co-60	0.00E+00	N/A	U	1.08E-01	5.20E-02	3.80E+00	0.00E+00	0.0046
	Cs-134	0.00E+00	N/A	U	1.45E-01	9.69E-02	5.70E+00	0.00E+00	
	Cs-137	4.65E-02	N/A	U	1.61E-01	7.29E-02	1.10E+01	4.23E-03	
	Eu-152	0.00E+00	N/A	U	3.06E-01	1.43E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.13E-01	1.30E-01	8.00E+00	0.00E+00	
	Eu-155	1.07E-01	N/A	U	4.31E-01	2.06E-01	2.80E+02	3.82E-04	
	Am-241	0.00E+00	N/A	U	3.65E-01	1.73E-01	2.10E+00	0.00E+00	

Table 2-69 7100 Gamma Spectroscopy Results for QC Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7100X-1-CQ-GSSX-003	Co-60	3.17E-02	N/A	U	8.53E-02	3.96E-02	3.80E+00	8.34E-03	0.0167
	Cs-134	0.00E+00	N/A	U	8.64E-02	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	6.07E-02	N/A	U	1.24E-01	5.57E-02	1.10E+01	5.52E-03	
	Eu-152	1.73E-02	N/A	U	2.63E-01	1.23E-01	8.70E+00	1.99E-03	
	Eu-154	0.00E+00	N/A	U	2.95E-01	1.26E-01	8.00E+00	0.00E+00	
	Eu-155	2.43E-01	N/A	U	3.47E-01	1.74E-01	2.80E+02	8.68E-04	
	Am-241	0.00E+00	N/A	U	2.89E-01	1.37E-01	2.10E+00	0.00E+00	
7100X-1-CJ-GSSB-007	Co-60	0.00E+00	N/A	U	9.00E-02	3.78E-02	3.80E+00	0.00E+00	0.0075
	Cs-134	0.00E+00	N/A	U	1.21E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	1.44E-02	1.86E-02		8.04E-02	3.40E-02	1.10E+01	1.31E-03	
	Eu-152	5.39E-02	N/A	U	2.96E-01	1.40E-01	8.70E+00	6.20E-03	
	Eu-154	0.00E+00	N/A	U	2.41E-01	9.89E-02	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	3.54E-01	1.69E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	3.28E-01	1.56E-01	2.10E+00	0.00E+00	

Table 2-70 7100 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	6.55E-02	1.87E-02	1.14E-02	2.28E-02
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	0.00E+00	7.77E-02	3.54E-02	4.50E-02	2.81E-02
Eu-152	0.00E+00	1.35E-01	2.76E-02	0.00E+00	4.72E-02
Eu-154	0.00E+00	1.12E-01	3.41E-02	3.32E-02	3.77E-02
Eu-155	0.00E+00	2.79E-01	9.74E-02	6.01E-02	1.12E-01
Am-241	0.00E+00	6.85E-02	6.85E-03	0.00E+00	2.17E-02

Total Number of Samples	
Random	0
Judgmental	10
QC	2

Judgmental	
SOF >0.5	0
Maximum SOF	0.0529
Minimum SOF	0.0046

Table 2-71 7200 Gamma Spectroscopy Results for Judgmental Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7200X-1-CJ-GSSX-001	Co-60	0.00E+00	N/A	U	7.03E-02	2.82E-02	3.80E+00	0.00E+00	0.0082
	Cs-134	0.00E+00	N/A	U	1.18E-01	8.99E-02	5.70E+00	0.00E+00	
	Cs-137	7.50E-02	N/A	U	1.13E-01	5.11E-02	1.10E+01	6.82E-03	
	Eu-152	2.23E-03	N/A	U	2.82E-01	1.33E-01	8.70E+00	2.56E-04	
	Eu-154	2.79E-03	N/A	U	2.75E-01	1.18E-01	8.00E+00	3.49E-04	
	Eu-155	2.06E-01	N/A	U	3.65E-01	1.75E-01	2.80E+02	7.36E-04	
	Am-241	0.00E+00	N/A	U	2.96E-01	1.40E-01	2.10E+00	0.00E+00	
7200X-1-CJ-GSSA-002	Co-60	1.31E-02	N/A	U	6.94E-02	3.81E-02	3.80E+00	3.45E-03	0.0110
	Cs-134	0.00E+00	N/A	U	8.27E-02	8.80E-02	5.70E+00	0.00E+00	
	Cs-137	2.70E-03	N/A	U	9.09E-02	3.99E-02	1.10E+01	2.45E-04	
	Eu-152	0.00E+00	N/A	U	2.57E-01	1.21E-01	8.70E+00	0.00E+00	
	Eu-154	5.80E-02	N/A	U	2.36E-01	9.92E-02	8.00E+00	7.25E-03	
	Eu-155	2.57E-02	N/A	U	3.11E-01	1.48E-01	2.80E+02	9.18E-05	
	Am-241	0.00E+00	N/A	U	2.53E-01	1.19E-01	2.10E+00	0.00E+00	
7200X-1-CJ-GSSX-003	Co-60	4.81E-02	N/A	U	9.41E-02	4.01E-02	3.80E+00	1.27E-02	0.0182
	Cs-134	0.00E+00	N/A	U	1.07E-01	8.09E-02	5.70E+00	0.00E+00	
	Cs-137	5.41E-02	N/A	U	1.09E-01	4.90E-02	1.10E+01	4.92E-03	
	Eu-152	0.00E+00	N/A	U	2.50E-01	1.18E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.69E-01	1.15E-01	8.00E+00	0.00E+00	
	Eu-155	1.62E-01	N/A	U	3.46E-01	1.66E-01	2.80E+02	5.79E-04	
	Am-241	0.00E+00	N/A	U	2.91E-01	1.38E-01	2.10E+00	0.00E+00	
7200X-1-CJ-GSSX-004	Co-60	9.09E-03	N/A	U	8.27E-02	3.45E-02	3.80E+00	2.39E-03	0.0122
	Cs-134	0.00E+00	N/A	U	1.13E-01	9.08E-02	5.70E+00	0.00E+00	
	Cs-137	4.47E-02	N/A	U	9.90E-02	4.39E-02	1.10E+01	4.06E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	4.97E-02	N/A	U	2.85E-01	1.35E-01	8.70E+00	5.71E-03	
	Eu-154	0.00E+00	N/A	U	2.67E-01	1.14E-01	8.00E+00	0.00E+00	
	Eu-155	1.31E-02	N/A	U	2.50E-01	1.18E-01	2.80E+02	4.68E-05	
	Am-241	0.00E+00	N/A	U	2.91E-01	1.38E-01	2.10E+00	0.00E+00	
7200X-1-CJ-GSSX-005	Co-60	5.19E-03	N/A	U	7.12E-02	2.92E-02	3.80E+00	1.37E-03	0.0226
	Cs-134	0.00E+00	N/A	U	1.07E-01	7.92E-02	5.70E+00	0.00E+00	
	Cs-137	5.71E-02	N/A	U	1.08E-01	4.88E-02	1.10E+01	5.19E-03	
	Eu-152	0.00E+00	N/A	U	2.54E-01	1.20E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.60E-01	1.12E-01	8.00E+00	0.00E+00	
	Eu-155	2.14E-01	N/A	U	3.47E-01	1.67E-01	2.80E+02	7.64E-04	
	Am-241	3.21E-02	N/A	U	3.16E-01	1.51E-01	2.10E+00	1.53E-02	
7200X-1-CJ-GSSX-006	Co-60	3.54E-02	N/A	U	8.30E-02	3.48E-02	3.80E+00	9.32E-03	0.0176
	Cs-134	0.00E+00	N/A	U	9.26E-02	9.12E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	9.02E-02	3.97E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.50E-01	1.18E-01	8.70E+00	0.00E+00	
	Eu-154	6.53E-02	N/A	U	2.46E-01	1.04E-01	8.00E+00	8.16E-03	
	Eu-155	2.97E-02	N/A	U	3.01E-01	1.43E-01	2.80E+02	1.06E-04	
	Am-241	0.00E+00	N/A	U	2.66E-01	1.26E-01	2.10E+00	0.00E+00	
7200X-1-CJ-GSSX-007	Co-60	2.23E-02	N/A	U	1.07E-01	4.66E-02	3.80E+00	5.87E-03	0.0143
	Cs-134	0.00E+00	N/A	U	9.60E-02	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	1.01E-04	N/A	U	9.83E-02	4.35E-02	1.10E+01	9.18E-06	
	Eu-152	0.00E+00	N/A	U	2.87E-01	1.36E-01	8.70E+00	0.00E+00	
	Eu-154	6.72E-02	N/A	U	3.52E-01	1.57E-01	8.00E+00	8.40E-03	
	Eu-155	0.00E+00	N/A	U	3.67E-01	1.76E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	3.27E-01	1.56E-01	2.10E+00	0.00E+00	
7200X-1-CJ-GSSX-008	Co-60	4.86E-02	N/A	U	9.74E-02	5.19E-02	3.80E+00	1.28E-02	0.0252

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	9.45E-02	7.72E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	9.96E-02	4.46E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.77E-01	1.31E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.54E-01	1.09E-01	8.00E+00	0.00E+00	
	Eu-155	6.15E-02	N/A	U	3.66E-01	1.76E-01	2.80E+02	2.20E-04	
	Am-241	2.56E-02	N/A	U	3.55E-01	1.70E-01	2.10E+00	1.22E-02	
7200X-1-CJ-GSBX-009	Co-60	2.94E-02	N/A	U	1.20E-01	5.15E-02	3.80E+00	7.74E-03	0.0289
	Cs-134	0.00E+00	N/A	U	1.45E-01	9.08E-02	5.70E+00	0.00E+00	
	Cs-137	9.62E-02	N/A	U	1.28E-01	5.73E-02	1.10E+01	8.75E-03	
	Eu-152	1.79E-02	N/A	U	3.01E-01	1.42E-01	8.70E+00	2.06E-03	
	Eu-154	0.00E+00	N/A	U	3.75E-01	1.63E-01	8.00E+00	0.00E+00	
	Eu-155	5.66E-02	N/A	U	3.75E-01	1.79E-01	2.80E+02	2.02E-04	
	Am-241	2.14E-02	N/A	U	3.29E-01	1.56E-01	2.10E+00	1.02E-02	
7200X-1-CJ-GSBX-010	Co-60	0.00E+00	N/A	U	1.21E-01	6.44E-02	3.80E+00	0.00E+00	0.0561
	Cs-134	0.00E+00	N/A	U	2.02E-01	1.40E-01	5.70E+00	0.00E+00	
	Cs-137	6.35E-02	N/A	U	1.93E-01	8.70E-02	1.10E+01	5.77E-03	
	Eu-152	1.18E-01	N/A	U	4.60E-01	2.18E-01	8.70E+00	1.36E-02	
	Eu-154	0.00E+00	N/A	U	5.17E-01	2.25E-01	8.00E+00	0.00E+00	
	Eu-155	1.65E-01	N/A	U	5.82E-01	2.85E-01	2.80E+02	5.89E-04	
	Am-241	7.60E-02	N/A	U	5.13E-01	2.44E-01	2.10E+00	3.62E-02	

Table 2-72 7200 Gamma Spectroscopy Results for QC Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7200X-1-CJ-GSSB-002	Co-60	2.27E-03	N/A	U	7.56E-02	4.23E-02	3.80E+00	5.97E-04	0.0101
	Cs-134	0.00E+00	N/A	U	7.85E-02	8.99E-02	5.70E+00	0.00E+00	
	Cs-137	1.81E-02	N/A	U	7.65E-02	3.30E-02	1.10E+01	1.65E-03	
	Eu-152	1.93E-02	N/A	U	2.34E-01	1.10E-01	8.70E+00	2.22E-03	
	Eu-154	4.24E-02	N/A	U	2.66E-01	1.15E-01	8.00E+00	5.30E-03	
	Eu-155	8.31E-02	N/A	U	2.89E-01	1.38E-01	2.80E+02	2.97E-04	
	Am-241	0.00E+00	N/A	U	2.67E-01	1.27E-01	2.10E+00	0.00E+00	
7200X-1-CQ-GSSX-006	Co-60	4.85E-02	N/A	U	6.86E-02	4.71E-02	3.80E+00	1.28E-02	0.0146
	Cs-134	0.00E+00	N/A	U	1.04E-01	9.23E-02	5.70E+00	0.00E+00	
	Cs-137	1.61E-02	N/A	U	1.04E-01	4.61E-02	1.10E+01	1.46E-03	
	Eu-152	0.00E+00	N/A	U	2.53E-01	1.19E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.38E-01	9.93E-02	8.00E+00	0.00E+00	
	Eu-155	1.04E-01	N/A	U	3.41E-01	1.63E-01	2.80E+02	3.71E-04	
	Am-241	0.00E+00	N/A	U	2.82E-01	1.33E-01	2.10E+00	0.00E+00	

Table 2-73 7200 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	4.86E-02	2.11E-02	1.77E-02	1.86E-02
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	0.00E+00	9.62E-02	3.93E-02	4.94E-02	3.59E-02
Eu-152	0.00E+00	1.18E-01	1.88E-02	0.00E+00	3.83E-02
Eu-154	0.00E+00	6.72E-02	1.93E-02	0.00E+00	3.06E-02
Eu-155	0.00E+00	2.14E-01	9.34E-02	5.91E-02	8.38E-02
Am-241	0.00E+00	7.60E-02	1.55E-02	0.00E+00	2.48E-02

Total Number of Samples	
Random	0
Judgmental	10
QC	2

Judgmental	
SOF >0.5	0
Maximum SOF	0.0561
Minimum SOF	0.0082

Table 2-74 7300 Gamma Spectroscopy Results for Judgmental Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7300X-1-CJ-GSSX-001	Co-60	4.75E-02	N/A	U	1.09E-01	6.24E-02	3.80E+00	1.25E-02	0.0610
	Cs-134	0.00E+00	N/A	U	1.02E-01	7.59E-02	5.70E+00	0.00E+00	
	Cs-137	1.68E-02	N/A	U	1.20E-01	5.38E-02	1.10E+01	1.53E-03	
	Eu-152	0.00E+00	N/A	U	2.70E-01	1.27E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.42E-01	1.00E-01	8.00E+00	0.00E+00	
	Eu-155	8.80E-02	N/A	U	3.57E-01	1.71E-01	2.80E+02	3.14E-04	
	Am-241	9.79E-02	N/A	U	3.34E-01	1.59E-01	2.10E+00	4.66E-02	
7300X-1-CJ-GSSX-002	Co-60	4.73E-02	N/A	U	1.18E-01	5.13E-02	3.80E+00	1.24E-02	0.0362
	Cs-134	0.00E+00	N/A	U	1.31E-01	7.34E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.08E-01	4.79E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.59E-01	1.22E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.82E-01	1.70E-01	8.00E+00	0.00E+00	
	Eu-155	6.32E-02	N/A	U	3.53E-01	1,69E-1	2.80E+02	2.26E-04	
	Am-241	4.95E-02	N/A	U	3.12E-01	1.48E-01	2.10E+00	2.36E-02	
7300X-1-CJ-GSSA-003	Co-60	4.39E-02	N/A	U	1.01E-01	4.33E-02	3.80E+00	1.16E-02	0.0253
	Cs-134	0.00E+00	N/A	U	1.46E-01	7.87E-02	5.70E+00	0.00E+00	
	Cs-137	1.86E-02	N/A	U	1.06E-01	4.74E-02	1.10E+01	1.69E-03	
	Eu-152	1.04E-01	N/A	U	2.74E-01	1.29E-01	8.70E+00	1.20E-02	
	Eu-154	0.00E+00	N/A	U	3.12E-01	1.36E-01	8.00E+00	0.00E+00	
	Eu-155	2.74E-02	N/A	U	3.53E-01	1.69E-01	2.80E+02	9.79E-05	
	Am-241	0.00E+00	N/A	U	3.25E-01	1.54E-01	2.10E+00	0.00E+00	
7300X-1-CJ-GSSX-004	Co-60	3.61E-02	N/A	U	9.36E-02	4.05E-02	3.80E+00	9.50E-03	0.0109
	Cs-134	0.00E+00	N/A	U	1.10E-01	7.70E-02	5.70E+00	0.00E+00	
	Cs-137	1.14E-02	N/A	U	9.90E-02	4.44E-02	1.10E+01	1.04E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	2.43E-01	1.15E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.17E-01	1.41E-01	8.00E+00	0.00E+00	
	Eu-155	1.13E-01	N/A	U	3.18E-01	1.61E-01	2.80E+02	4.04E-04	
	Am-241	0.00E+00	N/A	U	2.95E-01	1.40E-01	2.10E+00	0.00E+00	
7300X-1-CJ-GSSX-005	Co-60	1.25E-02	N/A	U	9.88E-02	4.29E-02	3.80E+00	3.29E-03	0.0079
	Cs-134	0.00E+00	N/A	U	1.23E-01	7.30E-02	5.70E+00	0.00E+00	
	Cs-137	4.38E-02	N/A	U	1.11E-01	4.97E-02	1.10E+01	3.98E-03	
	Eu-152	0.00E+00	N/A	U	2.56E-01	1.20E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.18E-01	1.38E-01	8.00E+00	0.00E+00	
	Eu-155	1.66E-01	N/A	U	3.27E-01	1.56E-01	2.80E+02	5.93E-04	
	Am-241	0.00E+00	N/A	U	2.81E-01	1.33E-01	2.10E+00	0.00E+00	
7300X-1-CJ-GSSX-006	Co-60	2.67E-02	N/A	U	7.03E-02	2.88E-02	3.80E+00	7.03E-03	0.0091
	Cs-134	0.00E+00	N/A	U	9.66E-02	8.94E-02	5.70E+00	0.00E+00	
	Cs-137	2.08E-02	N/A	U	8.91E-02	3.94E-02	1.10E+01	1.89E-03	
	Eu-152	0.00E+00	N/A	U	2.69E-01	1.27E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.19E-01	9.17E-02	8.00E+00	0.00E+00	
	Eu-155	3.88E-02	N/A	U	3.57E-01	1.72E-01	2.80E+02	1.39E-04	
	Am-241	0.00E+00	N/A	U	2.86E-01	1.35E-01	2.10E+00	0.00E+00	
7300X-1-CJ-GSSX-007	Co-60	0.00E+00	N/A	U	9.39E-02	5.42E-02	3.80E+00	0.00E+00	0.0095
	Cs-134	0.00E+00	N/A	U	1.16E-01	8.59E-02	5.70E+00	0.00E+00	
	Cs-137	3.69E-02	N/A	U	1.10E-01	4.90E-02	1.10E+01	3.35E-03	
	Eu-152	5.28E-02	N/A	U	2.84E-01	1.34E-01	8.70E+00	6.07E-03	
	Eu-154	0.00E+00	N/A	U	2.77E-01	1.18E-01	8.00E+00	0.00E+00	
	Eu-155	3.49E-02	N/A	U	3.53E-01	1.69E-01	2.80E+02	1.25E-04	
	Am-241	0.00E+00	N/A	U	2.99E-01	1.42E-01	2.10E+00	0.00E+00	
7300X-1-CJ-GSSX-008	Co-60	4.29E-02	N/A	U	8.32E-02	3.55E-02	3.80E+00	1.13E-02	0.0148

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.00E-01	8.19E-02	5.70E+00	0.00E+00	
	Cs-137	3.76E-02	N/A	U	1.05E-01	4.77E-02	1.10E+01	3.42E-03	
	Eu-152	0.00E+00	N/A	U	2.68E-01	1.27E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.82E-01	1.24E-01	8.00E+00	0.00E+00	
	Eu-155	3.79E-02	N/A	U	3.36E-01	1.61E-01	2.80E+02	1.35E-04	
	Am-241	0.00E+00	N/A	U	3.01E-01	1.43E-01	2.10E+00	0.00E+00	
7300X-1-CJ-GSBX-009	Co-60	3.43E-02	N/A	U	1.27E-01	5.37E-02	3.80E+00	9.03E-03	0.0206
	Cs-134	0.00E+00	N/A	U	1.46E-01	9.84E-02	5.70E+00	0.00E+00	
	Cs-137	1.63E-02	N/A	U	1.29E-01	5.71E-02	1.10E+01	1.48E-03	
	Eu-152	0.00E+00	N/A	U	3.11E-01	1.46E-01	8.70E+00	0.00E+00	
	Eu-154	7.46E-02	N/A	U	4.32E-01	1.89E-01	8.00E+00	9.33E-03	
	Eu-155	2.18E-01	N/A	U	4.21E-01	2.01E-01	2.80E+02	7.79E-04	
	Am-241	0.00E+00	N/A	U	3.71E-01	1.76E-01	2.10E+00	0.00E+00	
7300X-1-CJ-GSBX-010	Co-60	0.00E+00	N/A	U	9.62E-02	4.27E-02	3.80E+00	0.00E+00	0.0120
	Cs-134	6.54E-03	N/A	U	1.33E-01	8.18E-02	5.70E+00	1.15E-03	
	Cs-137	6.95E-03	N/A	U	1.18E-01	5.29E-02	1.10E+01	6.32E-04	
	Eu-152	0.00E+00	N/A	U	2.87E-01	1.36E-01	8.70E+00	0.00E+00	
	Eu-154	8.02E-02	N/A	U	4.09E-01	1.84E-01	8.00E+00	1.00E-02	
	Eu-155	5.16E-02	N/A	U	3.82E-01	1.83E-01	2.80E+02	1.84E-04	
	Am-241	0.00E+00	N/A	U	3.30E-01	1.57E-01	2.10E+00	0.00E+00	

Table 2-75 7300 Gamma Spectroscopy Results for QC Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7300X-1-CJ-GSSB-003	Co-60	2.15E-02	N/A	U	7.98E-02	3.30E-02	3.80E+00	5.66E-03	0.0103
	Cs-134	0.00E+00	N/A	U	1.03E-01	8.42E-02	5.70E+00	0.00E+00	
	Cs-137	4.11E-02	N/A	U	9.75E-02	4.32E-02	1.10E+01	3.74E-03	
	Eu-152	0.00E+00	N/A	U	2.52E-01	1.19E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.74E-01	1.18E-01	8.00E+00	0.00E+00	
	Eu-155	2.48E-01	N/A	U	3.29E-01	1.57E-01	2.80E+02	8.86E-04	
	Am-241	0.00E+00	N/A	U	2.80E-01	1.33E-01	2.10E+00	0.00E+00	
7300X-1-CQ-GSSX-005	Co-60	1.76E-02	N/A	U	8.29E-02	3.45E-02	3.80E+00	4.63E-03	0.0186
	Cs-134	0.00E+00	N/A	U	1.08E-01	7.58E-02	5.70E+00	0.00E+00	
	Cs-137	4.74E-02	N/A	U	1.01E-01	4.50E-02	1.10E+01	4.31E-03	
	Eu-152	8.18E-02	N/A	U	2.66E-01	1.25E-01	8.70E+00	9.40E-03	
	Eu-154	0.00E+00	N/A	U	2.94E-01	1.27E-01	8.00E+00	0.00E+00	
	Eu-155	6.70E-02	N/A	U	3.20E-01	1.53E-01	2.80E+02	2.39E-04	
	Am-241	0.00E+00	N/A	U	3.04E-01	1.44E-01	2.10E+00	0.00E+00	

Table 2-76 7300 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	4.75E-02	2.91E-02	3.52E-02	1.86E-02
Cs-134	0.00E+00	6.54E-03	6.54E-04	0.00E+00	2.07E-03
Cs-137	0.00E+00	4.38E-02	2.09E-02	1.77E-02	1.42E-02
Eu-152	0.00E+00	1.04E-01	1.57E-02	0.00E+00	3.52E-02
Eu-154	0.00E+00	8.02E-02	1.55E-02	0.00E+00	3.27E-02
Eu-155	2.74E-02	2.18E-01	8.39E-02	5.74E-02	6.39E-02
Am-241	0.00E+00	9.79E-02	1.47E-02	0.00E+00	3.31E-02

Total Number of Samples	
Random	0
Judgmental	10
QC	2

Judgmental	
SOF >0.5	0
Maximum SOF	0.0610
Minimum SOF	0.0079

Table 2-77 7400 Gamma Spectroscopy Results for Judgmental Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7400X-1-CJ-GSSX-001	Co-60	9.49E-03	N/A	U	7.41E-02	3.01E-02	3.80E+00	2.50E-03	0.0245
	Cs-134	0.00E+00	N/A	U	1.20E-01	7.95E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.24E-01	5.66E-02	1.10E+01	0.00E+00	
	Eu-152	3.37E-02	N/A	U	2.78E-01	1.32E-01	8.70E+00	3.87E-03	
	Eu-154	1.40E-01	N/A	U	3.24E-01	1.73E-01	8.00E+00	1.75E-02	
	Eu-155	1.66E-01	N/A	U	3.61E-01	1.73E-01	2.80E+02	5.93E-04	
	Am-241	0.00E+00	N/A	U	3.21E-01	1.53E-01	2.10E+00	0.00E+00	
7400X-1-CJ-GSSX-002	Co-60	0.00E+00	N/A	U	8.51E-02	3.64E-02	3.80E+00	0.00E+00	0.0105
	Cs-134	0.00E+00	N/A	U	1.03E-01	9.22E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	9.17E-02	4.06E-02	1.10E+01	0.00E+00	
	Eu-152	9.10E-02	N/A	U	2.84E-01	1.35E-01	8.70E+00	1.05E-02	
	Eu-154	0.00E+00	N/A	U	2.19E-01	9.11E-02	8.00E+00	0.00E+00	
	Eu-155	0.00E+00	N/A	U	3.43E-01	1.64E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	2.91E-01	1.38E-01	2.10E+00	0.00E+00	
7400X-1-CJ-GSSX-003	Co-60	6.72E-03	N/A	U	7.61E-02	3.15E-02	3.80E+00	1.77E-03	0.0059
	Cs-134	0.00E+00	N/A	U	1.10E-01	8.54E-02	5.70E+00	0.00E+00	
	Cs-137	1.89E-02	N/A	U	1.03E-01	4.63E-02	1.10E+01	1.72E-03	
	Eu-152	1.79E-02	N/A	U	2.62E-01	1.24E-01	8.70E+00	2.06E-03	
	Eu-154	0.00E+00	N/A	U	2.55E-01	1.09E-01	8.00E+00	0.00E+00	
	Eu-155	9.02E-02	N/A	U	3.22E-01	1.54E-01	2.80E+02	3.22E-04	
	Am-241	0.00E+00	N/A	U	2.82E-01	1.33E-01	2.10E+00	0.00E+00	
7400X-1-CJ-GSSX-004	Co-60	8.31E-02	N/A	U	9.65E-02	6.24E-02	3.80E+00	2.19E-02	0.0472
	Cs-134	0.00E+00	N/A	U	1.06E-01	6.76E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	9.60E-02	4.28E-02	1.10E+01	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	1.51E-01	N/A	U	2.63E-01	1.25E-01	8.70E+00	1.74E-02	
	Eu-154	6.10E-02	N/A	U	2.91E-01	1.28E-01	8.00E+00	7.63E-03	
	Eu-155	8.46E-02	N/A	U	3.16E-01	1.51E-01	2.80E+02	3.02E-04	
	Am-241	0.00E+00	N/A	U	2.68E-01	1.27E-01	2.10E+00	0.00E+00	
7400X-1-CJ-GSSA-005	Co-60	2.63E-02	N/A	U	8.57E-02	3.63E-02	3.80E+00	6.92E-03	0.0133
	Cs-134	0.00E+00	N/A	U	1.14E-01	7.53E-02	5.70E+00	0.00E+00	
	Cs-137	6.47E-02	N/A	U	1.15E-01	5.23E-02	1.10E+01	5.88E-03	
	Eu-152	0.00E+00	N/A	U	2.65E-01	1.25E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.23E-01	1.43E-01	8.00E+00	0.00E+00	
	Eu-155	1.33E-01	N/A	U	3.47E-01	1.66E-01	2.80E+02	4.75E-04	
	Am-241	0.00E+00	N/A	U	2.98E-01	1.42E-01	2.10E+00	0.00E+00	
7400X-1-CJ-GSSX-006	Co-60	0.00E+00	N/A	U	9.23E-02	5.27E-02	3.80E+00	0.00E+00	0.0429
	Cs-134	0.00E+00	N/A	U	1.36E-01	8.63E-02	5.70E+00	0.00E+00	
	Cs-137	2.52E-02	N/A	U	1.18E-01	5.28E-02	1.10E+01	2.29E-03	
	Eu-152	0.00E+00	N/A	U	2.92E-01	1.37E-01	8.70E+00	0.00E+00	
	Eu-154	6.95E-02	N/A	U	3.39E-01	1.47E-01	8.00E+00	8.69E-03	
	Eu-155	9.81E-02	N/A	U	3.61E-01	1.73E-01	2.80E+02	3.50E-04	
	Am-241	6.62E-02	N/A	U	3.47E-01	1.65E-01	2.10E+00	3.15E-02	
7400X-1-CJ-GSSX-007	Co-60	0.00E+00	N/A	U	7.54E-02	4.70E-02	3.80E+00	0.00E+00	0.0012
	Cs-134	0.00E+00	N/A	U	1.28E-01	9.12E-02	5.70E+00	0.00E+00	
	Cs-137	6.23E-03	N/A	U	1.26E-01	5.72E-02	1.10E+01	5.66E-04	
	Eu-152	0.00E+00	N/A	U	2.84E-01	1.34E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.40E-01	1.49E-01	8.00E+00	0.00E+00	
	Eu-155	1.68E-01	N/A	U	3.70E-01	1.79E-01	2.80E+02	6.00E-04	
	Am-241	0.00E+00	N/A	U	3.35E-01	1.59E-01	2.10E+00	0.00E+00	
7400X-1-CJ-GSSX-008	Co-60	9.42E-03	N/A	U	8.25E-02	5.08E-02	3.80E+00	2.48E-03	0.0039

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.09E-01	7.08E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	9.87E-02	4.38E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.51E-01	1.18E-01	8.70E+00	0.00E+00	
	Eu-154	8.38E-03	N/A	U	3.15E-01	1.38E-01	8.00E+00	1.05E-03	
	Eu-155	1.14E-01	N/A	U	3.31E-01	1.58E-01	2.80E+02	4.07E-04	
	Am-241	0.00E+00	N/A	U	2.80E-01	1.33E-01	2.10E+00	0.00E+00	
7400X-1-CJ-GSBX-009	Co-60	0.00E+00	N/A	U	9.97E-02	6.04E-02	3.80E+00	0.00E+00	0.0276
	Cs-134	0.00E+00	N/A	U	1.52E-01	9.70E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.39E-01	6.22E-02	1.10E+01	0.00E+00	
	Eu-152	6.35E-02	N/A	U	3.24E-01	1.52E-01	8.70E+00	7.30E-03	
	Eu-154	1.59E-01	N/A	U	3.72E-01	1.61E-01	8.00E+00	1.99E-02	
	Eu-155	1.18E-01	N/A	U	4.42E-01	2.14E-01	2.80E+02	4.21E-04	
	Am-241	0.00E+00	N/A	U	3.84E-01	1.83E-01	2.10E+00	0.00E+00	
7400X-1-CJ-GSBX-010	Co-60	7.12E-02	N/A	U	1.04E-01	5.98E-02	3.80E+00	1.87E-02	0.0290
	Cs-134	0.00E+00	N/A	U	1.39E-01	8.34E-02	5.70E+00	0.00E+00	
	Cs-137	1.59E-02	N/A	U	1.22E-01	5.45E-02	1.10E+01	1.45E-03	
	Eu-152	5.97E-02	N/A	U	2.89E-01	1.36E-01	8.70E+00	6.86E-03	
	Eu-154	1.50E-02	N/A	U	3.29E-01	1.42E-01	8.00E+00	1.88E-03	
	Eu-155	1.93E-02	N/A	U	3.71E-01	1.77E-01	2.80E+02	6.89E-05	
	Am-241	0.00E+00	N/A	U	3.35E-01	1.59E-01	2.10E+00	0.00E+00	

Table 2-78 7400 Gamma Spectroscopy Results for QC Samples

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7400X-1-CQ-GSSX-004	Co-60	2.82E-02	N/A	U	8.82E-02	3.76E-02	3.80E+00	7.42E-03	0.0124
	Cs-134	0.00E+00	N/A	U	1.04E-01	7.24E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	9.71E-02	4.32E-02	1.10E+01	0.00E+00	
	Eu-152	3.83E-02	N/A	U	2.61E-01	1.23E-01	8.70E+00	4.40E-03	
	Eu-154	4.37E-04	N/A	U	3.04E-01	1.34E-01	8.00E+00	5.46E-05	
	Eu-155	1.50E-01	N/A	U	3.21E-01	1.54E-01	2.80E+02	5.36E-04	
	Am-241	0.00E+00	N/A	U	3.01E-01	1.43E-01	2.10E+00	0.00E+00	
7400X-1-CJ-GSSB-005	Co-60	0.00E+00	N/A	U	8.11E-02	3.44E-02	3.80E+00	0.00E+00	0.0205
	Cs-134	0.00E+00	N/A	U	1.09E-01	6.97E-02	5.70E+00	0.00E+00	
	Cs-137	3.23E-02	N/A	U	1.08E-01	4.90E-02	1.10E+01	2.94E-03	
	Eu-152	2.42E-03	N/A	U	2.53E-01	1.20E-01	8.70E+00	2.78E-04	
	Eu-154	3.73E-02	N/A	U	2.62E-01	1.14E-01	8.00E+00	4.66E-03	
	Eu-155	2.16E-01	N/A	U	3.21E-01	1.54E-01	2.80E+02	7.71E-04	
	Am-241	2.49E-02	N/A	U	2.94E-01	1.40E-01	2.10E+00	1.19E-02	

Table 2-79 7400 Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	8.31E-02	2.06E-02	8.07E-03	3.10E-02
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	0.00E+00	6.47E-02	1.31E-02	3.12E-03	2.04E-02
Eu-152	0.00E+00	1.51E-01	4.17E-02	2.58E-02	5.03E-02
Eu-154	0.00E+00	1.59E-01	4.53E-02	1.17E-02	6.08E-02
Eu-155	0.00E+00	1.68E-01	9.91E-02	1.06E-01	5.52E-02
Am-241	0.00E+00	6.62E-02	6.62E-03	0.00E+00	2.09E-02

Total Number of Samples	
Random	0
Judgmental	10
QC	2

Judgmental	
SOF >0.5	0
Maximum SOF	0.0472
Minimum SOF	0.0012

Figure 2-37 Survey Unit 7000 Subsurface Sample Locations

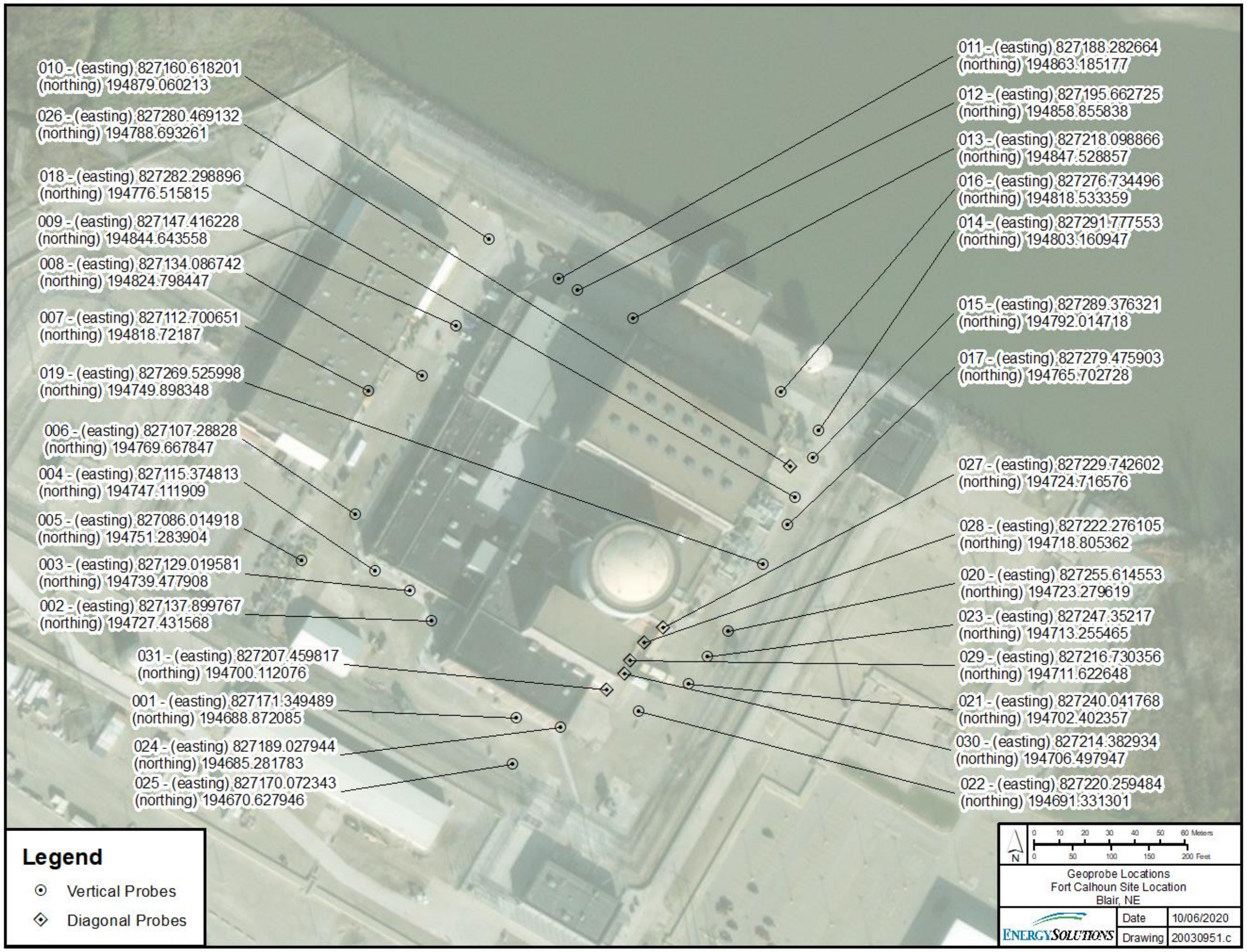


Table 2-80 7000 Judgmental Subsurface Soil Samples Gamma Spectroscopy Results

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7000X-1-CJ-GSB1-001	Co-60	1.44E-02	N/A	U	9.34E-02	5.77E-02	3.80E+00	3.79E-03	0.0129
	Cs-134	0.00E+00	N/A	U	1.55E-01	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	6.87E-02	N/A	U	1.61E-01	7.40E-02	1.10E+01	6.25E-03	
	Eu-152	2.23E-02	N/A	U	3.33E-01	1.58E-01	8.70E+00	2.56E-03	
	Eu-154	0.00E+00	N/A	U	3.25E-01	1.39E-01	8.00E+00	0.00E+00	
	Eu-155	8.97E-02	N/A	U	1.96E-01	8.97E-02	2.80E+02	3.20E-04	
	Am-241	0.00E+00	N/A	U	3.57E-01	1.70E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-001	Co-60	0.00E+00	N/A	U	1.17E-01	6.93E-02	3.80E+00	0.00E+00	0.0373
	Cs-134	0.00E+00	N/A	U	1.62E-01	1.22E-01	5.70E+00	0.00E+00	
	Cs-137	3.95E-02	N/A	U	1.67E-01	7.62E-02	1.10E+01	3.59E-03	
	Eu-152	0.00E+00	N/A	U	3.68E-01	1.74E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.81E-01	1.64E-01	8.00E+00	0.00E+00	
	Eu-155	3.61E-01	N/A	U	4.98E-01	2.40E-01	2.80E+02	1.29E-03	
	Am-241	6.80E-02	N/A	U	4.36E-01	2.08E-01	2.10E+00	3.24E-02	
7000X-1-CJ-GSB3-001	Co-60	3.22E-02	N/A	U	1.40E-01	6.02E-02	3.80E+00	8.47E-03	0.0332
	Cs-134	0.00E+00	N/A	U	1.72E-01	1.19E-01	5.70E+00	0.00E+00	
	Cs-137	1.78E-03	N/A	U	1.60E-01	7.24E-02	1.10E+01	1.62E-04	
	Eu-152	0.00E+00	N/A	U	3.65E-01	1.72E-01	8.70E+00	0.00E+00	
	Eu-154	1.90E-01	N/A	U	5.08E-01	2.27E-01	8.00E+00	2.38E-02	
	Eu-155	1.15E-01	N/A	U	4.84E-01	2.32E-01	2.80E+02	4.11E-04	
	Am-241	8.26E-04	N/A	U	4.30E-01	2.05E-01	2.10E+00	3.93E-04	
7000X-1-CJ-GSB1-002	Co-60	3.90E-02	N/A	U	1.16E-01	4.87E-02	3.80E+00	1.03E-02	0.0413
	Cs-134	0.00E+00	N/A	U	1.55E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	1.99E-02	2.12E-01		9.05E-02	3.78E-02	1.10E+01	1.81E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	2.92E-02	N/A	U	3.33E-01	1.57E-01	8.70E+00	3.36E-03	
	Eu-154	2.00E-01	N/A	U	4.20E-01	1.84E-01	8.00E+00	2.50E-02	
	Eu-155	2.42E-01	N/A	U	4.70E-01	2.26E-01	2.80E+02	8.64E-04	
	Am-241	0.00E+00	N/A	U	3.67E-01	1.74E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-002	Co-60	0.00E+00	N/A	U	1.26E-01	5.92E-02	3.80E+00	0.00E+00	0.0138
	Cs-134	0.00E+00	N/A	U	1.61E-01	1.21E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.47E-01	6.58E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.62E-01	1.71E-01	8.70E+00	0.00E+00	
	Eu-154	9.79E-02	N/A	U	4.53E-01	2.00E-01	8.00E+00	1.22E-02	
	Eu-155	4.30E-01	N/A	U	5.06E-01	2.43E-01	2.80E+02	1.54E-03	
	Am-241	0.00E+00	N/A	U	4.17E-01	1.99E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-002	Co-60	3.20E-02	N/A	U	1.63E-01	7.33E-02	3.80E+00	8.42E-03	0.0228
	Cs-134	0.00E+00	N/A	U	1.69E-01	1.27E-01	5.70E+00	0.00E+00	
	Cs-137	2.82E-02	N/A	U	1.59E-01	7.20E-02	1.10E+01	2.56E-03	
	Eu-152	8.97E-02	N/A	U	3.82E-01	1.82E-01	8.70E+00	1.03E-02	
	Eu-154	0.00E+00	N/A	U	4.07E-01	1.77E-01	8.00E+00	0.00E+00	
	Eu-155	4.27E-01	N/A	U	5.14E-01	2.48E-01	2.80E+02	1.53E-03	
	Am-241	0.00E+00	N/A	U	4.14E-01	1.98E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-003	Co-60	0.00E+00	N/A	U	1.50E-01	6.46E-02	3.80E+00	0.00E+00	0.0193
	Cs-134	0.00E+00	N/A	U	1.77E-01	1.25E-01	5.70E+00	0.00E+00	
	Cs-137	3.42E-02	N/A	U	1.79E-01	8.13E-02	1.10E+01	3.11E-03	
	Eu-152	0.00E+00	N/A	U	3.77E-01	1.78E-01	8.70E+00	0.00E+00	
	Eu-154	1.21E-01	N/A	U	4.65E-01	2.03E-01	8.00E+00	1.51E-02	
	Eu-155	3.11E-01	N/A	U	5.20E-01	2.50E-01	2.80E+02	1.11E-03	
	Am-241	0.00E+00	N/A	U	4.46E-01	2.12E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-003	Co-60	3.19E-02	N/A	U	9.81E-02	6.85E-02	3.80E+00	8.39E-03	0.0759

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.57E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	7.59E-03	N/A	U	1.38E-01	6.23E-02	1.10E+01	6.90E-04	
	Eu-152	0.00E+00	N/A	U	3.40E-01	1.61E-01	8.70E+00	0.00E+00	
	Eu-154	1.79E-02	N/A	U	4.14E-01	1.83E-01	8.00E+00	2.24E-03	
	Eu-155	3.41E-01	N/A	U	3.95E-01	1.89E-01	2.80E+02	1.22E-03	
	Am-241	1.33E-01	N/A	U	4.08E-01	1.95E-01	2.10E+00	6.33E-02	
7000X-1-CJ-GSB3-003	Co-60	2.76E-02	N/A	U	1.30E-01	5.59E-02	3.80E+00	7.26E-03	0.0356
	Cs-134	0.00E+00	N/A	U	1.69E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	7.40E-02	N/A	U	1.49E-01	6.71E-02	1.10E+01	6.73E-03	
	Eu-152	6.92E-02	N/A	U	3.82E-01	1.82E-01	8.70E+00	7.95E-03	
	Eu-154	1.02E-01	N/A	U	4.02E-01	1.76E-01	8.00E+00	1.28E-02	
	Eu-155	2.47E-01	N/A	U	4.96E-01	2.39E-01	2.80E+02	8.82E-04	
	Am-241	0.00E+00	N/A	U	4.49E-01	2.15E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-004	Co-60	5.74E-02	N/A	U	1.36E-01	5.90E-02	3.80E+00	1.51E-02	0.0247
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.09E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.38E-01	6.18E-02	1.10E+01	0.00E+00	
	Eu-152	7.53E-02	N/A	U	3.53E-01	1.67E-01	8.70E+00	8.66E-03	
	Eu-154	0.00E+00	N/A	U	3.99E-01	1.75E-01	8.00E+00	0.00E+00	
	Eu-155	2.72E-01	N/A	U	4.73E-01	2.28E-01	2.80E+02	9.71E-04	
	Am-241	0.00E+00	N/A	U	3.76E-01	1.79E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-004	Co-60	0.00E+00	N/A	U	1.07E-01	5.62E-02	3.80E+00	0.00E+00	0.0011
	Cs-134	0.00E+00	N/A	U	1.63E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.48E-01	6.71E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.06E-01	1.44E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.59E-01	1.55E-01	8.00E+00	0.00E+00	
	Eu-155	3.07E-01	N/A	U	4.67E-01	2.24E-01	2.80E+02	1.10E-03	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	3.96E-01	1.89E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-004	Co-60	2.84E-02	N/A	U	1.23E-01	5.26E-02	3.80E+00	7.47E-03	0.0142
	Cs-134	0.00E+00	N/A	U	1.76E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	6.63E-02	N/A	U	1.47E-01	6.64E-02	1.10E+01	6.03E-03	
	Eu-152	0.00E+00	N/A	U	3.49E-01	1.65E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.22E-01	1.86E-01	8.00E+00	0.00E+00	
	Eu-155	2.03E-01	N/A	U	4.56E-01	2.19E-01	2.80E+02	7.25E-04	
	Am-241	0.00E+00	N/A	U	3.94E-01	1.88E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-005	Co-60	0.00E+00	N/A	U	1.15E-01	6.10E-02	3.80E+00	0.00E+00	0.0073
	Cs-134	0.00E+00	N/A	U	1.71E-01	1.12E-01	5.70E+00	0.00E+00	
	Cs-137	6.70E-02	N/A	U	1.43E-01	6.51E-02	1.10E+01	6.09E-03	
	Eu-152	0.00E+00	N/A	U	3.15E-01	1.49E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.26E-01	1.40E-01	8.00E+00	0.00E+00	
	Eu-155	3.40E-01	N/A	U	4.49E-01	2.16E-01	2.80E+02	1.21E-03	
	Am-241	0.00E+00	N/A	U	3.64E-01	1.74E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-005	Co-60	2.37E-03	N/A	U	1.05E-01	4.39E-02	3.80E+00	6.24E-04	0.0250
	Cs-134	0.00E+00	N/A	U	1.50E-01	1.05E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.32E-01	5.92E-02	1.10E+01	0.00E+00	
	Eu-152	8.14E-02	N/A	U	3.39E-01	1.61E-01	8.70E+00	9.36E-03	
	Eu-154	0.00E+00	N/A	U	3.49E-01	1.51E-01	8.00E+00	0.00E+00	
	Eu-155	1.67E-01	N/A	U	4.48E-01	2.15E-01	2.80E+02	5.96E-04	
	Am-241	3.02E-02	N/A	U	4.09E-01	1.96E-01	2.10E+00	1.44E-02	
7000X-1-CJ-GSB3-005	Co-60	7.97E-02	N/A	U	9.00E-02	6.91E-02	3.80E+00	2.10E-02	0.0321
	Cs-134	0.00E+00	N/A	U	1.26E-01	9.06E-02	5.70E+00	0.00E+00	
	Cs-137	6.68E-03	N/A	U	1.13E-01	5.02E-02	1.10E+01	6.07E-04	
	Eu-152	4.02E-03	N/A	U	2.87E-01	1.35E-01	8.70E+00	4.62E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	3.19E-01	1.37E-01	8.00E+00	0.00E+00	
	Eu-155	1.81E-01	N/A	U	3.73E-01	1.78E-01	2.80E+02	6.46E-04	
	Am-241	1.97E-02	N/A	U	3.52E-01	1.67E-01	2.10E+00	9.38E-03	
7000X-1-CJ-GSB1-006	Co-60	0.00E+00	N/A	U	1.00E-01	5.80E-02	3.80E+00	0.00E+00	0.0054
	Cs-134	0.00E+00	N/A	U	1.59E-01	1.02E-01	5.70E+00	0.00E+00	
	Cs-137	5.75E-02	N/A	U	1.37E-01	6.24E-02	1.10E+01	5.23E-03	
	Eu-152	0.00E+00	N/A	U	3.06E-01	1.45E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.87E-01	1.72E-01	8.00E+00	0.00E+00	
	Eu-155	3.61E-02	N/A	U	4.09E-01	1.96E-01	2.80E+02	1.29E-04	
	Am-241	0.00E+00	N/A	U	3.72E-01	1.77E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-006	Co-60	0.00E+00	N/A	U	9.68E-02	5.96E-02	3.80E+00	0.00E+00	0.0338
	Cs-134	0.00E+00	N/A	U	1.67E-01	1.01E-01	5.70E+00	0.00E+00	
	Cs-137	2.29E-02	N/A	U	1.38E-01	6.24E-02	1.10E+01	2.08E-03	
	Eu-152	7.92E-02	N/A	U	3.44E-01	1.64E-01	8.70E+00	9.10E-03	
	Eu-154	0.00E+00	N/A	U	4.06E-01	1.80E-01	8.00E+00	0.00E+00	
	Eu-155	2.12E-01	N/A	U	4.64E-01	2.24E-01	2.80E+02	7.57E-04	
	Am-241	4.58E-02	N/A	U	3.82E-01	1.83E-01	2.10E+00	2.18E-02	
7000X-1-CJ-GSB3-006	Co-60	8.26E-02	N/A	U	1.09E-01	6.76E-02	3.80E+00	2.17E-02	0.0238
	Cs-134	0.00E+00	N/A	U	1.34E-01	8.95E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.14E-01	5.08E-02	1.10E+01	0.00E+00	
	Eu-152	1.15E-02	N/A	U	3.11E-01	1.47E-01	8.70E+00	1.32E-03	
	Eu-154	0.00E+00	N/A	U	3.56E-01	1.56E-01	8.00E+00	0.00E+00	
	Eu-155	2.21E-01	N/A	U	4.11E-01	1.97E-01	2.80E+02	7.89E-04	
	Am-241	0.00E+00	N/A	U	3.43E-01	1.63E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-007	Co-60	0.00E+00	N/A	U	9.04E-02	5.14E-02	3.80E+00	0.00E+00	0.0054
	Cs-134	0.00E+00	N/A	U	1.15E-01	1.10E-01	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	3.35E-02	N/A	U	1.36E-01	6.16E-02	1.10E+01	3.05E-03	
	Eu-152	1.51E-02	N/A	U	3.29E-01	1.56E-01	8.70E+00	1.74E-03	
	Eu-154	3.91E-03	N/A	U	4.00E-01	1.92E-01	8.00E+00	4.89E-04	
	Eu-155	4.96E-02	N/A	U	4.00E-01	1.92E-01	2.80E+02	1.77E-04	
	Am-241	0.00E+00	N/A	U	3.53E-01	1.68E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-007	Co-60	5.40E-02	N/A	U	1.35E-01	5.61E-02	3.80E+00	1.42E-02	0.1142
	Cs-134	0.00E+00	N/A	U	2.05E-01	1.22E-01	5.70E+00	0.00E+00	
	Cs-137	2.55E-01	N/A	U	2.39E-01	1.11E-01	1.10E+01	2.32E-02	
	Eu-152	3.67E-02	N/A	U	3.71E-01	1.74E-01	8.70E+00	4.22E-03	
	Eu-154	9.38E-02	N/A	U	5.19E-01	2.28E-01	8.00E+00	1.17E-02	
	Eu-155	1.10E-01	N/A	U	5.04E-01	2.41E-01	2.80E+02	3.93E-04	
	Am-241	1.27E-01	N/A	U	4.47E-01	2.13E-01	2.10E+00	6.05E-02	
7000X-1-CJ-GSB3-007	Co-60	3.20E-02	N/A	U	1.22E-01	5.13E-02	3.80E+00	8.42E-03	0.0283
	Cs-134	0.00E+00	N/A	U	1.58E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	2.10E-01	N/A	U	2.12E-01	9.81E-02	1.10E+01	1.91E-02	
	Eu-152	0.00E+00	N/A	U	3.19E-01	1.50E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.40E-01	1.93E-01	8.00E+00	0.00E+00	
	Eu-155	2.19E-01	N/A	U	4.62E-01	2.22E-01	2.80E+02	7.82E-04	
	Am-241	0.00E+00	N/A	U	3.74E-01	1.78E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-008	Co-60	2.80E-02	N/A	U	9.10E-02	3.79E-02	3.80E+00	7.37E-03	0.0130
	Cs-134	0.00E+00	N/A	U	1.33E-01	8.99E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.27E-01	5.75E-02	1.10E+01	0.00E+00	
	Eu-152	4.06E-02	N/A	U	3.01E-01	1.43E-01	8.70E+00	4.67E-03	
	Eu-154	0.00E+00	N/A	U	3.13E-01	1.36E-01	8.00E+00	0.00E+00	
	Eu-155	2.58E-01	N/A	U	3.94E-01	1.89E-01	2.80E+02	9.21E-04	
	Am-241	0.00E+00	N/A	U	3.38E-01	1.61E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7000X-1-CJ-GSB2-008	Co-60	3.67E-02	N/A	U	1.13E-01	6.06E-02	3.80E+00	9.66E-03	0.0341
	Cs-134	0.00E+00	N/A	U	1.62E-01	9.36E-02	5.70E+00	0.00E+00	
	Cs-137	9.85E-02	N/A	U	1.42E-01	6.49E-02	1.10E+01	8.95E-03	
	Eu-152	1.32E-01	N/A	U	3.27E-01	1.55E-01	8.70E+00	1.52E-02	
	Eu-154	0.00E+00	N/A	U	3.21E-01	1.39E-01	8.00E+00	0.00E+00	
	Eu-155	7.54E-02	N/A	U	3.02E-01	1.43E-01	2.80E+02	2.69E-04	
	Am-241	0.00E+00	N/A	U	3.70E-01	1.77E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-008	Co-60	6.68E-02	N/A	U	1.14E-01	8.26E-02	3.80E+00	1.76E-02	0.0293
	Cs-134	0.00E+00	N/A	U	1.90E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	1.15E-01	3.60E-02		1.28E-01	5.64E-02	1.10E+01	1.05E-02	
	Eu-152	0.00E+00	N/A	U	3.53E-01	1.67E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.64E-01	1.56E-01	8.00E+00	0.00E+00	
	Eu-155	3.66E-01	N/A	U	4.87E-01	2.35E-01	2.80E+02	1.31E-03	
	Am-241	0.00E+00	N/A	U	4.00E-01	1.91E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-009	Co-60	2.33E-02	N/A	U	1.12E-01	4.79E-02	3.80E+00	6.13E-03	0.0319
	Cs-134	0.00E+00	N/A	U	1.55E-01	1.07E-01	5.70E+00	0.00E+00	
	Cs-137	1.25E-02	N/A	U	1.25E-01	5.61E-02	1.10E+01	1.14E-03	
	Eu-152	0.00E+00	N/A	U	3.36E-01	1.59E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.71E-01	1.63E-01	8.00E+00	0.00E+00	
	Eu-155	1.69E-01	N/A	U	4.53E-01	2.18E-01	2.80E+02	6.04E-04	
	Am-241	5.05E-02	N/A	U	3.91E-01	1.87E-01	2.10E+00	2.40E-02	
7000X-1-CJ-GSB2-009	Co-60	1.80E-02	N/A	U	1.40E-01	7.08E-02	3.80E+00	4.74E-03	0.0166
	Cs-134	0.00E+00	N/A	U	1.75E-01	1.21E-01	5.70E+00	0.00E+00	
	Cs-137	1.13E-01	N/A	U	2.12E-01	9.80E-02	1.10E+01	1.03E-02	
	Eu-152	0.00E+00	N/A	U	3.68E-01	1.74E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.07E-01	1.75E-01	8.00E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-155	4.43E-01	N/A	U	5.06E-01	2.56E-01	2.80E+02	1.58E-03	
	Am-241	0.00E+00	N/A	U	4.19E-01	2.00E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-009	Co-60	0.00E+00	N/A	U	1.05E-01	6.65E-02	3.80E+00	0.00E+00	0.0307
	Cs-134	0.00E+00	N/A	U	1.45E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	6.31E-02	N/A	U	1.59E-01	7.25E-02	1.10E+01	5.74E-03	
	Eu-152	6.67E-03	N/A	U	3.33E-01	1.57E-01	8.70E+00	7.67E-04	
	Eu-154	7.11E-03	N/A	U	3.83E-01	1.67E-01	8.00E+00	8.89E-04	
	Eu-155	5.96E-02	N/A	U	4.55E-01	2.19E-01	2.80E+02	2.13E-04	
	Am-241	4.85E-02	N/A	U	3.85E-01	1.83E-01	2.10E+00	2.31E-02	
7000X-1-CJ-GSB1-010	Co-60	1.40E-02	N/A	U	8.49E-02	3.58E-02	3.80E+00	3.68E-03	0.0403
	Cs-134	0.00E+00	N/A	U	1.27E-01	7.31E-02	5.70E+00	0.00E+00	
	Cs-137	7.83E-04	N/A	U	9.81E-02	4.38E-02	1.10E+01	7.12E-05	
	Eu-152	2.69E-02	N/A	U	2.84E-01	1.35E-01	8.70E+00	3.09E-03	
	Eu-154	7.66E-02	N/A	U	2.91E-01	1.27E-01	8.00E+00	9.58E-03	
	Eu-155	3.87E-02	N/A	U	3.40E-01	1.63E-01	2.80E+02	1.38E-04	
	Am-241	4.98E-02	N/A	U	2.99E-01	1.42E-01	2.10E+00	2.37E-02	
7000X-1-CJ-GSB2-010	Co-60	0.00E+00	N/A	U	1.16E-01	5.75E-02	3.80E+00	0.00E+00	0.0085
	Cs-134	0.00E+00	N/A	U	1.23E-01	8.30E-02	5.70E+00	0.00E+00	
	Cs-137	8.02E-02	N/A	U	1.38E-01	6.22E-02	1.10E+01	7.29E-03	
	Eu-152	0.00E+00	N/A	U	2.86E-01	1.34E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.30E-01	1.42E-01	8.00E+00	0.00E+00	
	Eu-155	3.37E-01	N/A	U	4.16E-01	2.00E-01	2.80E+02	1.20E-03	
	Am-241	0.00E+00	N/A	U	3.39E-01	1.61E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-010	Co-60	2.21E-02	N/A	U	1.07E-01	4.46E-02	3.80E+00	5.82E-03	0.0067
	Cs-134	0.00E+00	N/A	U	1.58E-01	8.59E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.45E-01	6.55E-02	1.10E+01	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	3.05E-01	1.43E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.06E-01	1.78E-01	8.00E+00	0.00E+00	
	Eu-155	2.36E-01	N/A	U	3.97E-01	1.90E-01	2.80E+02	8.43E-04	
	Am-241	0.00E+00	N/A	U	3.26E-01	1.54E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-011	Co-60	3.94E-02	N/A	U	1.05E-01	6.73E-02	3.80E+00	1.04E-02	0.0333
	Cs-134	0.00E+00	N/A	U	1.44E-01	9.39E-02	5.70E+00	0.00E+00	
	Cs-137	2.58E-02	N/A	U	1.33E-01	6.00E-02	1.10E+01	2.35E-03	
	Eu-152	4.11E-03	N/A	U	3.22E-01	1.52E-01	8.70E+00	4.72E-04	
	Eu-154	0.00E+00	N/A	U	3.86E-01	1.70E-01	8.00E+00	0.00E+00	
	Eu-155	2.11E-01	N/A	U	4.50E-01	2.17E-01	2.80E+02	7.54E-04	
	Am-241	4.06E-02	N/A	U	3.67E-01	1.75E-01	2.10E+00	1.93E-02	
7000X-1-CJ-GSB2-011	Co-60	2.54E-02	N/A	U	1.08E-01	5.90E-02	3.80E+00	6.68E-03	0.0077
	Cs-134	0.00E+00	N/A	U	1.40E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.15E-01	5.07E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.25E-01	1.54E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.32E-01	1.43E-01	8.00E+00	0.00E+00	
	Eu-155	2.92E-01	N/A	U	4.47E-01	2.19E-01	2.80E+02	1.04E-03	
	Am-241	0.00E+00	N/A	U	3.75E-01	1.79E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-011	Co-60	5.61E-02	N/A	U	1.10E-01	4.69E-02	3.80E+00	1.48E-02	0.0209
	Cs-134	0.00E+00	N/A	U	1.74E-01	9.62E-02	5.70E+00	0.00E+00	
	Cs-137	1.17E-02	N/A	U	1.25E-01	5.62E-02	1.10E+01	1.06E-03	
	Eu-152	4.29E-02	N/A	U	3.15E-01	1.49E-01	8.70E+00	4.93E-03	
	Eu-154	0.00E+00	N/A	U	3.50E-01	1.52E-01	8.00E+00	0.00E+00	
	Eu-155	4.24E-02	N/A	U	2.79E-01	1.31E-01	2.80E+02	1.51E-04	
	Am-241	0.00E+00	N/A	U	3.75E-01	1.80E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-012	Co-60	6.62E-02	N/A	U	1.30E-01	5.78E-02	3.80E+00	1.74E-02	0.0222

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.28E-01	9.42E-02	5.70E+00	0.00E+00	
	Cs-137	4.99E-02	N/A	U	1.25E-01	5.66E-02	1.10E+01	4.54E-03	
	Eu-152	0.00E+00	N/A	U	3.04E-01	1.45E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.80E-01	1.70E-01	8.00E+00	0.00E+00	
	Eu-155	5.57E-02	N/A	U	3.96E-01	1.91E-01	2.80E+02	1.99E-04	
	Am-241	0.00E+00	N/A	U	3.63E-01	1.74E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-012	Co-60	0.00E+00	N/A	U	1.24E-01	7.26E-02	3.80E+00	0.00E+00	0.0182
	Cs-134	0.00E+00	N/A	U	1.57E-01	1.17E-01	5.70E+00	0.00E+00	
	Cs-137	5.12E-02	N/A	U	1.52E-01	6.83E-02	1.10E+01	4.65E-03	
	Eu-152	0.00E+00	N/A	U	3.85E-01	1.82E-01	8.70E+00	0.00E+00	
	Eu-154	1.03E-01	N/A	U	3.63E-01	1.55E-01	8.00E+00	1.29E-02	
	Eu-155	1.90E-01	N/A	U	4.98E-01	2.39E-01	2.80E+02	6.79E-04	
	Am-241	0.00E+00	N/A	U	4.64E-01	2.22E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-012	Co-60	3.33E-02	N/A	U	1.14E-01	6.76E-02	3.80E+00	8.76E-03	0.0308
	Cs-134	0.00E+00	N/A	U	1.64E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.41E-01	6.32E-02	1.10E+01	0.00E+00	
	Eu-152	7.28E-02	N/A	U	3.36E-01	1.59E-01	8.70E+00	8.37E-03	
	Eu-154	9.76E-02	N/A	U	4.17E-01	1.83E-01	8.00E+00	1.22E-02	
	Eu-155	4.19E-01	N/A	U	4.63E-01	2.37E-01	2.80E+02	1.50E-03	
	Am-241	0.00E+00	N/A	U	3.83E-01	1.83E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-013	Co-60	4.44E-02	N/A	U	1.19E-01	6.90E-02	3.80E+00	1.17E-02	0.0215
	Cs-134	0.00E+00	N/A	U	1.62E-01	1.11E-01	5.70E+00	0.00E+00	
	Cs-137	1.02E-01	N/A	U	1.44E-01	6.50E-02	1.10E+01	9.27E-03	
	Eu-152	0.00E+00	N/A	U	3.51E-01	1.66E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.92E-01	1.72E-01	8.00E+00	0.00E+00	
	Eu-155	1.65E-01	N/A	U	2.72E-01	1.27E-01	2.80E+02	5.89E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	4.19E-01	2.00E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-013	Co-60	1.92E-02	N/A	U	1.22E-01	5.18E-02	3.80E+00	5.05E-03	0.0195
	Cs-134	0.00E+00	N/A	U	1.81E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	4.59E-02	N/A	U	1.50E-01	6.78E-02	1.10E+01	4.17E-03	
	Eu-152	8.45E-02	N/A	U	3.78E-01	1.80E-01	8.70E+00	9.71E-03	
	Eu-154	0.00E+00	N/A	U	4.09E-01	1.79E-01	8.00E+00	0.00E+00	
	Eu-155	1.68E-01	N/A	U	4.77E-01	2.30E-01	2.80E+02	6.00E-04	
	Am-241	0.00E+00	N/A	U	4.14E-01	1.98E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-013	Co-60	0.00E+00	N/A	U	1.13E-01	5.57E-02	3.80E+00	0.00E+00	0.0096
	Cs-134	0.00E+00	N/A	U	1.58E-01	1.17E-01	5.70E+00	0.00E+00	
	Cs-137	3.40E-02	N/A	U	1.44E-01	6.49E-02	1.10E+01	3.09E-03	
	Eu-152	4.54E-02	N/A	U	3.53E-01	1.68E-01	8.70E+00	5.22E-03	
	Eu-154	0.00E+00	N/A	U	4.14E-01	1.83E-01	8.00E+00	0.00E+00	
	Eu-155	3.52E-01	N/A	U	5.04E-01	2.43E-01	2.80E+02	1.26E-03	
	Am-241	0.00E+00	N/A	U	3.82E-01	1.82E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-014	Co-60	1.78E-02	N/A	U	9.74E-02	4.08E-02	3.80E+00	4.68E-03	0.0534
	Cs-134	1.12E-02	N/A	U	1.29E-01	8.24E-02	5.70E+00	1.96E-03	
	Cs-137	1.53E-02	1.49E-02		6.37E-02	2.57E-02	1.10E+01	1.39E-03	
	Eu-152	8.87E-03	N/A	U	2.78E-01	1.31E-01	8.70E+00	1.02E-03	
	Eu-154	0.00E+00	N/A	U	3.24E-01	1.40E-01	8.00E+00	0.00E+00	
	Eu-155	5.46E-02	N/A	U	3.54E-01	1.69E-01	2.80E+02	1.95E-04	
	Am-241	9.27E-02	N/A	U	3.24E-01	1.54E-01	2.10E+00	4.41E-02	
7000X-1-CJ-GSB2-014	Co-60	1.64E-02	N/A	U	8.38E-02	5.33E-02	3.80E+00	4.32E-03	0.0058
	Cs-134	0.00E+00	N/A	U	1.17E-01	7.24E-02	5.70E+00	0.00E+00	
	Cs-137	9.58E-03	N/A	U	1.06E-01	4.74E-02	1.10E+01	8.71E-04	
	Eu-152	0.00E+00	N/A	U	2.50E-01	1.17E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	3.55E-01	1.58E-01	8.00E+00	0.00E+00	
	Eu-155	1.66E-01	N/A	U	2.41E-01	1.13E-01	2.80E+02	5.93E-04	
	Am-241	0.00E+00	N/A	U	2.83E-01	1.34E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-014	Co-60	0.00E+00	N/A	U	9.78E-02	4.64E-02	3.80E+00	0.00E+00	0.0037
	Cs-134	0.00E+00	N/A	U	1.15E-01	7.59E-02	5.70E+00	0.00E+00	
	Cs-137	3.31E-02	N/A	U	1.15E-01	5.19E-02	1.10E+01	3.01E-03	
	Eu-152	0.00E+00	N/A	U	2.61E-01	1.23E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	2.84E-01	1.23E-01	8.00E+00	0.00E+00	
	Eu-155	2.01E-01	N/A	U	3.64E-01	1.75E-01	2.80E+02	7.18E-04	
	Am-241	0.00E+00	N/A	U	2.94E-01	1.40E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-015	Co-60	2.39E-02	N/A	U	1.19E-01	5.57E-02	3.80E+00	6.29E-03	0.0306
	Cs-134	0.00E+00	N/A	U	1.43E-01	9.95E-02	5.70E+00	0.00E+00	
	Cs-137	6.97E-02	N/A	U	1.44E-01	6.58E-02	1.10E+01	6.34E-03	
	Eu-152	3.81E-02	N/A	U	2.95E-01	1.39E-01	8.70E+00	4.38E-03	
	Eu-154	6.81E-02	N/A	U	4.05E-01	1.81E-01	8.00E+00	8.51E-03	
	Eu-155	9.22E-02	N/A	U	3.13E-01	1.49E-01	2.80E+02	3.29E-04	
	Am-241	9.93E-03	N/A	U	3.33E-01	1.59E-01	2.10E+00	4.73E-03	
7000X-1-CJ-GSB2-015	Co-60	1.26E-02	N/A	U	8.93E-02	3.79E-02	3.80E+00	3.32E-03	0.0095
	Cs-134	0.00E+00	N/A	U	1.00E-01	6.97E-02	5.70E+00	0.00E+00	
	Cs-137	6.48E-03	N/A	U	1.05E-01	4.69E-02	1.10E+01	5.89E-04	
	Eu-152	4.65E-02	N/A	U	2.83E-01	1.34E-01	8.70E+00	5.34E-03	
	Eu-154	0.00E+00	N/A	U	2.74E-01	1.18E-01	8.00E+00	0.00E+00	
	Eu-155	1.76E-02	N/A	U	3.49E-01	1.67E-01	2.80E+02	6.29E-05	
	Am-241	3.33E-04	N/A	U	2.94E-01	1.40E-01	2.10E+00	1.59E-04	
7000X-1-CJ-GSB3-015	Co-60	2.12E-02	N/A	U	1.00E-01	4.33E-02	3.80E+00	5.58E-03	0.0182
	Cs-134	0.00E+00	N/A	U	1.23E-01	6.95E-02	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	0.00E+00	N/A	U	1.08E-01	4.83E-02	1.10E+01	0.00E+00	
	Eu-152	8.30E-03	N/A	U	2.70E-01	1.27E-01	8.70E+00	9.54E-04	
	Eu-154	8.47E-02	N/A	U	3.78E-01	1.70E-01	8.00E+00	1.06E-02	
	Eu-155	3.10E-01	N/A	U	3.67E-01	1.76E-01	2.80E+02	1.11E-03	
	Am-241	0.00E+00	N/A	U	3.06E-01	1.45E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-016	Co-60	0.00E+00	N/A	U	9.34E-02	6.78E-02	3.80E+00	0.00E+00	0.0021
	Cs-134	0.00E+00	N/A	U	1.35E-01	9.82E-02	5.70E+00	0.00E+00	
	Cs-137	1.74E-02	N/A	U	1.19E-01	5.30E-02	1.10E+01	1.58E-03	
	Eu-152	0.00E+00	N/A	U	3.21E-01	1.52E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.68E-01	1.61E-01	8.00E+00	0.00E+00	
	Eu-155	1.34E-01	N/A	U	4.13E-01	1.98E-01	2.80E+02	4.79E-04	
	Am-241	0.00E+00	N/A	U	3.43E-01	1.63E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-016	Co-60	3.38E-02	N/A	U	1.20E-01	5.59E-02	3.80E+00	8.89E-03	0.0273
	Cs-134	0.00E+00	N/A	U	1.37E-01	9.19E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.26E-01	5.70E-02	1.10E+01	0.00E+00	
	Eu-152	6.15E-02	N/A	U	3.04E-01	1.44E-01	8.70E+00	7.07E-03	
	Eu-154	8.91E-02	N/A	U	3.38E-01	1.48E-01	8.00E+00	1.11E-02	
	Eu-155	4.71E-02	N/A	U	3.94E-01	1.89E-01	2.80E+02	1.68E-04	
	Am-241	0.00E+00	N/A	U	3.45E-01	1.65E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-016	Co-60	3.81E-02	N/A	U	1.06E-01	6.31E-02	3.80E+00	1.00E-02	0.0370
	Cs-134	7.96E-03	N/A	U	1.48E-01	9.41E-02	5.70E+00	1.40E-03	
	Cs-137	3.90E-02	N/A	U	1.29E-01	5.79E-02	1.10E+01	3.55E-03	
	Eu-152	0.00E+00	N/A	U	2.81E-01	1.32E-01	8.70E+00	0.00E+00	
	Eu-154	1.71E-01	N/A	U	3.81E-01	1.68E-01	8.00E+00	2.14E-02	
	Eu-155	1.89E-01	N/A	U	4.10E-01	1.97E-01	2.80E+02	6.75E-04	
	Am-241	0.00E+00	N/A	U	3.62E-01	1.73E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7000X-1-CJ-GSB1-017	Co-60	1.83E-02	N/A	U	1.05E-01	4.33E-02	3.80E+00	4.82E-03	0.0093
	Cs-134	0.00E+00	N/A	U	1.95E-01	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	4.72E-02	N/A	U	1.51E-01	6.83E-02	1.10E+01	4.29E-03	
	Eu-152	0.00E+00	N/A	U	3.38E-01	1.60E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.26E-01	1.88E-01	8.00E+00	0.00E+00	
	Eu-155	6.12E-02	N/A	U	4.40E-01	2.11E-01	2.80E+02	2.19E-04	
	Am-241	0.00E+00	N/A	U	3.94E-01	1.88E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-017	Co-60	2.69E-02	N/A	U	1.44E-01	7.10E-02	3.80E+00	7.08E-03	0.0293
	Cs-134	2.42E-03	N/A	U	1.66E-01	1.15E-01	5.70E+00	4.25E-04	
	Cs-137	3.82E-02	N/A	U	1.56E-01	7.06E-02	1.10E+01	3.47E-03	
	Eu-152	1.11E-01	N/A	U	3.60E-01	1.70E-01	8.70E+00	1.28E-02	
	Eu-154	0.00E+00	N/A	U	3.73E-01	1.60E-01	8.00E+00	0.00E+00	
	Eu-155	1.89E-01	N/A	U	4.90E-01	2.36E-01	2.80E+02	6.75E-04	
	Am-241	1.03E-02	N/A	U	4.03E-01	1.92E-01	2.10E+00	4.90E-03	
7000X-1-CJ-GSB3-017	Co-60	1.68E-02	N/A	U	1.37E-01	7.44E-02	3.80E+00	4.42E-03	0.0168
	Cs-134	0.00E+00	N/A	U	1.65E-01	1.01E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.21E-01	5.31E-02	1.10E+01	0.00E+00	
	Eu-152	4.80E-02	N/A	U	3.53E-01	1.67E-01	8.70E+00	5.52E-03	
	Eu-154	4.95E-02	N/A	U	4.32E-01	1.91E-01	8.00E+00	6.19E-03	
	Eu-155	1.93E-01	N/A	U	4.36E-01	2.08E-01	2.80E+02	6.89E-04	
	Am-241	0.00E+00	N/A	U	3.95E-01	1.87E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-018	Co-60	5.77E-02	N/A	U	9.81E-02	6.66E-02	3.80E+00	1.52E-02	0.0409
	Cs-134	0.00E+00	N/A	U	1.54E-01	9.49E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.13E-01	4.98E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.24E-01	1.53E-01	8.70E+00	0.00E+00	
	Eu-154	1.96E-01	N/A	U	3.94E-01	1.74E-01	8.00E+00	2.45E-02	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-155	3.28E-01	N/A	U	4.36E-01	2.10E-01	2.80E+02	1.17E-03	
	Am-241	0.00E+00	N/A	U	3.75E-01	1.79E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-018	Co-60	2.98E-04	N/A	U	1.16E-01	4.84E-02	3.80E+00	7.84E-05	0.0047
	Cs-134	0.00E+00	N/A	U	1.58E-01	1.13E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.32E-01	5.86E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.41E-01	1.61E-01	8.70E+00	0.00E+00	
	Eu-154	2.64E-02	N/A	U	4.26E-01	1.87E-01	8.00E+00	3.30E-03	
	Eu-155	3.73E-01	N/A	U	4.54E-01	2.23E-01	2.80E+02	1.33E-03	
	Am-241	0.00E+00	N/A	U	3.78E-01	1.80E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-018	Co-60	0.00E+00	N/A	U	9.23E-02	5.60E-02	3.80E+00	0.00E+00	0.0075
	Cs-134	0.00E+00	N/A	U	1.14E-01	6.90E-02	5.70E+00	0.00E+00	
	Cs-137	9.93E-03	N/A	U	1.14E-01	5.11E-02	1.10E+01	9.03E-04	
	Eu-152	0.00E+00	N/A	U	2.69E-01	1.27E-01	8.70E+00	0.00E+00	
	Eu-154	2.08E-02	N/A	U	3.72E-01	1.67E-01	8.00E+00	2.60E-03	
	Eu-155	8.42E-02	N/A	U	3.40E-01	1.63E-01	2.80E+02	3.01E-04	
	Am-241	7.71E-03	N/A	U	3.32E-01	1.58E-01	2.10E+00	3.67E-03	
7000X-1-CJ-GSB1-019	Co-60	1.76E-02	N/A	U	1.36E-01	6.59E-02	3.80E+00	4.63E-03	0.0119
	Cs-134	0.00E+00	N/A	U	1.81E-01	1.23E-01	5.70E+00	0.00E+00	
	Cs-137	4.84E-02	N/A	U	1.50E-01	6.74E-02	1.10E+01	4.40E-03	
	Eu-152	2.27E-02	N/A	U	3.36E-01	1.58E-01	8.70E+00	2.61E-03	
	Eu-154	0.00E+00	N/A	U	4.53E-01	1.99E-01	8.00E+00	0.00E+00	
	Eu-155	7.88E-02	N/A	U	4.34E-01	2.07E-01	2.80E+02	2.81E-04	
	Am-241	0.00E+00	N/A	U	3.67E-01	1.74E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-019	Co-60	2.85E-02	N/A	U	1.17E-01	6.77E-02	3.80E+00	7.50E-03	0.0393
	Cs-134	0.00E+00	N/A	U	1.49E-01	8.66E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.12E-01	4.90E-02	1.10E+01	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	9.51E-02	N/A	U	3.47E-01	1.64E-01	8.70E+00	1.09E-02	
	Eu-154	0.00E+00	N/A	U	4.10E-01	1.81E-01	8.00E+00	0.00E+00	
	Eu-155	1.13E-01	N/A	U	4.12E-01	1.96E-01	2.80E+02	4.04E-04	
	Am-241	4.29E-02	N/A	U	3.95E-01	1.88E-01	2.10E+00	2.04E-02	
7000X-1-CJ-GSB3-019	Co-60	8.44E-02	N/A	U	1.22E-01	8.37E-02	3.80E+00	2.22E-02	0.0307
	Cs-134	0.00E+00	N/A	U	1.41E-01	9.45E-02	5.70E+00	0.00E+00	
	Cs-137	1.43E-02	N/A	U	1.41E-01	6.31E-02	1.10E+01	1.30E-03	
	Eu-152	0.00E+00	N/A	U	3.53E-01	1.66E-01	8.70E+00	0.00E+00	
	Eu-154	5.73E-02	N/A	U	4.28E-01	1.87E-01	8.00E+00	7.16E-03	
	Eu-155	0.00E+00	N/A	U	4.44E-01	2.12E-01	2.80E+02	0.00E+00	
	Am-241	0.00E+00	N/A	U	4.37E-01	2.08E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-020	Co-60	5.94E-02	N/A	U	1.45E-01	6.31E-02	3.80E+00	1.56E-02	0.0618
	Cs-134	0.00E+00	N/A	U	1.65E-01	1.15E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.30E-01	5.75E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.38E-01	1.59E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.27E-01	1.87E-01	8.00E+00	0.00E+00	
	Eu-155	1.07E-01	N/A	U	3.42E-01	1.62E-01	2.80E+02	3.82E-04	
	Am-241	9.61E-02	N/A	U	4.16E-01	1.98E-01	2.10E+00	4.58E-02	
7000X-1-CJ-GSB2-020	Co-60	7.83E-02	N/A	U	1.67E-01	8.58E-02	3.80E+00	2.06E-02	0.0279
	Cs-134	0.00E+00	N/A	U	1.89E-01	1.23E-01	5.70E+00	0.00E+00	
	Cs-137	7.37E-02	N/A	U	1.80E-01	8.19E-02	1.10E+01	6.70E-03	
	Eu-152	0.00E+00	N/A	U	3.54E-01	1.66E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.27E-01	1.85E-01	8.00E+00	0.00E+00	
	Eu-155	1.57E-01	N/A	U	4.30E-01	2.05E-01	2.80E+02	5.61E-04	
	Am-241	0.00E+00	N/A	U	4.40E-01	2.10E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-020	Co-60	6.47E-02	N/A	U	1.16E-01	5.00E-02	3.80E+00	1.70E-02	0.0264

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-134	0.00E+00	N/A	U	1.51E-01	9.46E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.37E-01	6.22E-02	1.10E+01	0.00E+00	
	Eu-152	7.35E-02	N/A	U	3.44E-01	1.63E-01	8.70E+00	8.45E-03	
	Eu-154	0.00E+00	N/A	U	3.85E-01	1.70E-01	8.00E+00	0.00E+00	
	Eu-155	2.58E-01	N/A	U	4.41E-01	2.12E-01	2.80E+02	9.21E-04	
	Am-241	0.00E+00	N/A	U	3.84E-01	1.83E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-021	Co-60	1.59E-02	N/A	U	1.10E-01	5.92E-02	3.80E+00	4.18E-03	0.0243
	Cs-134	0.00E+00	N/A	U	1.36E-01	9.94E-02	5.70E+00	0.00E+00	
	Cs-137	4.19E-02	N/A	U	1.40E-01	6.35E-02	1.10E+01	3.81E-03	
	Eu-152	3.77E-02	N/A	U	3.25E-01	1.54E-01	8.70E+00	4.33E-03	
	Eu-154	7.74E-02	N/A	U	3.78E-01	1.66E-01	8.00E+00	9.68E-03	
	Eu-155	2.79E-01	N/A	U	4.31E-01	2.07E-01	2.80E+02	9.96E-04	
	Am-241	2.69E-03	N/A	U	3.93E-01	1.88E-01	2.10E+00	1.28E-03	
7000X-1-CJ-GSB2-021	Co-60	0.00E+00	N/A	U	1.16E-01	6.73E-02	3.80E+00	0.00E+00	0.0056
	Cs-134	0.00E+00	N/A	U	1.62E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	5.91E-02	N/A	U	1.53E-01	6.97E-02	1.10E+01	5.37E-03	
	Eu-152	0.00E+00	N/A	U	3.16E-01	1.49E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.40E-01	1.46E-01	8.00E+00	0.00E+00	
	Eu-155	6.42E-02	N/A	U	4.73E-01	2.28E-01	2.80E+02	2.29E-04	
	Am-241	0.00E+00	N/A	U	4.01E-01	1.91E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-021	Co-60	1.63E-02	N/A	U	1.23E-01	7.56E-02	3.80E+00	4.29E-03	0.0062
	Cs-134	0.00E+00	N/A	U	1.61E-01	1.02E-01	5.70E+00	0.00E+00	
	Cs-137	1.16E-02	N/A	U	1.44E-01	6.47E-02	1.10E+01	1.05E-03	
	Eu-152	4.42E-03	N/A	U	3.53E-01	1.67E-01	8.70E+00	5.08E-04	
	Eu-154	0.00E+00	N/A	U	3.99E-01	1.75E-01	8.00E+00	0.00E+00	
	Eu-155	9.41E-02	N/A	U	4.73E-01	2.27E-01	2.80E+02	3.36E-04	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Am-241	0.00E+00	N/A	U	3.89E-01	1.85E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-022	Co-60	4.39E-04	N/A	U	1.32E-01	6.64E-02	3.80E+00	1.16E-04	0.0032
	Cs-134	0.00E+00	N/A	U	1.55E-01	1.08E-01	5.70E+00	0.00E+00	
	Cs-137	2.40E-02	N/A	U	1.35E-01	6.09E-02	1.10E+01	2.18E-03	
	Eu-152	0.00E+00	N/A	U	3.22E-01	1.52E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.20E-01	1.37E-01	8.00E+00	0.00E+00	
	Eu-155	2.48E-01	N/A	U	4.73E-01	2.28E-01	2.80E+02	8.86E-04	
	Am-241	0.00E+00	N/A	U	3.80E-01	1.81E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-022	Co-60	7.07E-03	N/A	U	1.11E-01	4.74E-02	3.80E+00	1.86E-03	0.0044
	Cs-134	0.00E+00	N/A	U	1.42E-01	9.63E-02	5.70E+00	0.00E+00	
	Cs-137	2.04E-02	N/A	U	1.33E-01	6.04E-02	1.10E+01	1.85E-03	
	Eu-152	0.00E+00	N/A	U	2.96E-01	1.40E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.75E-01	1.65E-01	8.00E+00	0.00E+00	
	Eu-155	1.92E-01	N/A	U	4.27E-01	2.07E-01	2.80E+02	6.86E-04	
	Am-241	0.00E+00	N/A	U	3.70E-01	1.77E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-022	Co-60	3.03E-02	N/A	U	1.43E-01	7.73E-02	3.80E+00	7.97E-03	0.0284
	Cs-134	0.00E+00	N/A	U	1.89E-01	1.16E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.47E-01	6.48E-02	1.10E+01	0.00E+00	
	Eu-152	1.73E-01	N/A	U	3.69E-01	1.74E-01	8.70E+00	1.99E-02	
	Eu-154	0.00E+00	N/A	U	3.61E-01	1.50E-01	8.00E+00	0.00E+00	
	Eu-155	1.40E-01	N/A	U	4.99E-01	2.39E-01	2.80E+02	5.00E-04	
	Am-241	0.00E+00	N/A	U	3.78E-01	1.79E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-023	Co-60	4.59E-02	N/A	U	1.44E-01	6.43E-02	3.80E+00	1.21E-02	0.0198
	Cs-134	0.00E+00	N/A	U	1.48E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	6.99E-02	N/A	U	1.46E-01	6.58E-02	1.10E+01	6.35E-03	
	Eu-152	0.00E+00	N/A	U	3.02E-01	1.42E-01	8.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-154	0.00E+00	N/A	U	3.68E-01	1.59E-01	8.00E+00	0.00E+00	
	Eu-155	3.69E-01	N/A	U	4.44E-01	2.13E-01	2.80E+02	1.32E-03	
	Am-241	0.00E+00	N/A	U	3.67E-01	1.75E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-023	Co-60	0.00E+00	N/A	U	1.19E-01	6.34E-02	3.80E+00	0.00E+00	0.0171
	Cs-134	0.00E+00	N/A	U	1.38E-01	9.85E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.36E-01	6.07E-02	1.10E+01	0.00E+00	
	Eu-152	7.30E-02	N/A	U	3.25E-01	1.53E-01	8.70E+00	8.39E-03	
	Eu-154	1.77E-02	N/A	U	3.88E-01	1.67E-01	8.00E+00	2.21E-03	
	Eu-155	2.93E-01	N/A	U	4.48E-01	2.15E-01	2.80E+02	1.05E-03	
	Am-241	1.14E-02	N/A	U	3.53E-01	1.67E-01	2.10E+00	5.43E-03	
7000X-1-CJ-GSB3-023	Co-60	5.19E-02	N/A	U	1.27E-01	5.81E-02	3.80E+00	1.37E-02	0.0179
	Cs-134	0.00E+00	N/A	U	1.69E-01	1.16E-01	5.70E+00	0.00E+00	
	Cs-137	4.27E-02	N/A	U	1.57E-01	7.04E-02	1.10E+01	3.88E-03	
	Eu-152	0.00E+00	N/A	U	3.26E-01	1.53E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.90E-01	1.66E-01	8.00E+00	0.00E+00	
	Eu-155	9.37E-02	N/A	U	4.70E-01	2.25E-01	2.80E+02	3.35E-04	
	Am-241	0.00E+00	N/A	U	3.79E-01	1.80E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-024	Co-60	1.46E-02	N/A	U	1.22E-01	5.20E-02	3.80E+00	3.84E-03	0.0110
	Cs-134	0.00E+00	N/A	U	1.45E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	2.73E-02	N/A	U	1.43E-01	6.45E-02	1.10E+01	2.48E-03	
	Eu-152	0.00E+00	N/A	U	3.10E-01	1.46E-01	8.70E+00	0.00E+00	
	Eu-154	3.35E-02	N/A	U	3.46E-01	1.49E-01	8.00E+00	4.19E-03	
	Eu-155	1.45E-01	N/A	U	4.48E-01	2.15E-01	2.80E+02	5.18E-04	
	Am-241	0.00E+00	N/A	U	3.82E-01	1.82E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-024	Co-60	0.00E+00	N/A	U	9.59E-02	5.93E-02	3.80E+00	0.00E+00	0.0779
	Cs-134	0.00E+00	N/A	U	1.68E-01	9.98E-02	5.70E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Cs-137	0.00E+00	N/A	U	1.37E-01	6.18E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	3.26E-01	1.54E-01	8.70E+00	0.00E+00	
	Eu-154	6.65E-02	N/A	U	4.33E-01	1.93E-01	8.00E+00	8.31E-03	
	Eu-155	1.61E-01	N/A	U	4.45E-01	2.27E-01	2.80E+02	5.75E-04	
	Am-241	1.45E-01	N/A	U	4.08E-01	1.95E-01	2.10E+00	6.90E-02	
7000X-1-CJ-GSB3-024	Co-60	6.46E-02	N/A	U	1.51E-01	6.59E-02	3.80E+00	1.70E-02	0.0363
	Cs-134	0.00E+00	N/A	U	1.72E-01	1.14E-01	5.70E+00	0.00E+00	
	Cs-137	5.71E-02	N/A	U	1.57E-01	7.07E-02	1.10E+01	5.19E-03	
	Eu-152	0.00E+00	N/A	U	3.40E-01	1.60E-01	8.70E+00	0.00E+00	
	Eu-154	1.08E-01	N/A	U	4.45E-01	1.95E-01	8.00E+00	1.35E-02	
	Eu-155	1.75E-01	N/A	U	4.96E-01	2.38E-01	2.80E+02	6.25E-04	
	Am-241	0.00E+00	N/A	U	4.02E-01	1.91E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-025	Co-60	2.04E-02	N/A	U	9.59E-02	6.58E-02	3.80E+00	5.37E-03	0.0172
	Cs-134	0.00E+00	N/A	U	1.27E-01	1.03E-01	5.70E+00	0.00E+00	
	Cs-137	4.91E-02	N/A	U	1.49E-01	6.77E-02	1.10E+01	4.46E-03	
	Eu-152	0.00E+00	N/A	U	3.02E-01	1.42E-01	8.70E+00	0.00E+00	
	Eu-154	5.76E-02	N/A	U	1.96E-01	4.39E-01	8.00E+00	7.20E-03	
	Eu-155	4.16E-02	N/A	U	2.71E-01	1.27E-01	2.80E+02	1.49E-04	
	Am-241	0.00E+00	N/A	U	3.50E-01	1.66E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-025	Co-60	2.59E-02	N/A	U	1.30E-01	5.66E-02	3.80E+00	6.82E-03	0.0238
	Cs-134	0.00E+00	N/A	U	1.50E-01	1.04E-01	5.70E+00	0.00E+00	
	Cs-137	4.30E-02	N/A	U	1.42E-01	6.40E-02	1.10E+01	3.91E-03	
	Eu-152	1.12E-01	N/A	U	3.46E-01	1.64E-01	8.70E+00	1.29E-02	
	Eu-154	0.00E+00	N/A	U	3.77E-01	1.65E-01	8.00E+00	0.00E+00	
	Eu-155	5.23E-02	N/A	U	4.56E-01	2.19E-01	2.80E+02	1.87E-04	
	Am-241	0.00E+00	N/A	U	3.86E-01	1.84E-01	2.10E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
7000X-1-CJ-GSB3-025	Co-60	3.01E-02	N/A	U	1.25E-01	5.40E-02	3.80E+00	7.92E-03	0.0255
	Cs-134	0.00E+00	N/A	U	1.64E-01	1.09E-01	5.70E+00	0.00E+00	
	Cs-137	6.14E-02	N/A	U	1.51E-01	6.83E-02	1.10E+01	5.58E-03	
	Eu-152	1.02E-01	N/A	U	3.51E-01	1.66E-01	8.70E+00	1.17E-02	
	Eu-154	0.00E+00	N/A	U	3.55E-01	1.53E-01	8.00E+00	0.00E+00	
	Eu-155	7.67E-02	N/A	U	4.53E-01	2.18E-01	2.80E+02	2.74E-04	
	Am-241	0.00E+00	N/A	U	3.99E-01	1.90E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-029	Co-60	3.16E-02	N/A	U	1.03E-01	5.74E-02	3.80E+00	8.32E-03	0.0141
	Cs-134	0.00E+00	N/A	U	1.39E-01	9.30E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.13E-01	5.03E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.87E-01	1.35E-01	8.70E+00	0.00E+00	
	Eu-154	4.55E-02	N/A	U	4.04E-01	1.81E-01	8.00E+00	5.69E-03	
	Eu-155	3.29E-02	N/A	U	3.91E-01	1.88E-01	2.80E+02	1.18E-04	
	Am-241	0.00E+00	N/A	U	3.31E-01	1.58E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-029	Co-60	1.40E-02	N/A	U	7.91E-02	5.36E-02	3.80E+00	3.68E-03	0.0042
	Cs-134	0.00E+00	N/A	U	1.08E-01	8.72E-02	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.09E-01	4.84E-02	1.10E+01	0.00E+00	
	Eu-152	0.00E+00	N/A	U	2.64E-01	1.23E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.55E-01	1.56E-01	8.00E+00	0.00E+00	
	Eu-155	1.34E-01	N/A	U	3.84E-01	1.84E-01	2.80E+02	4.79E-04	
	Am-241	0.00E+00	N/A	U	3.52E-01	1.67E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-031	Co-60	1.70E-02	N/A	U	1.09E-01	7.05E-02	3.80E+00	4.47E-03	0.0129
	Cs-134	0.00E+00	N/A	U	1.81E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	9.03E-02	N/A	U	1.52E-01	6.86E-02	1.10E+01	8.21E-03	
	Eu-152	0.00E+00	N/A	U	3.53E-01	1.67E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.80E-01	1.64E-01	8.00E+00	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-155	7.32E-02	N/A	U	4.61E-01	2.21E-01	2.80E+02	2.61E-04	
	Am-241	0.00E+00	N/A	U	3.79E-01	1.80E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-031	Co-60	6.84E-02	N/A	U	1.33E-01	5.72E-02	3.80E+00	1.80E-02	0.0230
	Cs-134	0.00E+00	N/A	U	1.93E-01	1.10E-01	5.70E+00	0.00E+00	
	Cs-137	3.79E-02	N/A	U	1.38E-01	6.69E-02	1.10E+01	3.45E-03	
	Eu-152	0.00E+00	N/A	U	3.34E-01	1.58E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	4.55E-01	2.02E-01	8.00E+00	0.00E+00	
	Eu-155	4.23E-01	N/A	U	4.70E-01	2.41E-01	2.80E+02	1.51E-03	
	Am-241	0.00E+00	N/A	U	3.85E-01	1.84E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB3-031	Co-60	3.64E-02	N/A	U	1.24E-01	5.45E-02	3.80E+00	9.58E-03	0.0344
	Cs-134	0.00E+00	N/A	U	1.80E-01	1.06E-01	5.70E+00	0.00E+00	
	Cs-137	3.23E-02	N/A	U	1.37E-01	6/14E-2	1.10E+01	2.94E-03	
	Eu-152	0.00E+00	N/A	U	3.19E-01	1.51E-01	8.70E+00	0.00E+00	
	Eu-154	1.65E-01	N/A	U	4.12E-01	1.81E-01	8.00E+00	2.06E-02	
	Eu-155	3.57E-01	N/A	U	4.55E-01	2.22E-01	2.80E+02	1.28E-03	
	Am-241	0.00E+00	N/A	U	3.65E-01	1.74E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB1-031A	Co-60	5.19E-02	N/A	U	1.12E-01	5.57E-02	3.80E+00	1.37E-02	0.0156
	Cs-134	0.00E+00	N/A	U	1.68E-01	1.18E-01	5.70E+00	0.00E+00	
	Cs-137	1.54E-02	N/A	U	1.35E-01	5.96E-02	1.10E+01	1.40E-03	
	Eu-152	0.00E+00	N/A	U	3.36E-01	1.58E-01	8.70E+00	0.00E+00	
	Eu-154	0.00E+00	N/A	U	3.64E-01	1.54E-01	8.00E+00	0.00E+00	
	Eu-155	1.42E-01	N/A	U	4.39E-01	2.10E-01	2.80E+02	5.07E-04	
	Am-241	0.00E+00	N/A	U	3.58E-01	1.69E-01	2.10E+00	0.00E+00	
7000X-1-CJ-GSB2-031A	Co-60	0.00E+00	N/A	U	1.57E-01	6.37E-02	3.80E+00	0.00E+00	0.0039
	Cs-134	0.00E+00	N/A	U	1.86E-01	1.32E-01	5.70E+00	0.00E+00	
	Cs-137	0.00E+00	N/A	U	1.67E-01	7.19E-02	1.10E+01	0.00E+00	

Sample ID	Radionuclide	Activity (pCi/g)	Uncertainty (1 sigma)	Qualifier	MDA (pCi/g)	Decision Level (pCi/g)	Interim Screening Level Fraction	Fraction of Screening Level	SOF
	Eu-152	0.00E+00	N/A	U	4.07E-01	1.88E-01	8.70E+00	0.00E+00	
	Eu-154	2.58E-02	N/A	U	5.23E-01	2.21E-01	8.00E+00	3.23E-03	
	Eu-155	1.84E-01	N/A	U	5.57E-01	2.63E-01	2.80E+02	6.57E-04	
	Am-241	0.00E+00	N/A	U	5.06E-01	2.37E-01	2.10E+00	0.00E+00	

Table 2-81 7000 Subsurface Soil Samples Summary Statistics

Radionuclide	Min (pCi/g)	Max (pCi/g)	Mean (pCi/g)	Median (pCi/g)	SD (pCi/g)
Co-60	0.00E+00	8.44E-02	2.67E-02	2.33E-02	2.35E-02
Cs-134	0.00E+00	1.12E-02	2.60E-04	0.00E+00	1.52E-03
Cs-137	0.00E+00	2.55E-01	3.62E-02	2.58E-02	4.43E-02
Eu-152	0.00E+00	1.73E-01	2.39E-02	0.00E+00	3.82E-02
Eu-154	0.00E+00	2.00E-01	2.98E-02	0.00E+00	5.22E-02
Eu-155	0.00E+00	4.43E-01	1.89E-01	1.75E-01	1.19E-01
Am-241	0.00E+00	1.45E-01	1.24E-02	0.00E+00	3.11E-02

Total Number of Samples	
Random	0
Judgmental	82

Judgmental	
SOF >0.5	0
Maximum SOF	0.1142
Minimum SOF	0.0011

Table 2-82 GEL Results for Open Land Area Soil Samples (pCi/g)

Sample ID	Am-241	Sb-125	C-14	Ce-144	Cs-134	Cs-137	Co-57	Co-58	Co-60	Cm-242	Cm-243/244	Eu-152	Eu-154	Eu-155	Fe-55
7000X-1-CJ-GSB1-002	8.58E-03	0.00E+00	2.04E-01	0.00E+00	1.60E-02	**1.05E-01**	9.84E-03	1.93E-02	3.95E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.86E-02	7.30E-01
7000X-1-CJ-GSB1-012	2.28E-01	0.00E+00	0.00E+00	0.00E+00	5.50E-02	3.88E-03	1.49E-02	1.40E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.22E-01	0.00E+00	0.00E+00
7000X-1-CJ-GSB1-014	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.90E-02	7.37E-03	0.00E+00	0.00E+00	3.40E-02	0.00E+00	3.58E-03	0.00E+00	3.18E-02	4.08E-02	0.00E+00
7000X-1-CJ-GSB3-008	4.55E-03	2.48E-01	0.00E+00	0.00E+00	5.93E-02	**6.93E-02**	0.00E+00	6.06E-03	3.10E-02	0.00E+00	8.80E-03	0.00E+00	8.20E-03	9.45E-02	0.00E+00
7000X-1-CJ-GSB3-018	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.45E-02	0.00E+00	8.10E-03	0.00E+00	1.12E-02	0.00E+00	2.81E-03	0.00E+00	0.00E+00	9.92E-05	0.00E+00
7000X-1-CJ-GSB3-020	3.13E-03	0.00E+00	0.00E+00	5.20E-04	3.89E-02	3.83E-02	6.48E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.15E-01	0.00E+00
7000X-1-CJ-GSB3-021	1.07E-02	0.00E+00	0.00E+00	0.00E+00	5.33E-02	0.00E+00	1.40E-04	0.00E+00	0.00E+00	1.42E-02	2.80E-04	4.58E-02	3.80E-02	2.05E-02	1.25E+00
7000X-1-CJ-GSB3-023	3.54E-03	4.19E-03	0.00E+00	4.76E-02	0.00E+00	9.15E-03	1.32E-02	0.00E+00	0.00E+00	2.21E-03	0.00E+00	3.88E-02	3.61E-02	5.70E-02	1.03E+00
7100X-1-CJ-GSBX-002	0.00E+00	1.18E-02	0.00E+00	0.00E+00	2.37E-02	**1.73E-01**	5.10E-04	0.00E+00	4.24E-02	0.00E+00	0.00E+00	1.06E-02	0.00E+00	1.09E-02	0.00E+00
7200X-1-CJ-GSBX-009	7.89E-03	0.00E+00	0.00E+00	3.32E-02	2.48E-02	1.15E-02	5.46E-03	0.00E+00	8.80E-04	0.00E+00	0.00E+00	8.98E-03	0.00E+00	1.64E-02	0.00E+00
7300X-1-CJ-GSSX-001	0.00E+00	2.40E-03	0.00E+00	0.00E+00	9.97E-03	6.12E-03	1.07E-03	1.69E-02	8.66E-03	0.00E+00	1.39E-02	9.75E-03	0.00E+00	8.84E-02	0.00E+00
7400X-1-CJ-GSSX-004	0.00E+00	0.00E+00	0.00E+00	4.08E-04	1.59E-02	**4.21E-02**	0.00E+00	5.87E-03	4.93E-03	1.53E-02	9.44E-03	1.62E-02	0.00E+00	2.32E-02	0.00E+00
8100X-3-CJ-GSSX-044	7.07E-02	4.66E-02	0.00E+00	1.02E-01	8.37E-02	**1.72E-01**	4.55E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.02E-03	0.00E+00	7.90E-02	2.18E+00
8100X-3-CJ-GSSX-058	2.72E-02	2.70E-02	1.37E-01	0.00E+00	1.87E-02	**1.80E-01**	0.00E+00	1.75E-02	0.00E+00	0.00E+00	3.58E-02	1.11E-03	5.66E-03	1.90E-02	2.45E+00
8100X-3-CJ-GSSX-064	4.72E-02	6.77E-02	1.07E-01	0.00E+00	5.11E-02	**1.24E-01**	6.78E-03	0.00E+00	0.00E+00	2.57E-02	9.67E-02	8.78E-02	1.29E-01	1.79E-02	4.43E+00
8100X-3-CR-GSSX-003	0.00E+00	0.00E+00	0.00E+00	7.80E-02	4.36E-02	**1.29E-01**	0.00E+00	4.06E-02	5.21E-03	0.00E+00	8.71E-03	0.00E+00	0.00E+00	6.16E-02	3.18E+00
8100X-3-CR-GSSX-021	1.20E-02	1.16E-01	0.00E+00	2.28E-01	0.00E+00	**2.05E-01**	2.37E-03	6.82E-02	2.29E-02	0.00E+00	2.23E-02	0.00E+00	3.91E-02	4.19E-02	1.88E+00
8100X-3-CR-GSSX-027	1.26E-02	2.24E-02	4.79E-01	7.07E-02	4.96E-02	**3.08E-01**	0.00E+00	0.00E+00	2.98E-02	0.00E+00	2.50E-02	0.00E+00	7.78E-02	5.79E-02	0.00E+00
8100X-3-CR-GSSX-029	0.00E+00	1.74E-03	0.00E+00	0.00E+00	9.19E-02	**4.96E-01**	5.95E-02	6.93E-02	0.00E+00	0.00E+00	1.20E-02	2.82E-01	7.35E-03	6.79E-02	2.30E+00
8100X-3-CR-GSSX-038	0.00E+00	1.22E-01	2.73E-01	2.41E-01	0.00E+00	**1.41E-01**	8.79E-03	4.28E-02	0.00E+00	0.00E+00	7.52E-03	2.67E-02	0.00E+00	0.00E+00	4.32E+00
8200X-3-CJ-GSSX-029	2.31E-02	0.00E+00	6.90E-01	1.52E-01	0.00E+00	**2.38E-01**	0.00E+00	0.00E+00	2.63E-02	0.00E+00	1.72E-02	3.34E-02	9.98E-03	5.20E-02	7.30E-01
8200X-3-CJ-GSSX-033	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.65E-02	**2.23E-01**	0.00E+00	9.29E-03	1.19E-02	6.40E-03	1.07E-02	0.00E+00	5.30E-04	0.00E+00	3.43E+00
8200X-3-CJ-GSSX-035	0.00E+00	5.18E-02	2.73E-01	0.00E+00	4.19E-03	**2.31E-01**	1.12E-02	3.30E-02	1.62E-02	6.79E-03	0.00E+00	0.00E+00	0.00E+00	1.48E-01	4.19E+00
8200X-3-CJ-GSSX-037	8.52E-04	0.00E+00	0.00E+00	2.46E-02	3.43E-02	**9.29E-02**	0.00E+00	0.00E+00	3.63E-02	1.32E-02	1.90E-02	1.83E-02	5.13E-02	4.93E-02	4.86E+00
8200X-3-CR-GSSX-023	2.05E-03	0.00E+00	0.00E+00	0.00E+00	5.91E-02	**1.71E-01**	1.25E-02	0.00E+00	0.00E+00	7.64E-03	9.30E-03	0.00E+00	4.61E-02	4.57E-02	5.14E+00
8200X-3-CR-GSSX-026	1.12E-02	4.25E-02	4.23E-01	6.57E-02	9.33E-02	**2.28E-01**	3.40E-03	0.00E+00	1.45E-02	0.00E+00	1.99E-02	2.83E-02	0.00E+00	9.41E-02	0.00E+00
8300X-3-CR-GSSX-002	4.44E-02	7.87E-02	0.00E+00	0.00E+00	5.26E-02	**1.28E-01**	1.45E-02	0.00E+00	0.00E+00	0.00E+00	1.83E-02	0.00E+00	0.00E+00	6.19E-02	0.00E+00
8300X-3-CR-GSSX-016	5.89E-02	5.53E-02	0.00E+00	6.45E-02	0.00E+00	**1.06E-01**	0.00E+00	0.00E+00	1.66E-02	3.27E-02	0.00E+00	0.00E+00	0.00E+00	1.27E-01	5.12E+00
8300X-3-CR-GSSX-021	0.00E+00	4.35E-02	6.73E-01	2.54E-02	6.99E-02	**1.72E-01**	1.95E-02	3.47E-02	1.30E-03	1.98E-02	2.66E-02	1.20E-04	8.65E-03	0.00E+00	1.49E+00
8300X-3-CR-GSSX-030	8.14E-04	8.38E-02	0.00E+00	1.40E-01	8.15E-02	1.13E-01	8.64E-04	2.62E-01	1.69E-02	6.13E-02	6.76E-02	9.47E-03	6.35E-02	6.07E-02	1.81E+00

Sample ID	Am-241	Sb-125	C-14	Ce-144	Cs-134	Cs-137	Co-57	Co-58	Co-60	Cm-242	Cm-243/244	Eu-152	Eu-154	Eu-155	Fe-55
8300X-3-CR-GSSX-031	1.13E-02	2.64E-02	0.00E+00	0.00E+00	0.00E+00	**3.13E-01**	5.00E-03	0.00E+00	1.68E-02	0.00E+00	4.60E-02	0.00E+00	6.24E-02	1.37E-01	2.82E+00
8300X-3-CR-GSSX-035	0.00E+00	0.00E+00	0.00E+00	1.92E-01	0.00E+00	7.27E-02	1.62E-02	0.00E+00	2.27E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.05E-02	3.54E+00
8300X-3-CR-GSSX-037	0.00E+00	1.01E-01	0.00E+00	0.00E+00	3.29E-02	1.56E-01	0.00E+00	1.12E-01	2.37E-03	0.00E+00	2.05E-02	3.77E-02	0.00E+00	0.00E+00	3.40E+00
8300X-3-CR-GSSX-040	3.95E-02	4.25E-02	0.00E+00	0.00E+00	1.08E-02	**1.47E-01**	0.00E+00	0.00E+00	2.94E-02	0.00E+00	0.00E+00	7.50E-02	0.00E+00	9.33E-02	0.00E+00
8300X-3-CR-GSSX-042	1.93E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	**9.68E-02**	1.09E-02	1.32E-03	5.61E-03	0.00E+00	0.00E+00	5.53E-02	0.00E+00	0.00E+00	5.35E-02
8400X-2-CJ-GSSX-027	0.00E+00	0.00E+00	0.00E+00	5.74E-02	6.16E-02	**6.66E-02**	0.00E+00	0.00E+00	4.96E-02	0.00E+00	2.18E-03	0.00E+00	3.41E-02	0.00E+00	6.28E+00
8400X-2-CR-GSSX-004	0.00E+00	2.28E-02	0.00E+00	1.65E-02	4.76E-02	**3.57E-01**	0.00E+00	1.14E-03	1.71E-02	0.00E+00	3.25E-03	3.01E-02	1.35E-02	4.17E-02	0.00E+00
8400X-2-CR-GSSX-006	0.00E+00	2.99E-03	0.00E+00	1.64E-01	4.45E-02	4.61E-02	0.00E+00	9.08E-02	0.00E+00	0.00E+00	0.00E+00	4.78E-02	0.00E+00	9.27E-02	6.79E+00
8700X-2-CJ-GSSX-021	3.00E-03	4.84E-03	0.00E+00	1.29E-01	0.00E+00	**4.63E-02**	0.00E+00	1.29E-02	4.33E-03	6.60E-03	0.00E+00	0.00E+00	1.74E-03	8.18E-02	1.54E+01
8700X-2-CR-GSSX-005	1.40E-02	4.23E-02	0.00E+00	9.60E-02	5.16E-02	3.08E-02	1.22E-02	0.00E+00	1.57E-03	0.00E+00	0.00E+00	4.14E-02	1.02E-01	0.00E+00	6.81E+00

Sample ID	Mn-54	Np-237	Ni-59	Ni-63	Nb-94	Pu-238	Pu-239/240	Pu-241	Ag-110m	Sr-90	Tc-99	H-3	Zn-65
7000X-1-CJ-GSB1-002	0.00E+00	0.00E+00	0.00E+00	4.85E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.99E-03	6.32E-02	0.00E+00	6.70E-01	0.00E+00
7000X-1-CJ-GSB1-012	1.80E-02	0.00E+00	5.95E-01	3.23E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.44E-01	2.76E-01	1.04E+00	1.16E-02
7000X-1-CJ-GSB1-014	2.35E-01	0.00E+00	2.74E-01	5.96E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.28E-03	3.84E-01	8.72E-01	0.00E+00
7000X-1-CJ-GSB3-008	2.19E-02	0.00E+00	0.00E+00	2.08E-01	0.00E+00	0.00E+00	1.28E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.45E+00	0.00E+00
7000X-1-CJ-GSB3-018	2.58E-03	0.00E+00	1.39E+00	6.77E-01	0.00E+00	1.35E-02	0.00E+00	0.00E+00	1.71E-02	0.00E+00	0.00E+00	6.64E-01	0.00E+00
7000X-1-CJ-GSB3-020	2.55E-02	0.00E+00	0.00E+00	0.00E+00	1.39E-02	0.00E+00	6.02E-03	0.00E+00	7.47E-03	0.00E+00	6.47E-02	1.73E+00	1.65E-02
7000X-1-CJ-GSB3-021	1.23E-02	2.90E-04	6.04E-01	0.00E+00	1.32E-02	5.70E-03	1.99E-03	0.00E+00	0.00E+00	0.00E+00	3.95E-01	1.75E+00	3.45E-02
7000X-1-CJ-GSB3-023	5.53E-02	1.08E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.97E-01	3.00E-01	0.00E+00
7100X-1-CJ-GSBX-002	0.00E+00	3.20E-04	0.00E+00	9.01E-01	0.00E+00	4.70E-03	3.71E-03	0.00E+00	5.37E-03	9.07E-03	1.28E-01	0.00E+00	1.74E-02
7200X-1-CJ-GSBX-009	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.29E-03	1.84E-02	5.83E-03	0.00E+00	0.00E+00	5.33E-02	0.00E+00	0.00E+00	0.00E+00
7300X-1-CJ-GSSX-001	1.71E-02	1.11E-03	0.00E+00	0.00E+00	3.84E-03	1.03E-02	0.00E+00	0.00E+00	2.90E-03	4.98E-02	2.85E-01	3.08E-01	3.57E-02
7400X-1-CJ-GSSX-004	9.06E-03	2.59E-03	4.63E-01	0.00E+00	1.12E-02	1.03E-02	0.00E+00	0.00E+00	0.00E+00	2.92E-03	0.00E+00	0.00E+00	2.26E-03
8100X-3-CJ-GSSX-044	7.75E-03	0.00E+00	0.00E+00	0.00E+00	5.79E-02	0.00E+00	7.86E-03	1.26E+00	2.08E-02	1.33E-01	0.00E+00	0.00E+00	7.12E-02
8100X-3-CJ-GSSX-058	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.37E-03	2.85E-03	1.35E-02	1.36E+00	0.00E+00	1.09E-01	0.00E+00	0.00E+00	2.89E-03
8100X-3-CJ-GSSX-064	1.94E-02	8.29E-04	0.00E+00	5.00E-01	3.50E-02	1.50E-02	1.73E-02	2.17E+00	7.33E-02	1.33E-01	0.00E+00	0.00E+00	8.45E-03
8100X-3-CR-GSSX-003	3.03E-02	1.12E-03	1.39E-01	4.13E-01	8.78E-03	4.90E-03	0.00E+00	0.00E+00	0.00E+00	1.44E-01	0.00E+00	0.00E+00	4.71E-03
8100X-3-CR-GSSX-021	3.95E-03	1.39E-03	7.57E-01	3.50E-01	1.54E-02	0.00E+00	2.29E-02	0.00E+00	0.00E+00	5.79E-02	0.00E+00	0.00E+00	0.00E+00
8100X-3-CR-GSSX-027	4.56E-02	1.89E-03	7.73E-01	0.00E+00	0.00E+00	5.67E-03	1.16E-02	8.17E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8100X-3-CR-GSSX-029	0.00E+00	0.00E+00	0.00E+00	4.64E-01	1.79E-01	1.90E-02	9.91E-03	1.76E+00	0.00E+00	3.00E-02	0.00E+00	0.00E+00	0.00E+00

Sample ID	Mn-54	Np-237	Ni-59	Ni-63	Nb-94	Pu-238	Pu-239/240	Pu-241	Ag-110m	Sr-90	Tc-99	H-3	Zn-65
8100X-3-CR-GSSX-038	4.32E-03	0.00E+00	1.17E+00	4.66E-01	0.00E+00	4.82E-03	1.06E-02	1.70E+00	6.39E-02	0.00E+00	6.42E-03	0.00E+00	0.00E+00
8200X-3-CJ-GSSX-029	0.00E+00	6.96E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.22E-02	0.00E+00	4.84E-03	7.88E-02	0.00E+00	0.00E+00	4.35E-02
8200X-3-CJ-GSSX-033	1.21E-02	0.00E+00	0.00E+00	5.41E-01	1.49E-02	5.78E-03	0.00E+00	2.81E-01	2.81E-02	2.48E-02	0.00E+00	0.00E+00	1.02E-01
8200X-3-CJ-GSSX-035	1.82E-03	0.00E+00	9.72E-02	0.00E+00	0.00E+00	2.31E-03	1.48E-02	6.42E-01	5.00E-02	3.61E-03	0.00E+00	0.00E+00	2.69E-02
8200X-3-CJ-GSSX-037	0.00E+00	0.00E+00	9.04E-02	0.00E+00	1.33E-02	0.00E+00	0.00E+00	1.80E+00	0.00E+00	1.44E-01	0.00E+00	0.00E+00	0.00E+00
8200X-3-CR-GSSX-023	1.41E-03	0.00E+00	5.80E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.87E+00	4.43E-02	2.17E-01	0.00E+00	0.00E+00	1.22E-01
8200X-3-CR-GSSX-026	6.24E-02	0.00E+00	0.00E+00	0.00E+00	3.08E-02	0.00E+00	2.75E-02	1.55E+00	1.98E-02	1.06E-03	0.00E+00	0.00E+00	0.00E+00
8300X-3-CR-GSSX-002	0.00E+00	0.00E+00	3.69E-01	0.00E+00	0.00E+00	3.62E-02	0.00E+00	3.29E+00	0.00E+00	2.05E-02	1.15E-01	1.63E+00	1.06E-02
8300X-3-CR-GSSX-016	1.62E-03	0.00E+00	0.00E+00	0.00E+00	2.63E-03	3.59E-02	0.00E+00	1.51E+00	3.83E-02	0.00E+00	0.00E+00	3.06E-01	1.79E-02
8300X-3-CR-GSSX-021	0.00E+00	0.00E+00	5.23E-01	0.00E+00	5.33E-03	0.00E+00	0.00E+00	2.15E+00	0.00E+00	1.10E-01	0.00E+00	1.10E+00	7.99E-02
8300X-3-CR-GSSX-030	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.16E-02	0.00E+00	1.85E-02	1.16E+00	3.92E-02	0.00E+00	0.00E+00	2.52E-01	0.00E+00
8300X-3-CR-GSSX-031	1.78E-02	0.00E+00	0.00E+00	0.00E+00	2.02E-02	0.00E+00	0.00E+00	2.03E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.96E-02
8300X-3-CR-GSSX-035	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.28E-02	1.52E-02	0.00E+00	1.76E+00	0.00E+00	8.64E-02	0.00E+00	0.00E+00	6.23E-02
8300X-3-CR-GSSX-037	4.70E-02	1.03E-03	0.00E+00	0.00E+00	2.20E-02	7.38E-03	1.50E-02	0.00E+00	3.60E-02	0.00E+00	0.00E+00	1.31E+00	0.00E+00
8300X-3-CR-GSSX-040	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.36E-03	0.00E+00	2.02E-02	3.74E-01	6.70E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8300X-3-CR-GSSX-042	1.06E-02	0.00E+00	4.95E-05	5.65E-01	9.06E-03	0.00E+00	0.00E+00	0.00E+00	1.24E-02	2.58E-04	0.00E+00	0.00E+00	1.09E-01
8400X-2-CJ-GSSX-027	9.21E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.13E-03	0.00E+00	0.00E+00	0.00E+00	5.31E-02
8400X-2-CR-GSSX-004	1.39E-02	4.04E-04	1.62E+00	0.00E+00	8.42E-03	0.00E+00	9.38E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.31E-01	3.11E-02
8400X-2-CR-GSSX-006	8.04E-03	0.00E+00	0.00E+00	0.00E+00	2.31E-02	2.50E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.29E-03
8700X-2-CJ-GSSX-021	2.16E-02	0.00E+00	0.00E+00	7.63E-01	0.00E+00	0.00E+00	3.33E-03	0.00E+00	0.00E+00	9.87E-03	0.00E+00	0.00E+00	6.86E-02
8700X-2-CR-GSSX-005	0.00E+00	0.00E+00	1.11E+00	4.72E-01	0.00E+00	0.00E+00	1.60E-03	0.00E+00	5.46E-03	0.00E+00	5.42E-01	0.00E+00	0.00E+00

(Notes) Bold values indicate positively identified radionuclide concentrations. Negative values in the data were recorded as zero.

Attachment 2-3 Results for Structure Surface Measurements

Figure 2-38 Turbine Building 1036 Foot Elevation Surface Measurement Locations

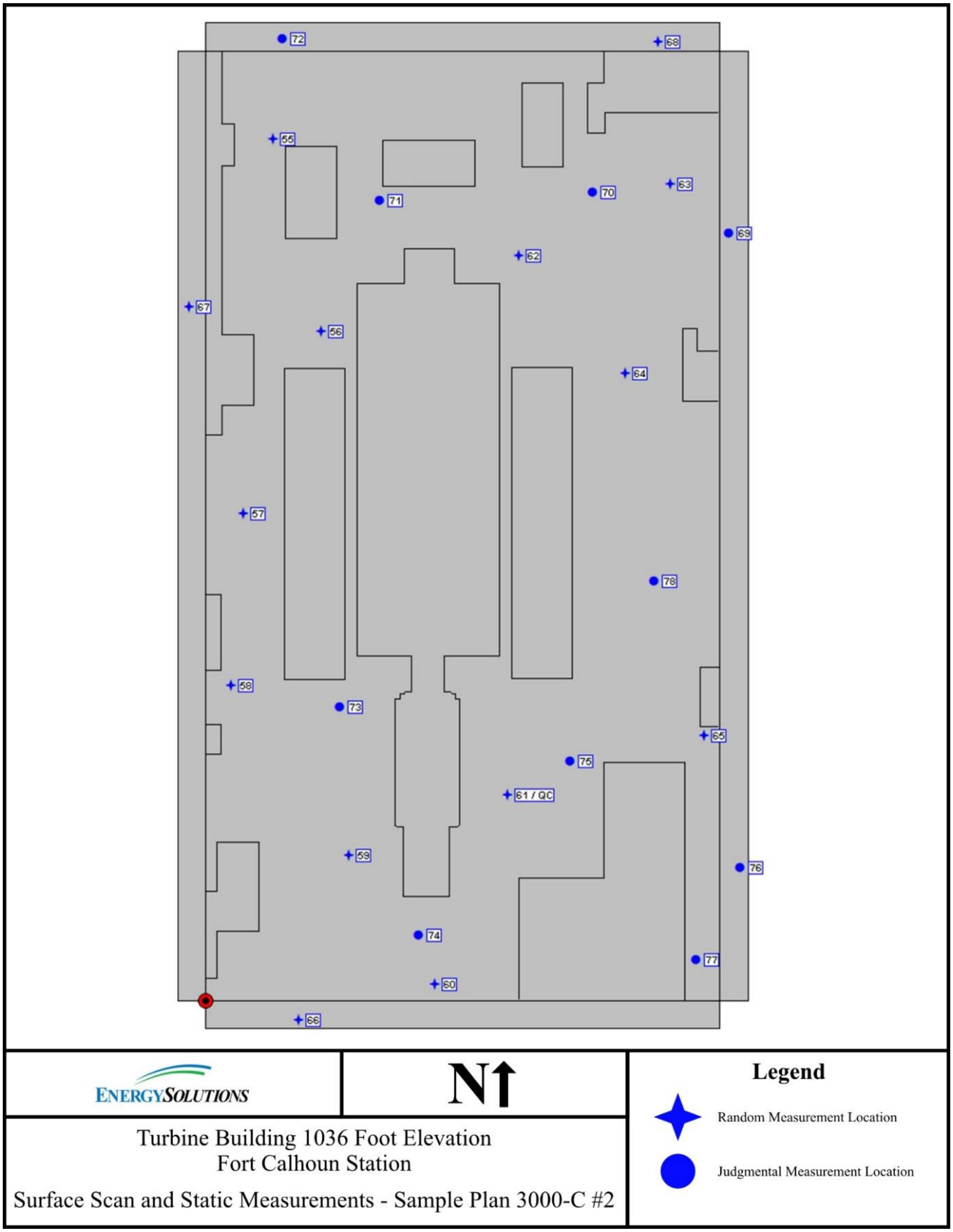
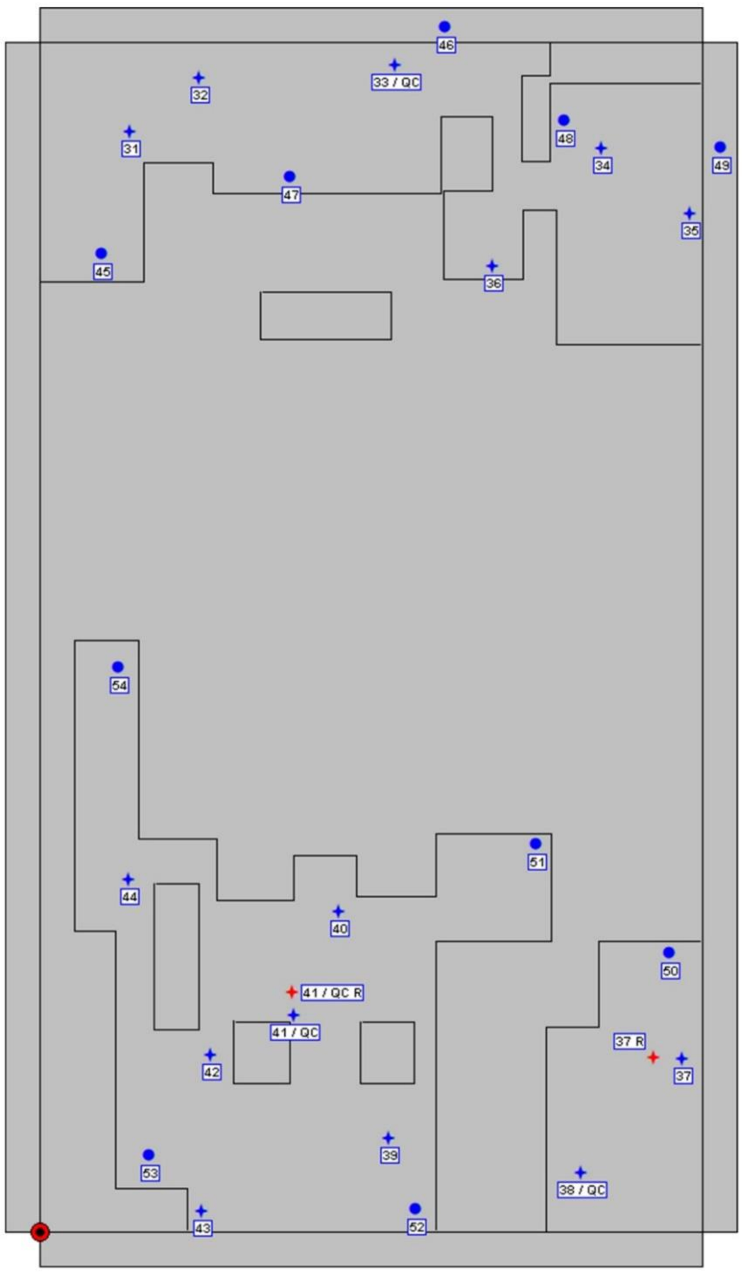


Figure 2-39 Turbine Building 1036 Foot Elevation Surface Measurement Locations



ENERGYSOLUTIONS	N↑	**Legend**
Turbine Building 1011 Foot Elevation Fort Calhoun Station Surface Scan and Static Measurements - Sample Plan 3000-C #2		Random Measurement Location Judgmental Measurement Location "R" = Relocated Measurement

Figure 2-40 Turbine Building 990 Foot Elevation Surface Measurement Locations

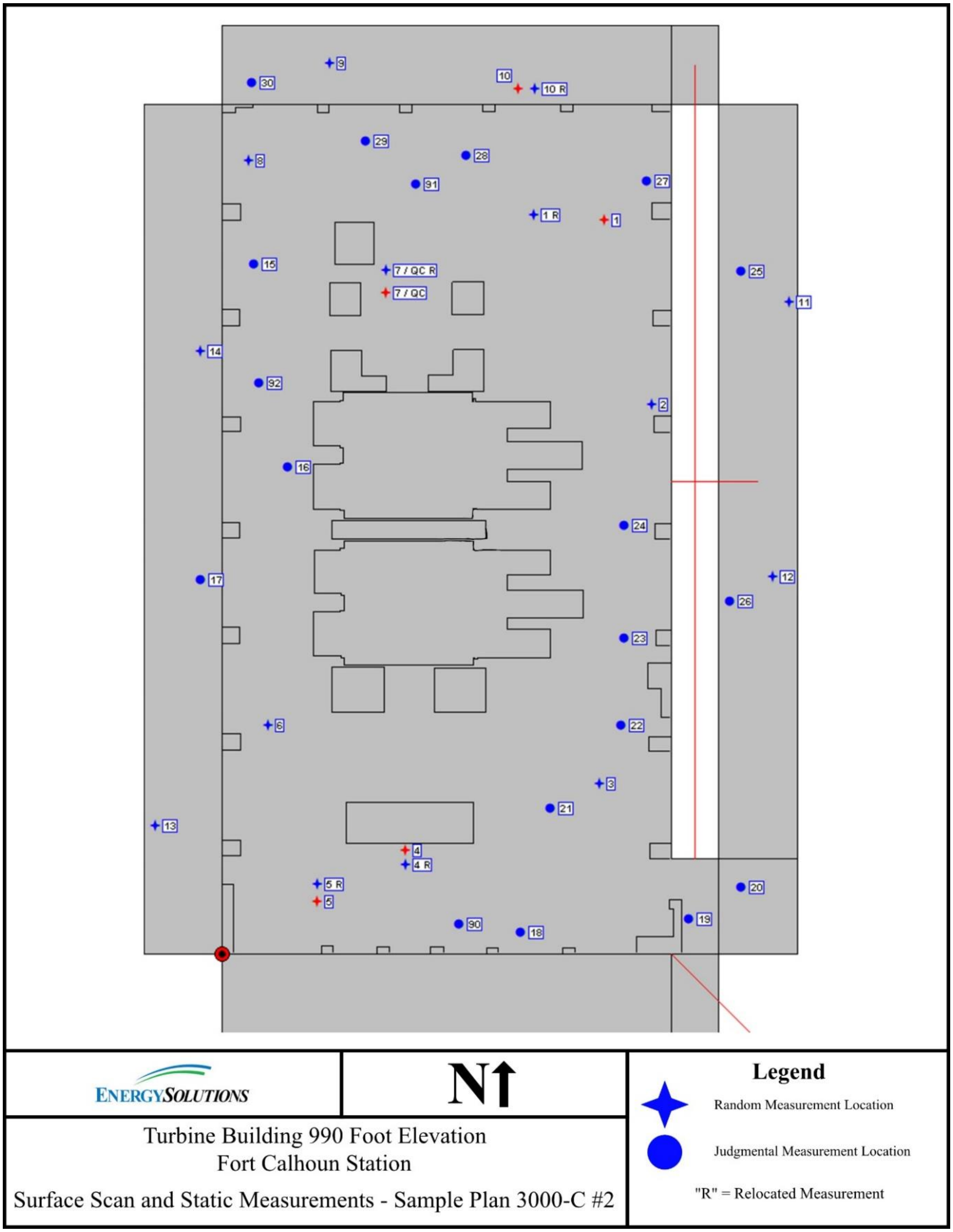


Figure 2-41 Turbine Building Roof Surface Measurement Locations

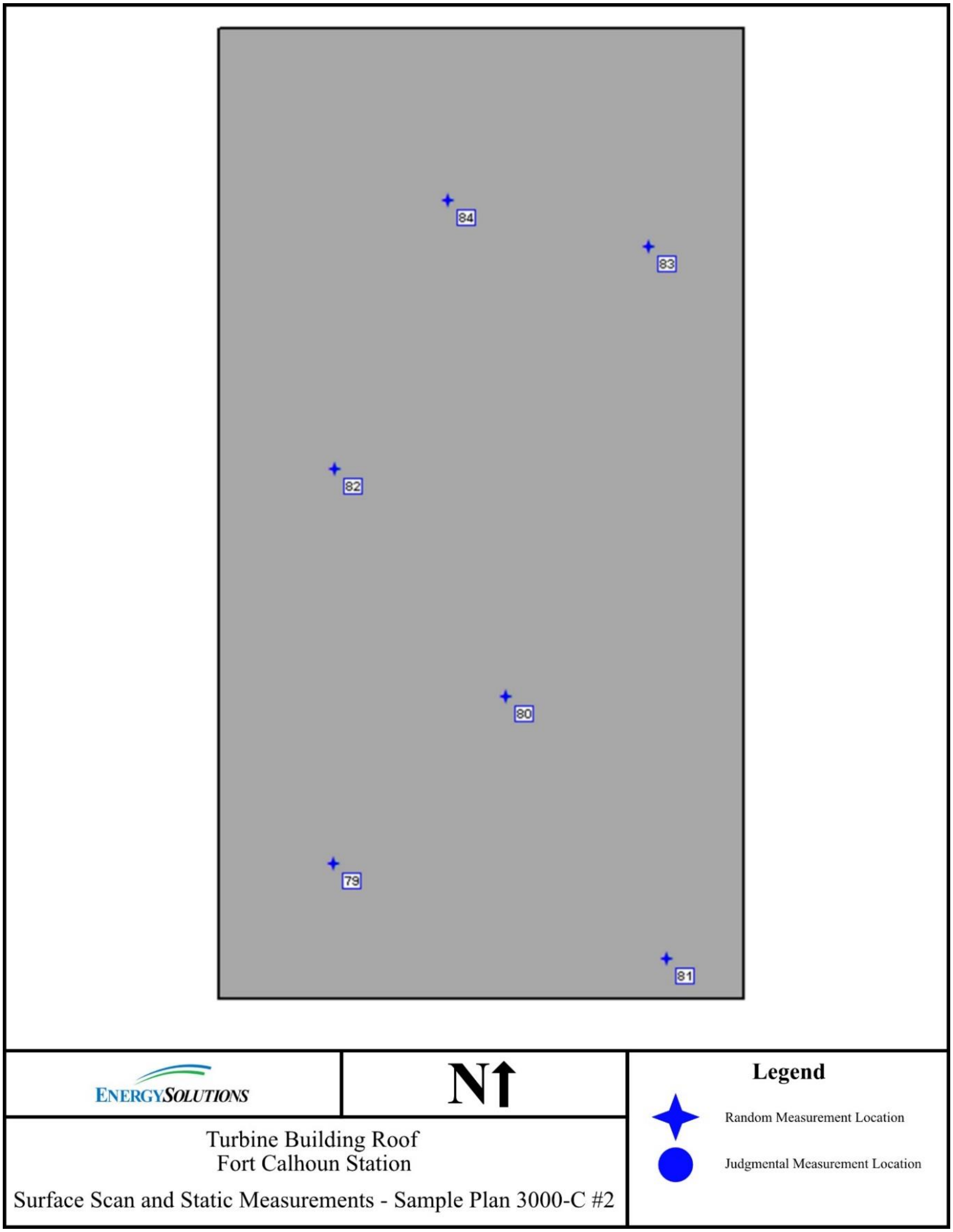
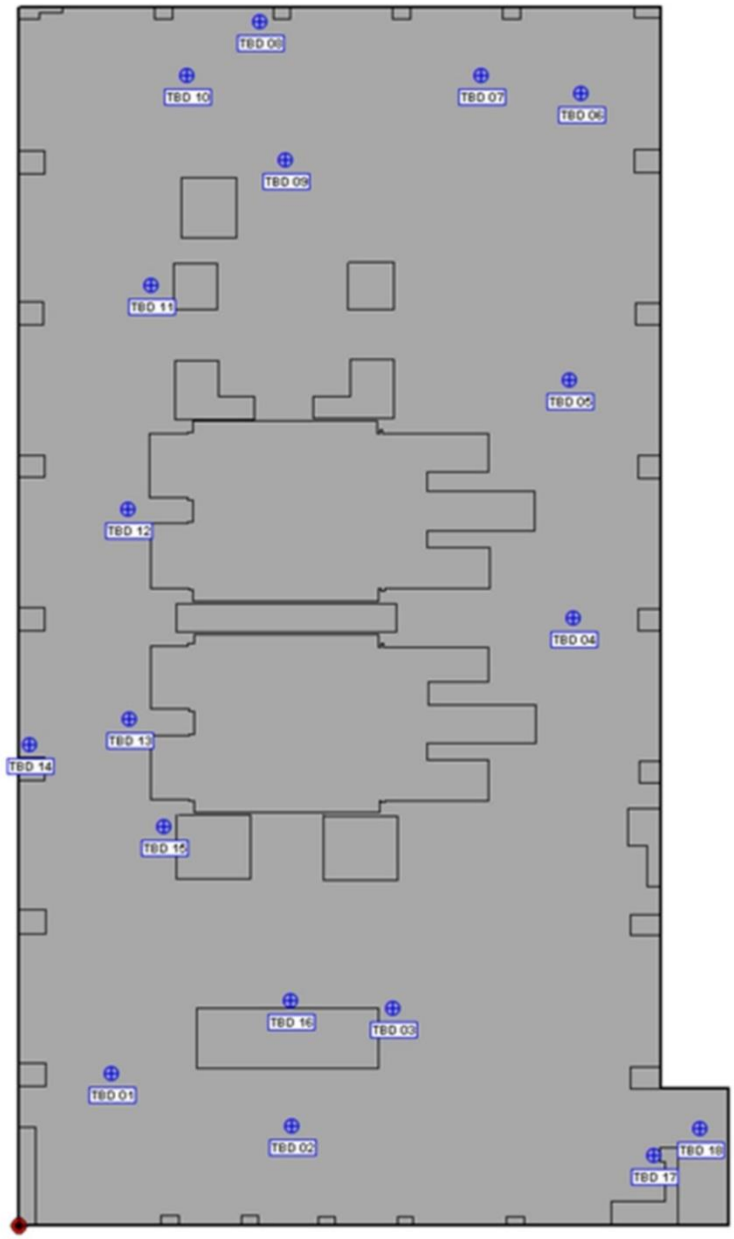


Figure 2-42 Turbine Building 990 Foot Elevation Floor Drain Measurement Locations



ENERGYSOLUTIONS

N↑

Turbine Building 990 Foot Elevation Floor Drains
Fort Calhoun Station

Gamma Static Measurements - Sample Plan 3000-C #2

Legend

⊕ Floor Drain Measurement Location

Table 2-83 Turbine Building Scan Results

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
1	492	274	524	0
2	457	294	552	0
3	428	294	552	0
4	396	220	444	0
5	390	220	444	0
6	360	220	444	0
7	359	220	444	0
8	473	274	524	0
9	377	278	530	0
10	396	296	556	0
11	417	296	556	0
12	422	296	556	0
13	445	249	487	0
14	352	249	487	0
15	396	220	444	0
16	464	220	444	1
17	438	249	487	0
18	537	294	552	0
19	508	294	552	0
20	437	249	487	0
21	485	294	552	0
22	454	294	552	0
23	552	294	552	1
24	530	294	552	0
25	435	296	556	0
26	428	249	487	0
27	433	274	524	0
28	523	274	524	0
29	476	274	524	0
30	528	278	530	0
31	469	283	537	0
32	436	283	537	0
33	512	283	537	0
34	506	283	537	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
35	536	283	537	0
36	504	283	537	0
37	404	283	537	0
38	497	283	537	0
39	470	283	537	0
40	456	259	502	0
41	537	288	543	0
42	526	288	543	0
43	396	259	502	0
44	495	288	543	0
45	513	288	543	0
46	244	134	309	0
47	376	259	502	0
48	523	283	537	0
49	499	271	519	0
50	529	283	537	0
51	470	259	502	0
52	478	288	543	0
53	439	288	543	0
54	521	288	543	0
55	668	485	818	0
56	777	485	818	0
57	795	485	818	0
58	792	485	818	0
59	922	485	818	1
60	732	485	818	0
61	731	467	794	0
62	794	525	871	0
63	798	525	871	0
64	848	525	871	0
65	769	467	794	0
66	355	246	483	0
67	421	220	443	0
68	431	226	453	0
69	439	306	569	0
70	721	467	794	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
71	694	467	794	0
72	281	226	453	0
73	815	485	818	0
74	738	485	818	0
75	715	467	794	0
76	452	226	453	0
77	704	467	794	0
78	580	525	871	0
79	810	639	1020	0
80	839	639	1020	0
81	853	639	1020	0
82	690	574	935	0
83	840	639	1020	0
84	690	574	935	0
85	545	337	614	0
86	543	337	614	0
87	572	337	614	0
88 TRB2A	504	344	624	0
89 TRB2B	342	344	624	0
90 GEN1	309	344	624	0
91 GEN2	318	344	624	0
92 TRB1B	308	344	624	0
93 TRB1A	489	344	624	0
94 FW152	453	240	473	0
95 MSEX1	382	240	473	0
96 MSEX2	381	240	473	0
97 MSEX3	437	240	473	0
98 MSEX4	448	240	473	0
99 FW2A	277	197	408	0
100 FW2B	291	197	408	0
101 FW2C	378	197	408	0
102 FW730	372	197	408	0
103 FW731	482	197	408	1
104 FW732	351	197	408	0
105 FW733	379	197	408	0
106 FW1B	296	197	408	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
107 FW1A	354	197	408	0
108 FW4A1	319	197	408	0
109 FW4A2	357	197	408	0
110 HDT17	388	203	417	0
111 FW15A	321	203	417	0
112 MS183	381	203	417	0
QC 33	486	288	543	0
QC 38	494	288	543	0
QC 41	415	283	537	0
QC 61	759	525	871	0
QC 7	555	337	614	0
QC 98 MSEX4	365	214	434	0

Table 2-84 Turbine Building Static Measurement Results

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
3000X-3-CR-FBDX-001	0	0.0000
3000X-3-CR-FBDX-002	0	0.0000
3000X-3-CR-FBDX-003	120	0.0338
3000X-3-CR-FBDX-004	580	0.1634
3000X-3-CR-FBDX-005	174	0.0490
3000X-3-CR-FBDX-006	0	0.0000
3000X-3-CR-FBDX-007	338	0.0952
3000X-3-CR-FBDX-008	0	0.0000
3000X-3-CR-WBDX-009	0	0.0000
3000X-3-CR-WBDX-010	0	0.0000
3000X-3-CR-WBDX-011	0	0.0000
3000X-3-CR-WBDX-012	0	0.0000
3000X-3-CR-WBDX-013	49	0.0138
3000X-3-CR-WBDX-014	0	0.0000
3000X-3-CJ-FBDX-015	453	0.1276
3000X-3-CJ-FBDX-016	0	0.0000
3000X-3-CJ-WBDX-017	24	0.0068
3000X-3-CJ-FBDX-018	193	0.0544
3000X-3-CJ-FBDX-019	0	0.0000
3000X-3-CJ-WBDX-020	166	0.0468
3000X-3-CJ-FBDX-021	228	0.0642
3000X-3-CJ-FBDX-022	51	0.0144
3000X-3-CJ-FBDX-023	47	0.0132
3000X-3-CJ-FBDX-024	166	0.0468
3000X-3-CJ-WBDX-025	0	0.0000
3000X-3-CJ-WBDX-026	0	0.0000
3000X-3-CJ-FBDX-027	281	0.0792
3000X-3-CJ-FBDX-028	0	0.0000
3000X-3-CJ-FBDX-029	229	0.0645
3000X-3-CJ-WBDX-030	0	0.0000
3000X-3-CR-FBDX-031	67	0.0189
3000X-3-CR-FBDX-032	116	0.0327
3000X-3-CR-FBDX-033	116	0.0327
3000X-3-CR-FBDX-034	19	0.0054

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
3000X-3-CR-FBDX-035	181	0.0510
3000X-3-CR-FBDX-036	201	0.0566
3000X-3-CR-FBDX-037	59	0.0166
3000X-3-CR-FBDX-038	153	0.0431
3000X-3-CR-FBDX-039	0	0.0000
3000X-3-CR-FBDX-040	18	0.0051
3000X-3-CR-FBDX-041	0	0.0000
3000X-3-CR-FBDX-042	0	0.0000
3000X-3-CR-FBDX-043	18	0.0051
3000X-3-CR-FBDX-044	0	0.0000
3000X-3-CJ-FBDX-045	30	0.0085
3000X-3-CJ-WBDX-046	63	0.0177
3000X-3-CJ-FBDX-047	0	0.0000
3000X-3-CJ-FBDX-048	0	0.0000
3000X-3-CJ-WBDX-049	287	0.0808
3000X-3-CJ-FBDX-050	116	0.0327
3000X-3-CJ-FBDX-051	0	0.0000
3000X-3-CJ-FBDX-052	0	0.0000
3000X-3-CJ-FBDX-053	0	0.0000
3000X-3-CJ-FBDX-054	0	0.0000
3000X-3-CR-FBDX-055	0	0.0000
3000X-3-CR-FBDX-056	42	0.0118
3000X-3-CR-FBDX-057	0	0.0000
3000X-3-CR-FBDX-058	0	0.0000
3000X-3-CR-FBDX-059	0	0.0000
3000X-3-CR-FBDX-060	0	0.0000
3000X-3-CR-FBDX-061	0	0.0000
3000X-3-CR-FBDX-062	113	0.0318
3000X-3-CR-FBDX-063	267	0.0752
3000X-3-CR-FBDX-064	267	0.0752
3000X-3-CR-FBDX-065	22	0.0062
3000X-3-CR-WBDX-066	0	0.0000
3000X-3-CR-WBDX-067	97	0.0273
3000X-3-CR-WBDX-068	95	0.0268
3000X-3-CJ-WBDX-069	86	0.0242

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
3000X-3-CJ-FBDX-070	82	0.0231
3000X-3-CJ-FBDX-071	0	0.0000
3000X-3-CJ-WBDX-072	0	0.0000
3000X-3-CJ-FBDX-073	0	0.0000
3000X-3-CJ-FBDX-074	0	0.0000
3000X-3-CJ-FBDX-075	14	0.0039
3000X-3-CJ-WBDX-076	286	0.0806
3000X-3-CJ-FBDX-077	0	0.0000
3000X-3-CJ-FBDX-078	97	0.0273
3000X-3-CJ-RBDX-079	87	0.0245
3000X-3-CJ-RBDX-080	76	0.0214
3000X-3-CJ-RBDX-081	308	0.0868
3000X-3-CJ-RBDX-082	251	0.0707
3000X-3-CJ-RBDX-083	117	0.0330
3000X-3-CJ-RBDX-084	158	0.0445
3000X-3-CJ-FBDX-085	120	0.0338
3000X-3-CJ-FBDX-086	0	0.0000
3000X-3-CJ-FBDX-087	0	0.0000
3000X-3-CJ-SBDX-088	0	0.0000
3000X-3-CJ-SBDX-089	0	0.0000
3000X-3-CJ-SBDX-090	0	0.0000
3000X-3-CJ-SBDX-091	0	0.0000
3000X-3-CJ-SBDX-092	0	0.0000
3000X-3-CJ-SBDX-093	0	0.0000
3000X-3-CJ-SBDX-094	0	0.0000
3000X-3-CJ-SBDX-095	0	0.0000
3000X-3-CJ-SBDX-096	0	0.0000
3000X-3-CJ-SBDX-097	0	0.0000
3000X-3-CJ-SBDX-098	0	0.0000
3000X-3-CJ-SBDX-099	0	0.0000
3000X-3-CJ-SBDX-100	0	0.0000
3000X-3-CJ-SBDX-101	0	0.0000
3000X-3-CJ-SBDX-102	0	0.0000
3000X-3-CJ-SBDX-103	0	0.0000
3000X-3-CJ-SBDX-104	0	0.0000

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
3000X-3-CJ-SBDX-105	0	0.0000
3000X-3-CJ-SBDX-106	0	0.0000
3000X-3-CJ-SBDX-107	0	0.0000
3000X-3-CJ-SBDX-108	0	0.0000
3000X-3-CJ-SBDX-109	0	0.0000
3000X-3-CJ-SBDX-110	0	0.0000
3000X-3-CJ-SBDX-111	0	0.0000
3000X-3-CJ-SBDX-112	0	0.0000
3000X-3-CQ-FBDX-033	14	0.0039
3000X-3-CQ-FBDX-038	0	0.0000
3000X-3-CQ-FBDX-041	63	0.0177
3000X-3-CQ-FBDX-061	84	0.0237
3000X-3-CQ-FBDX-007	0	0.0000
3000X-3-CQ-SBDX-098	0	0.0000

(a) Action level is equivalent to 50% of the Screening Value for Co-60, or 3,550 dpm/100 cm^2.

Table 2-85 Turbine Building Surface Measuremements Summary Statistics

Random				
Mean (dpm/100cm²)	**Median (dpm/100cm²)**	**Max (dpm/100cm²)**	**Min (dpm/100cm²)**	**Std. Dev. (dpm/100cm²)**
74	0	580	0	105

Judgmental				
Mean (dpm/100cm²)	**Median (dpm/100cm²)**	**Max (dpm/100cm²)**	**Min (dpm/100cm²)**	**Std. Dev. (dpm/100cm²)**
57	0	453	0	100

Combined				
Mean (dpm/100cm²)	**Median (dpm/100cm²)**	**Max (dpm/100cm²)**	**Min (dpm/100cm²)**	**Std. Dev. (dpm/100cm²)**
64	0	580	0	107

Total Number of Measurements	
Random	42
Judgmental	70
QC	6

Random	
Fraction >1	0
Maximum Fraction:	0.1634

Judgmental	
Fraction >1	0
Maximum Fraction:	0.1276

Table 2-86 Turbine Building Drain Survey Results

Drain	Gamma Scan Reading (cpm)	Gamma Static Reading (cpm)	Background (cpm)
1	7673	6372	7444
2	6713	5194	7444
3	6814	5360	7444
4	7769	6329	7444
5	7757	6283	7444
6	8077	6442	7279
7	6068	5218	7279
8	7997	6674	7279
9	7791	5771	7279
10	5950	4919	7279
11	6268	5019	7279
12	7575	5019	7279
13	8432	7091	7279
14	7229	5961	7279
15	7470	5852	7279
16	8309	5831	7279
17	11148	9360	10486
18	10836	7796	10486

Figure 2-43 Intake Building 974’8” and 985’ Elevations Surface Measurement Locations

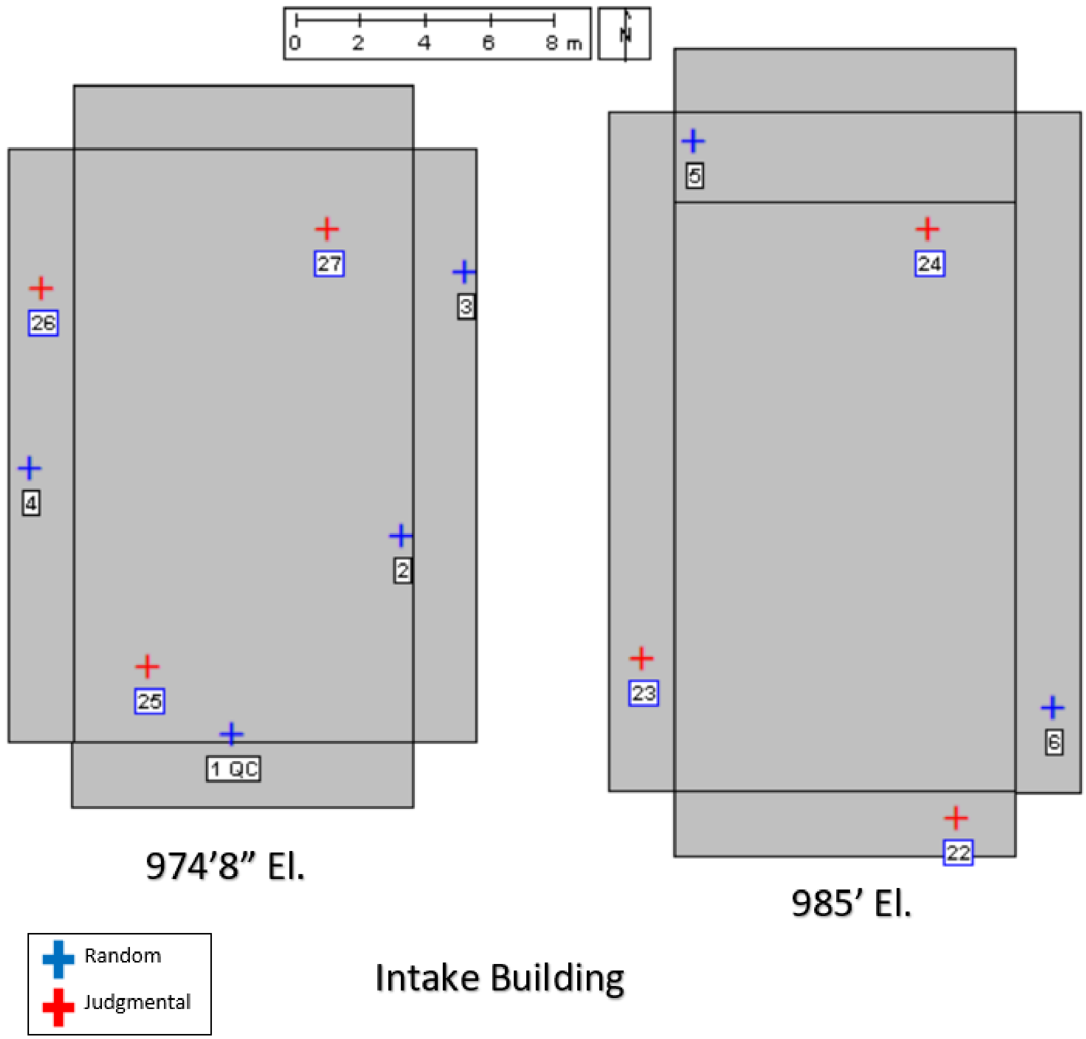


Figure 2-44 Intake Building 993'6" and 1007'6" Elevations Surface Measurement Locations

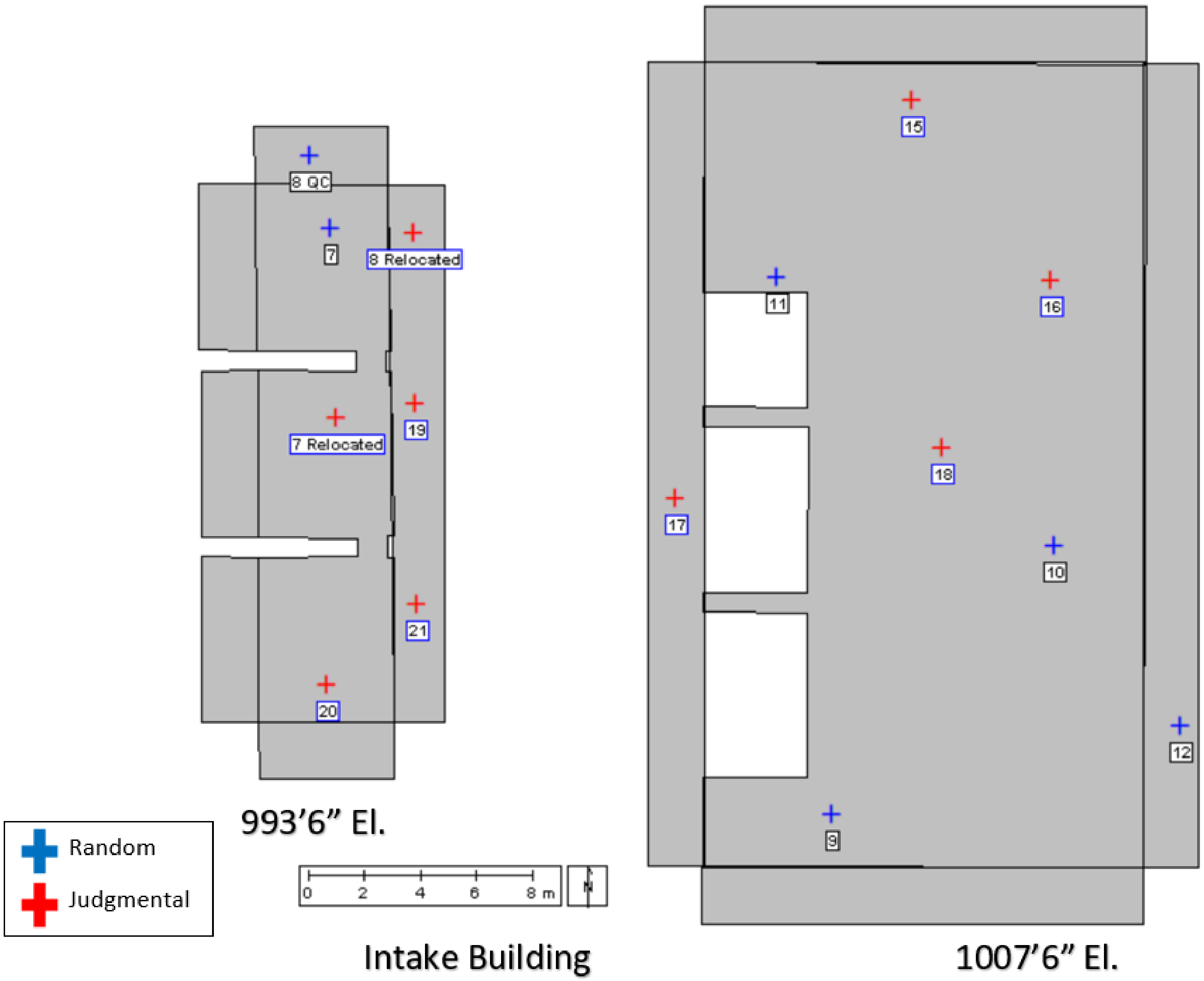


Figure 2-45 Intake Building Roof Surface Measurement Locations

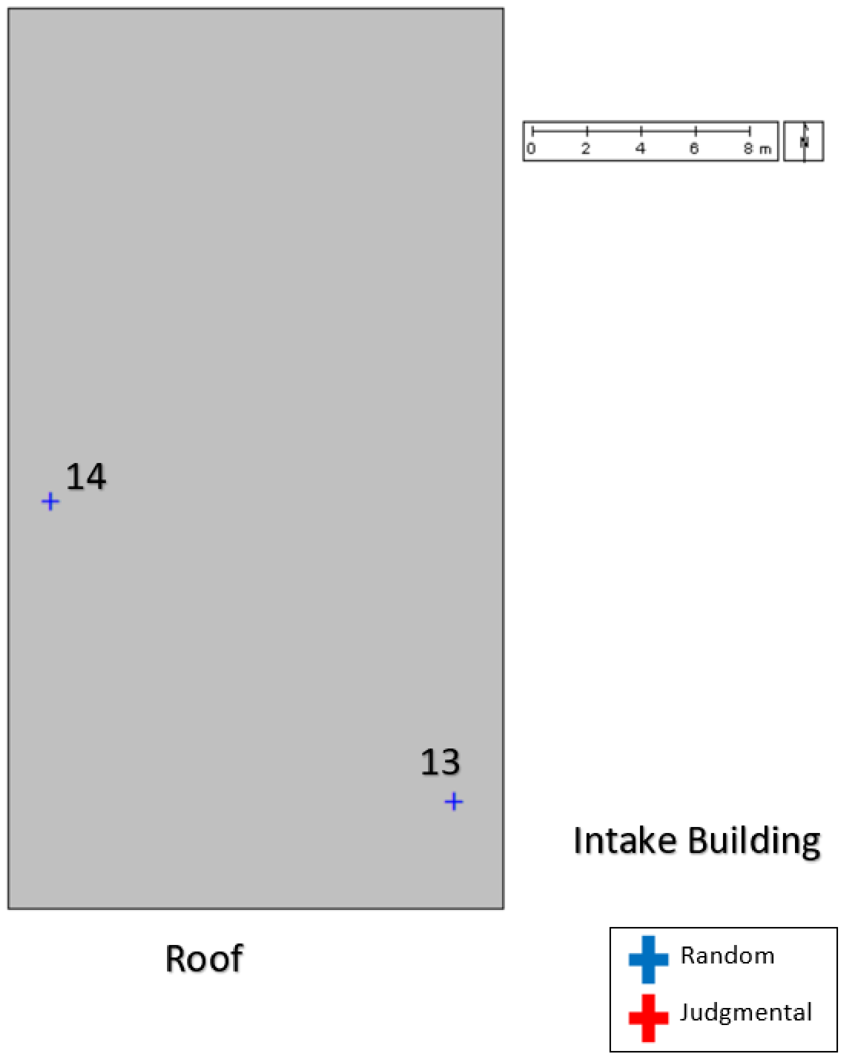


Figure 2-46 Security Building Surface Measurement Locations

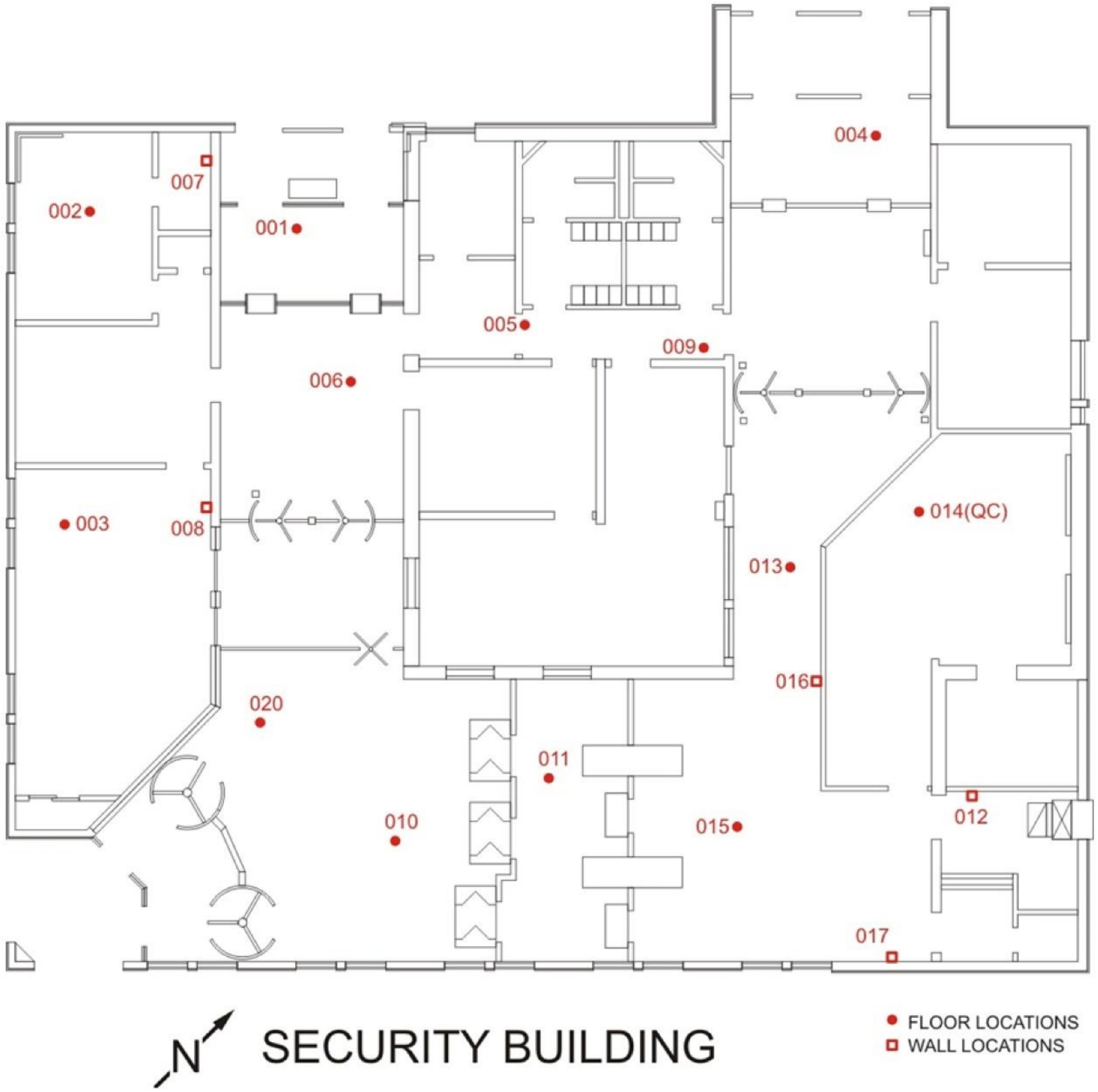


Figure 2-47 SAF Surface Measurement Locations

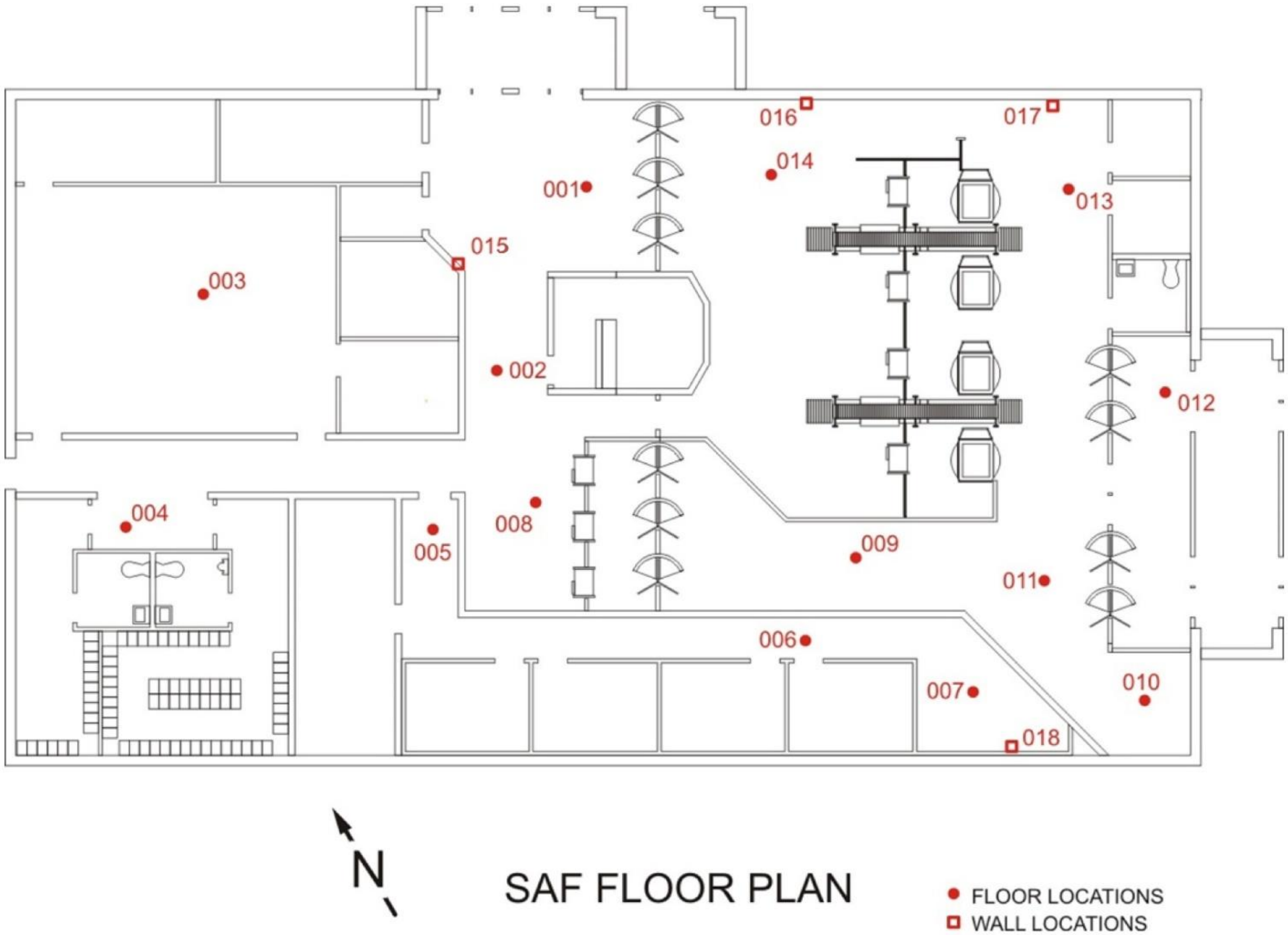


Figure 2-48 SAF Roof Surface Measurement Locations

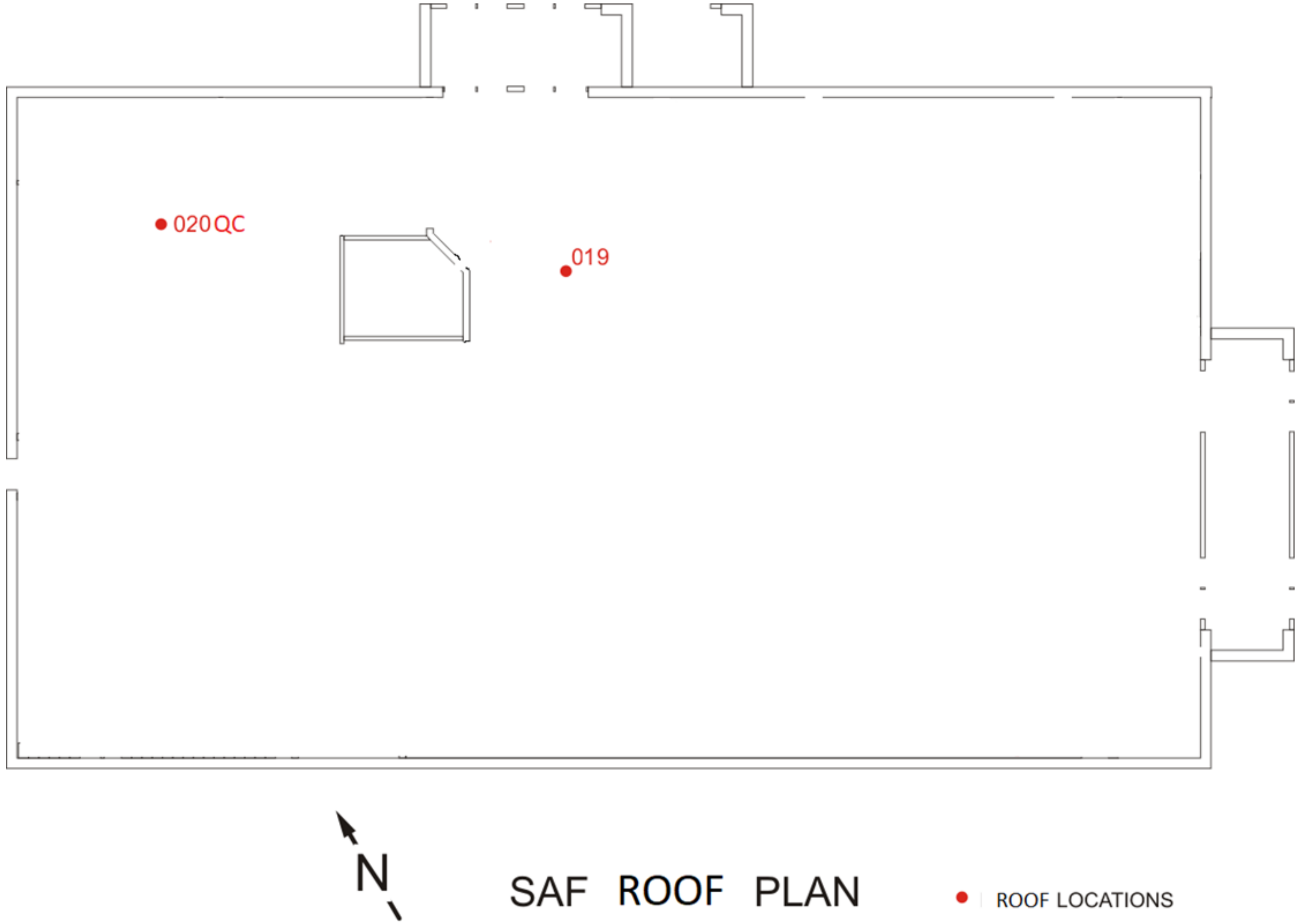


Figure 2-49 Service Building First Floor Surface Measurement Locations

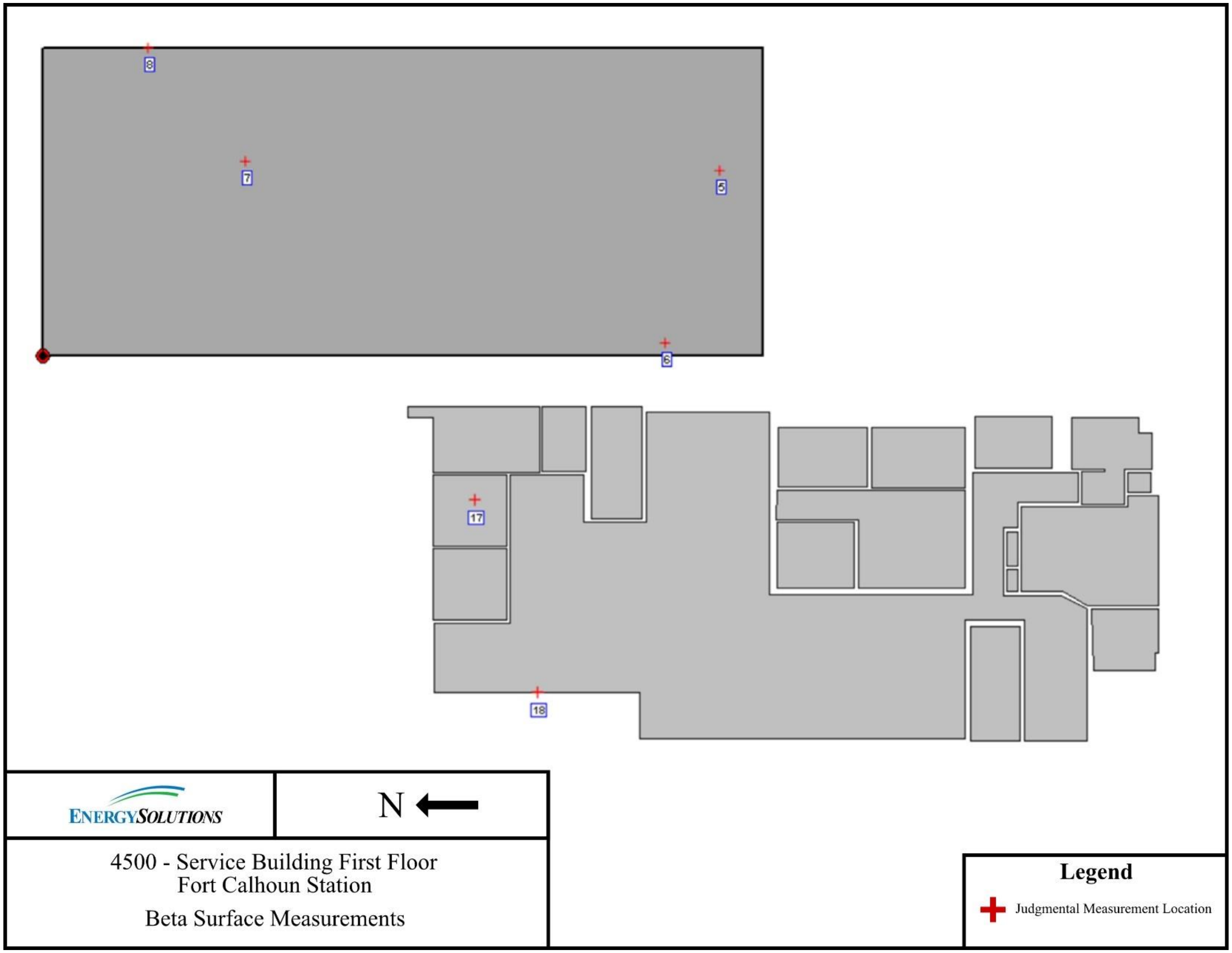


Figure 2-50 Service Building Third Floor Surface Measurement Locations

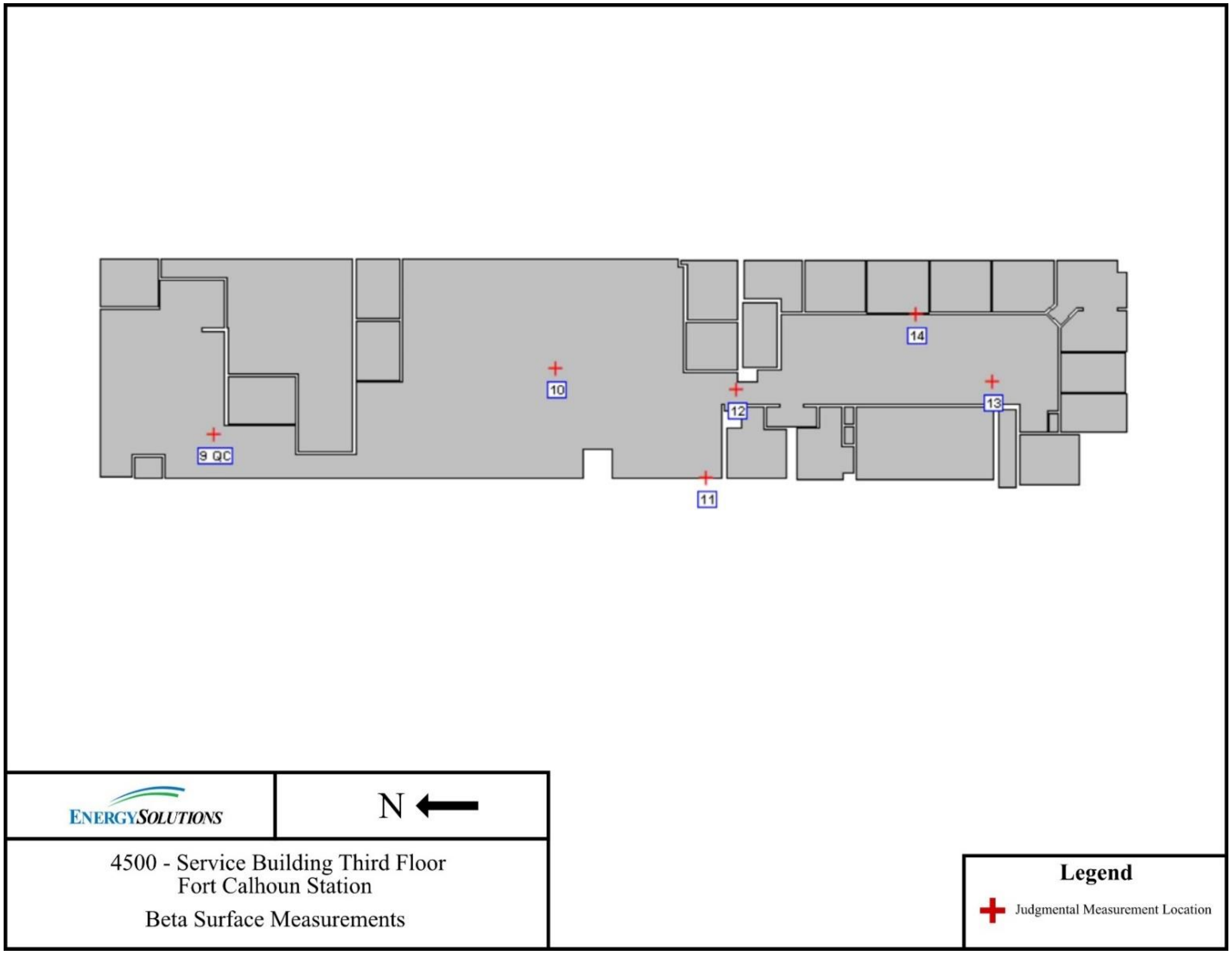


Figure 2-51 Service Building Mezzanine Surface Measurement Locations

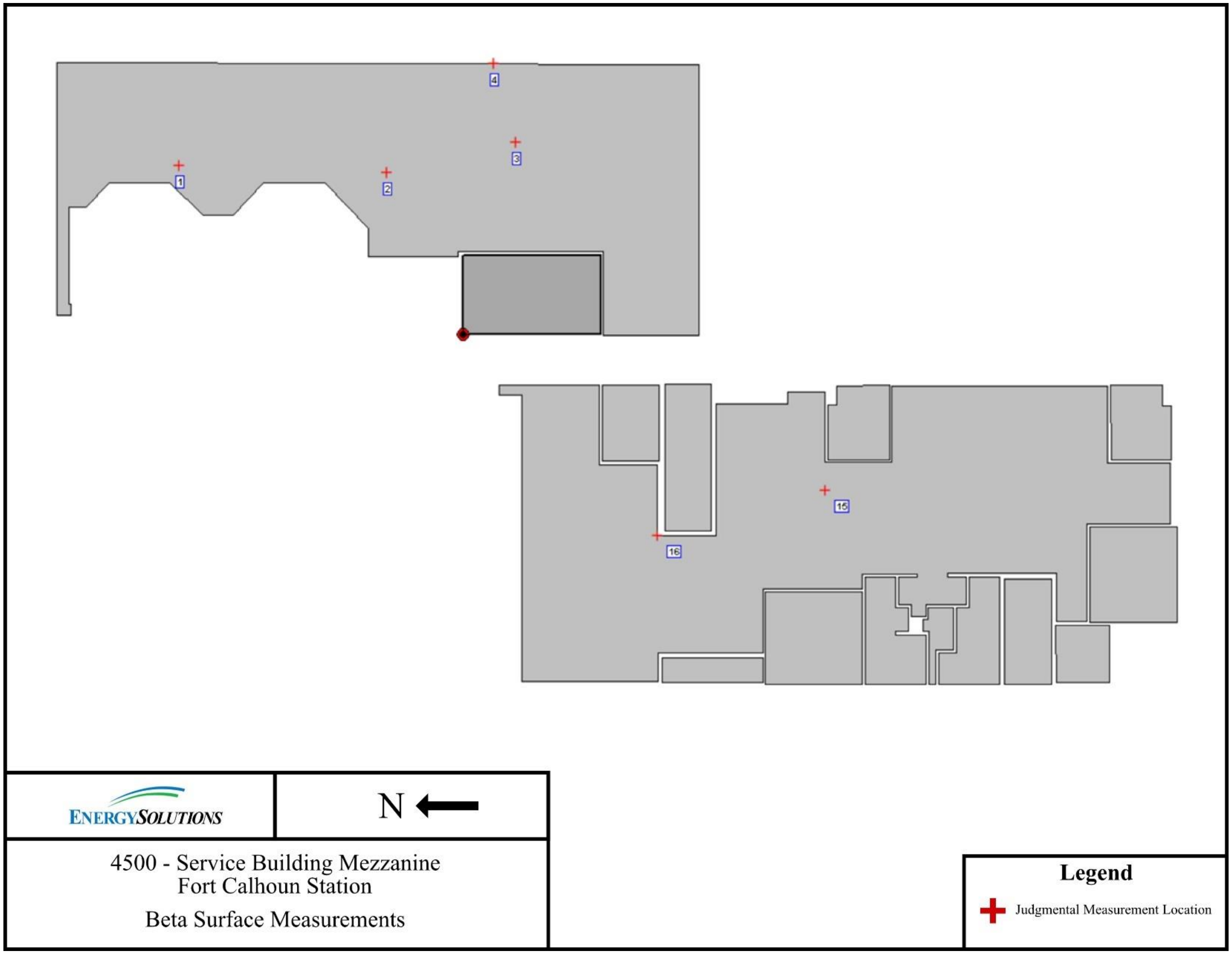


Figure 2-52 Service Building Roof Surface Measurement Locations

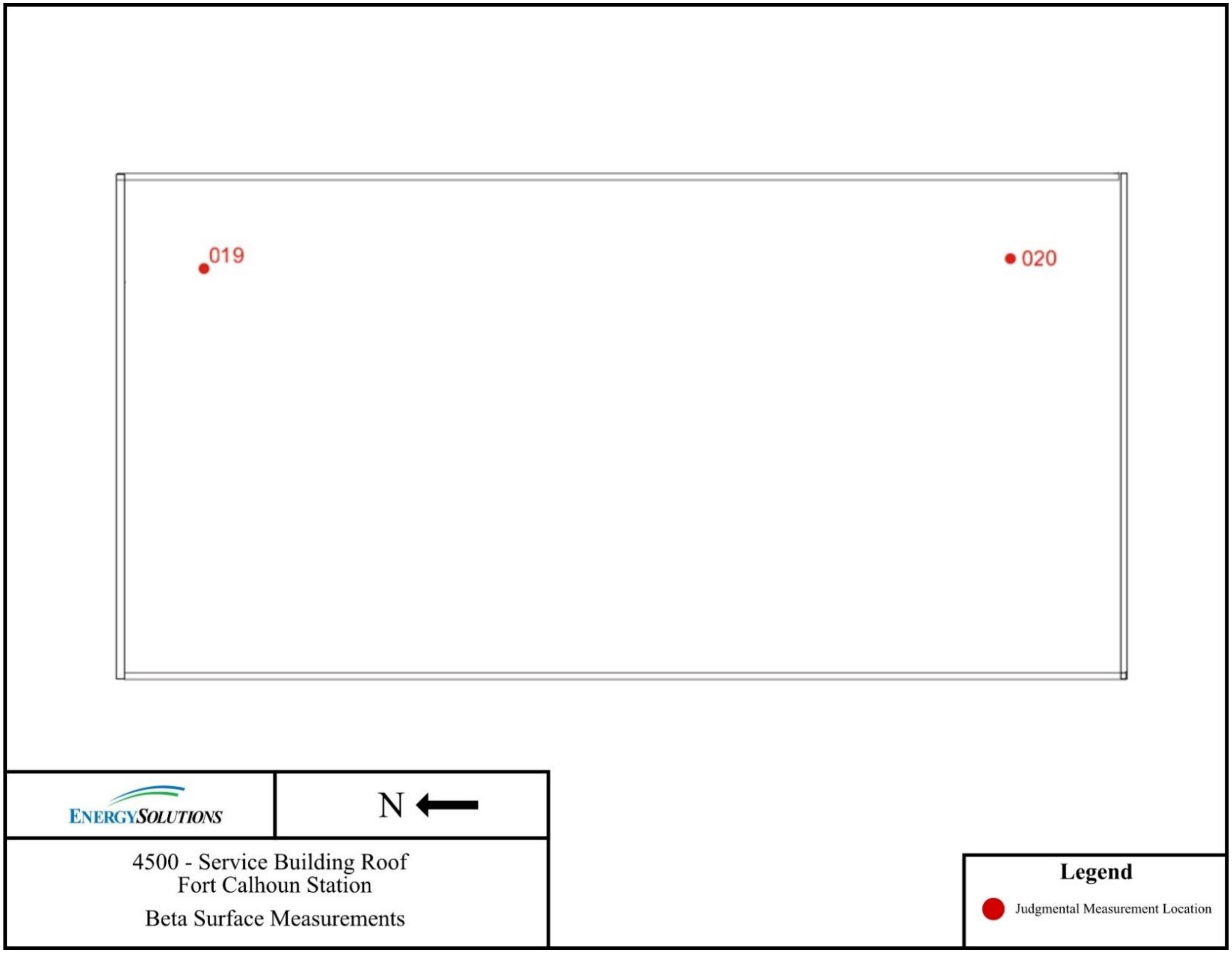


Figure 2-53 Maintenance Shop First Floor Surface Measurement Locations

Figure 2-54 Maintenance Shop Second Floor Surface Measurement Locations

Figure 2-55 Maintenance Shop Roof Surface Measurement Locations

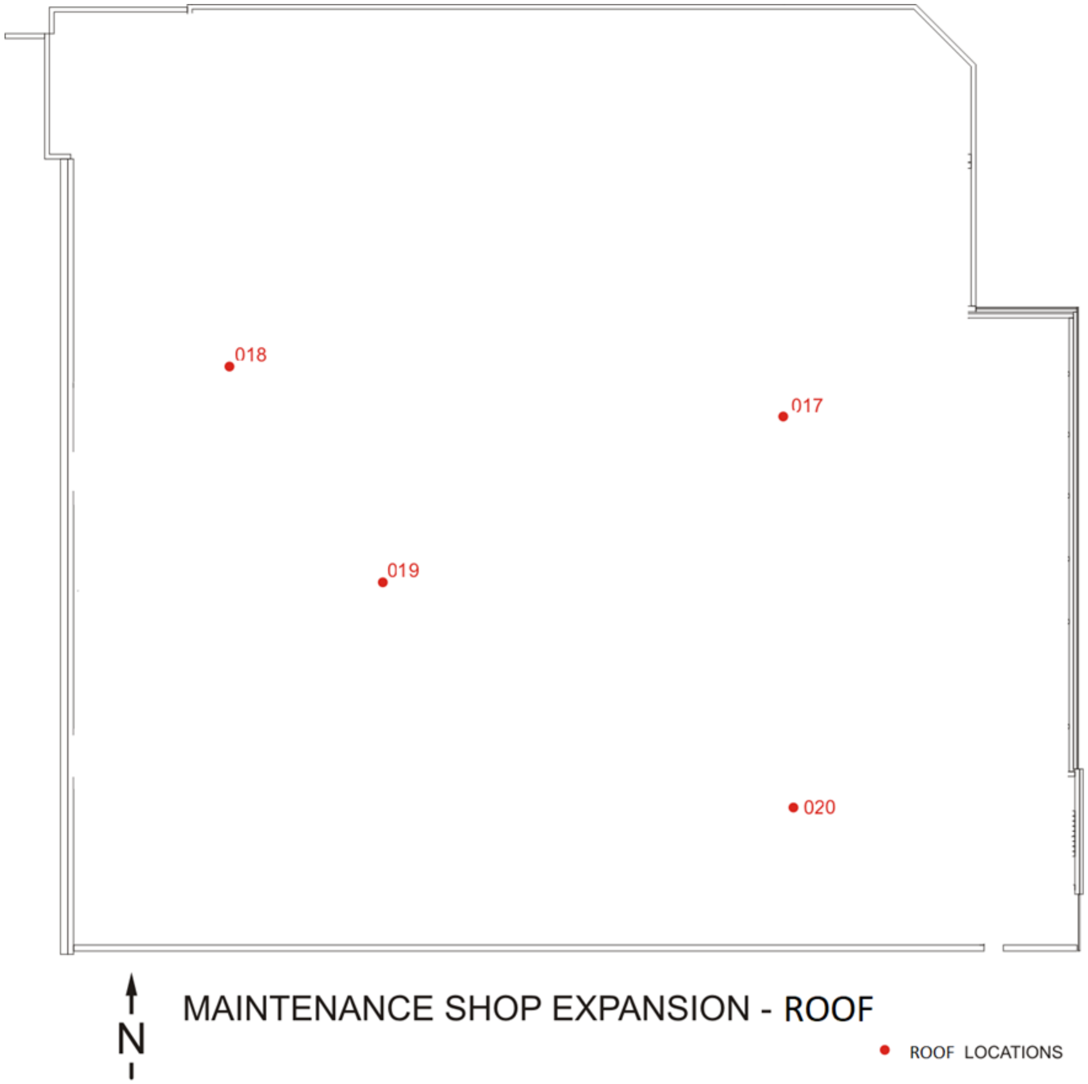


Figure 2-56 TSC Surface Measurement Locations

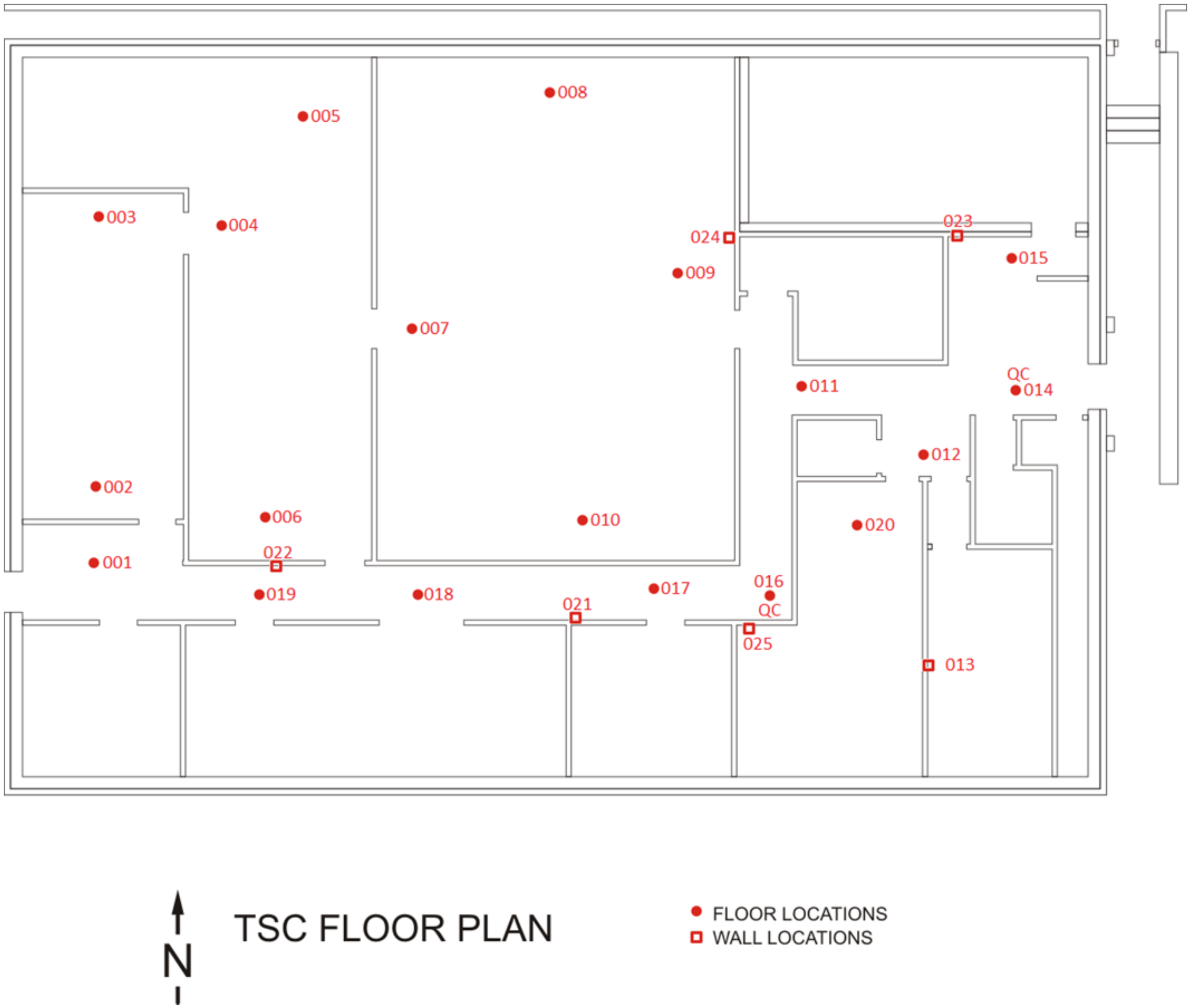


Figure 2-57 Chemistry and Radiation Protection Building Surface Measurement Locations

Figure 2-58 Chemistry and Radiation Protection Building Roof Surface Measurement Locations

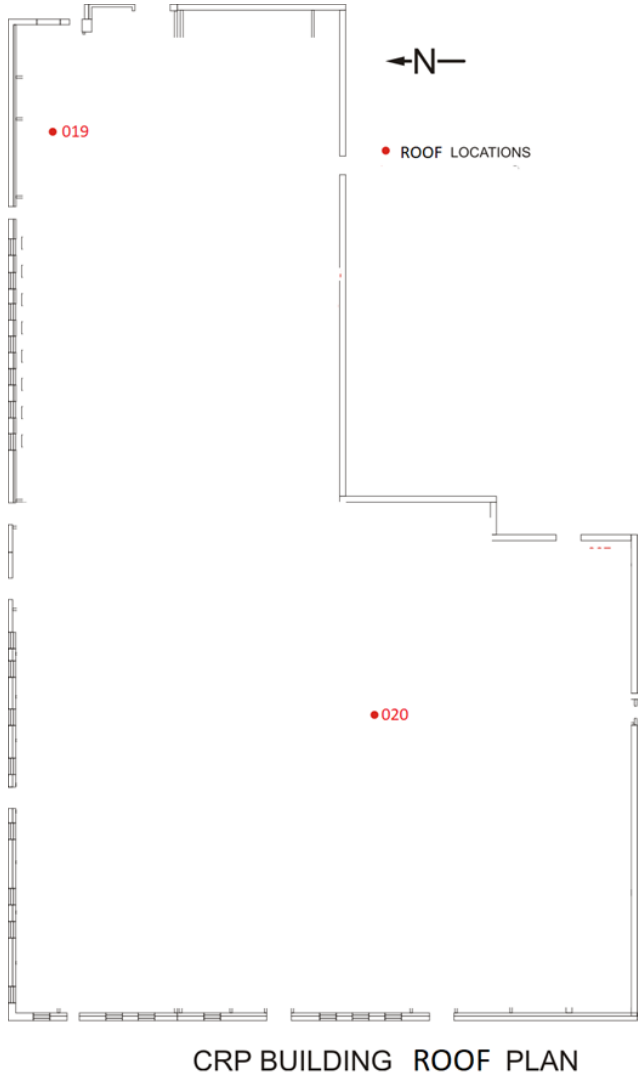


Figure 2-59 New Warehouse Surface Measurement Locations

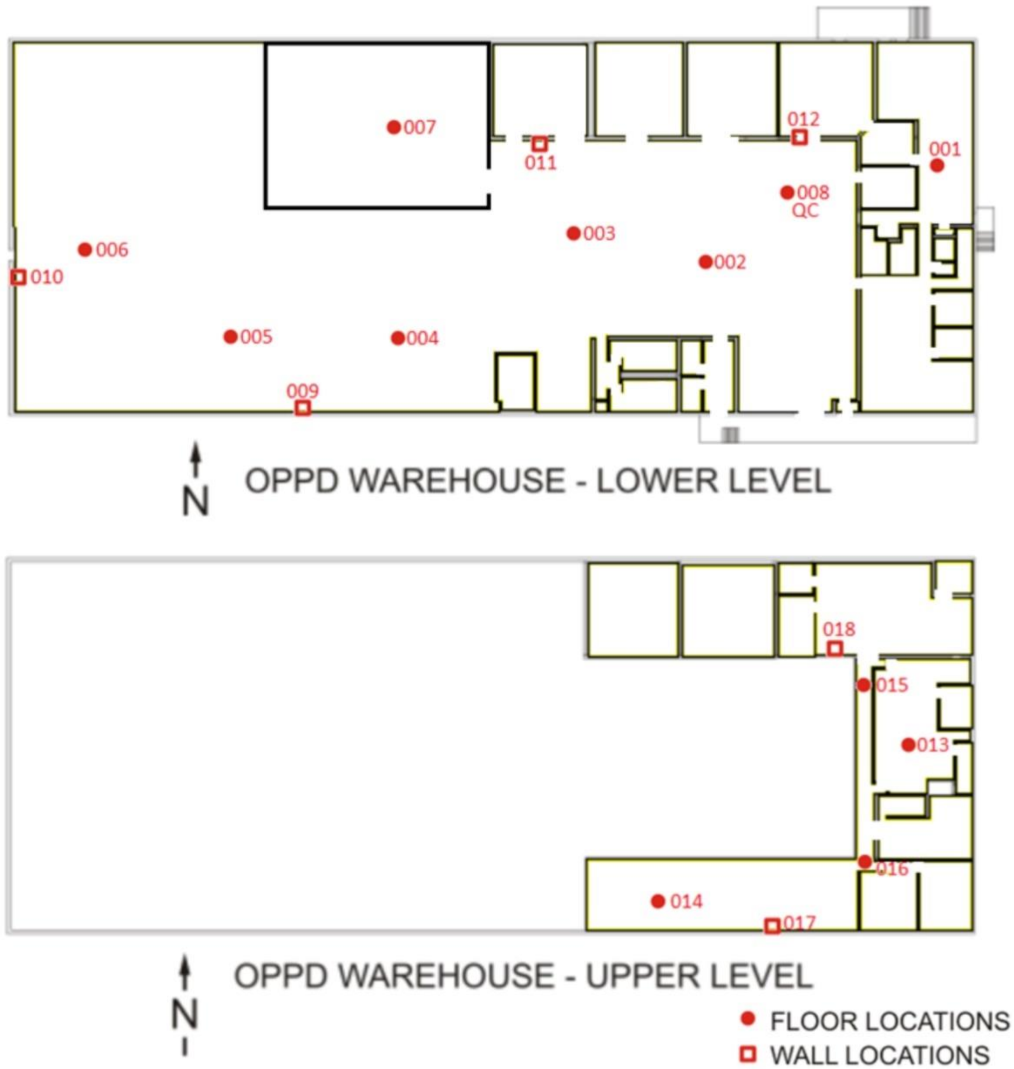


Figure 2-60 New Warehouse Roof Surface Measurement Locations

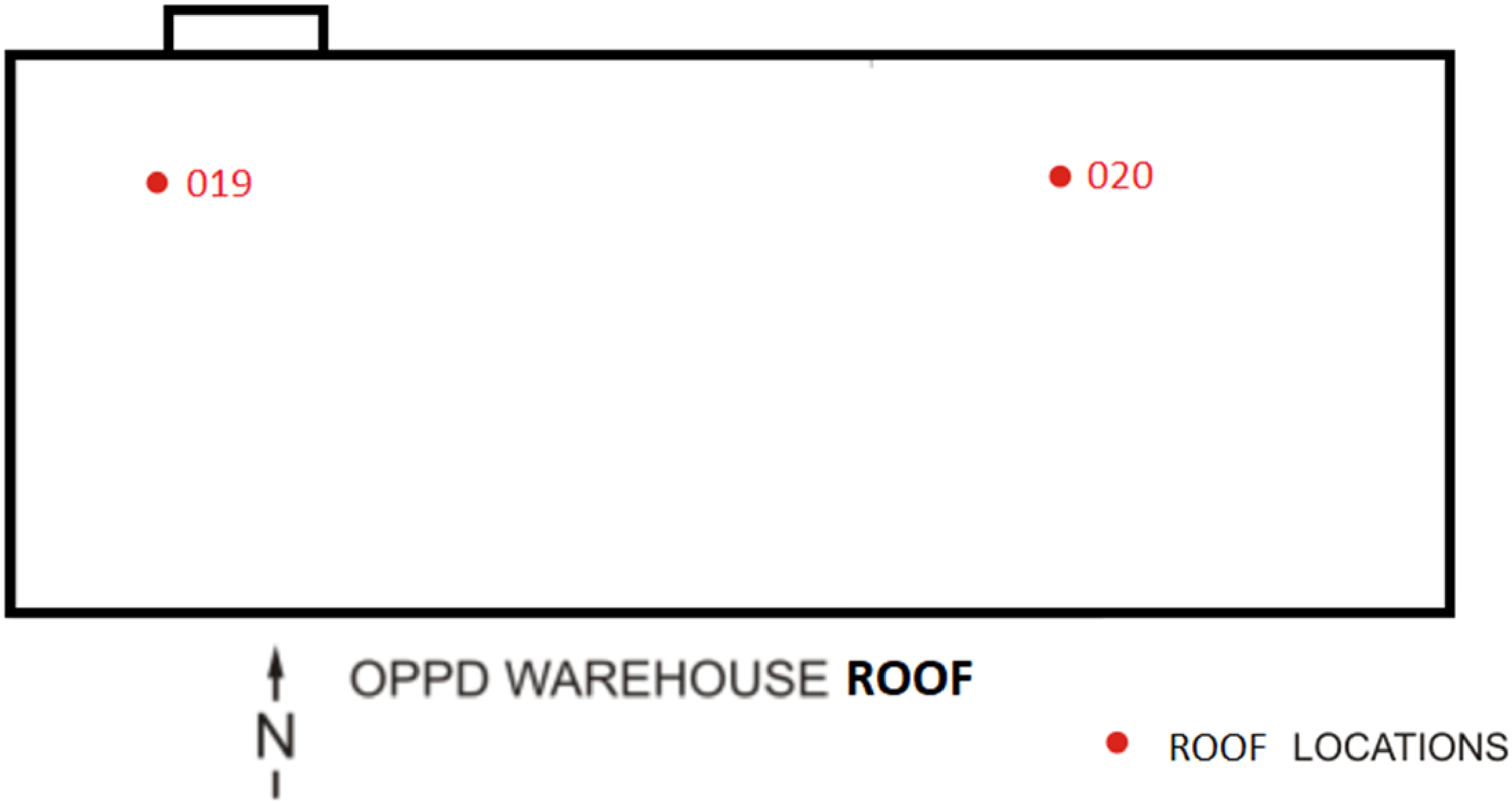


Table 2-87 BOP Buildings inside DA Scan Results

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
4200-01	394	283	536	0
4200-01Q	487	272	520	0
4200-02	340	272	520	0
4200-03	318	283	536	0
4200-04	327	283	536	0
4200-05	386	283	536	0
4200-06	489	291	549	0
4200-07	317	272	520	0
4200-08	517	291	549	0
4200-08Q	514	272	520	0
4200-09	441	262	506	0
4200-10	450	262	506	0
4200-11	491	262	506	0
4200-12	484	272	521	0
4200-13	536	488	822	0
4200-14	603	488	822	0
4200-15	358	262	506	0
4200-16	421	262	506	0
4200-17	516	272	521	0
4200-18	438	262	506	0
4200-19	466	291	549	0
4200-20	493	272	520	0
4200-21	531	291	549	0
4200-22	412	291	549	0
4200-23	49	291	549	0
4200-23	341	283	536	0
4200-24	355	272	520	0
4200-25	388	272	520	0
4200-26	337	283	536	0
4200-27	463	272	520	0
4300-01	814	586	951	0
4300-02	611	337	614	0
4300-03	566	337	614	0
4300-04	893	586	951	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
4300-05	500	337	614	0
4300-06	754	586	951	0
4300-07	575	337	614	0
4300-08	600	337	614	0
4300-09	559	337	614	0
4300-10	881	586	951	0
4300-11	778	586	951	0
4300-12	542	337	614	0
4300-13	723	586	951	0
4300-14	912	586	951	0
4300-14Q	580	431	744	0
4300-15	778	586	951	0
4300-16	460	337	614	0
4300-17	552	337	614	0
4300-18	363	256	497	0
4300-19	340	256	497	0
4300-20	915	586	951	0
4400-01	626	351	634	0
4400-02	446	351	634	0
4400-03	497	351	634	0
4400-04	476	351	634	0
4400-05	458	351	634	0
4400-06	502	351	634	0
4400-07	472	351	634	0
4400-08	437	351	634	0
4400-09	453	301	563	0
4400-10	431	301	563	0
4400-11	537	301	563	0
4400-12	452	301	563	0
4400-13	479	301	563	0
4400-14	533	301	563	0
4400-15	422	351	634	0
4400-16	559	301	563	0
4400-17	416	301	563	0
4400-18	341	351	634	0
4400-19	380	287	543	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
4400-20	373	287	543	0
4400-20Q	491	278	530	0
4500-01	563	313	579	0
4500-02	541	313	579	0
4500-03	563	313	579	0
4500-04	364	313	579	0
4500-05	503	313	579	0
4500-06	344	313	579	0
4500-07	344	313	579	0
4500-08	283	313	579	0
4500-09	401	342	621	0
4500-09Q	471	313	579	0
4500-10	379	342	621	0
4500-11	295	234	465	0
4500-12	420	342	621	0
4500-13	355	342	621	0
4500-14	239	234	465	0
4500-14Q	385	340	618	0
4500-15	392	342	621	0
4500-16	279	234	465	0
4500-17	371	342	621	0
4500-18	316	234	465	0
4500-19	549	490	824	0
4500-20	561	490	824	0
4600-01	637	405	709	0
4600-02	648	405	709	0
4600-03	499	405	709	0
4600-03Q	448	245	482	0
4600-04	484	405	709	0
4600-05	673	405	709	0
4600-06	530	405	709	0
4600-07	574	405	709	0
4600-08	618	405	709	0
4600-09	530	329	602	0
4600-10	295	329	602	0
4600-11	528	329	602	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
4600-12	580	329	602	0
4600-13	514	271	521	0
4600-14	321	271	521	0
4600-15	362	271	521	0
4600-16	416	271	521	0
4600-17	309	289	545	0
4600-18	466	289	545	0
4600-19	355	289	545	0
4600-20	349	289	545	0
4700-01	586	333	608	0
4700-02	525	333	608	0
4700-03	471	333	608	0
4700-04	591	333	608	0
4700-05	600	333	608	0
4700-06	527	333	608	0
4700-07	469	333	608	0
4700-08	485	333	608	0
4700-09	462	333	608	0
4700-10	603	333	608	0
4700-11	412	333	608	0
4700-12	568	333	608	0
4700-13	714	417	725	0
4700-14	525	333	608	0
4700-15	602	333	608	0
4700-16	576	333	608	0
4700-16Q	518	307	571	0
4700-17	565	333	608	0
4700-18	519	333	608	0
4700-19	549	333	608	0
4700-20	443	333	608	0
4700-21	422	284	539	0
4700-22	428	284	539	0
4700-23	513	351	634	0
4700-24	482	351	634	0
4700-25	521	351	634	0
4800-01	354	297	558	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
4800-01Q	376	219	442	0
4800-02	340	297	558	0
4800-03	405	297	558	0
4800-04	410	297	558	0
4800-05	548	297	558	0
4800-06	380	297	558	0
4800-07	334	297	558	0
4800-08	473	297	558	0
4800-09	396	297	558	0
4800-10	382	297	558	0
4800-11	417	219	442	0
4800-12	422	219	442	0
4800-13	386	219	442	0
4800-14	358	219	442	0
4800-15	336	219	442	0
4800-16	748	441	758	0
4800-17	285	219	442	0
4800-18	275	219	442	0
4800-19	442	248	485	0
4800-20	415	248	485	0
4900-01	769	517	860	0
4900-02	726	438	754	0
4900-03	679	438	754	0
4900-04	751	438	754	0
4900-05	682	438	754	0
4900-06	703	438	754	0
4900-07	634	438	754	0
4900-08	608	438	754	0
4900-08Q	745	512	854	0
4900-09	587	319	589	0
4900-10	519	319	589	0
4900-11	721	438	754	0
4900-12	508	319	589	0
4900-13	571	344	623	0
4900-14	599	328	601	0
4900-15	410	328	601	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
4900-16	580	328	601	0
4900-17	409	328	601	0
4900-18	528	328	601	0
4900-19	558	460	784	0
4900-20	524	460	784	0

Table 2-88 BOP Buildings Inside DA Static Measurement Results

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
4200X-3-CR-FBDX-001	113	0.0318
4200X-3-CR-FBDX-002	0	0.0000
4200X-3-CR-WBDX-003	0	0.0000
4200X-3-CR-WBDX-004	0	0.0000
4200X-3-CR-FBDX-005	0	0.0000
4200X-3-CR-WBDX-006	3	0.0008
4200X-3-CR-FBDX-007	44	0.0124
4200X-3-CR-WBDX-008	125	0.0352
4200X-3-CR-FBDX-009	0	0.0000
4200X-3-CR-FBDX-010	203	0.0572
4200X-3-CR-FBDX-011	0	0.0000
4200X-3-CR-WBDX-012	0	0.0000
4200X-3-CR-RBDX-013	59	0.0166
4200X-3-CR-RBDX-014	0	0.0000
4200X-3-CJ-FBDX-015	0	0.0000
4200X-3-CJ-FBDX-016	0	0.0000
4200X-3-CJ-WBDX-017	84	0.0237
4200X-3-CJ-FBDX-018	0	0.0000
4200X-3-CJ-WBDX-019	263	0.0741
4200X-3-CJ-FBDX-020	0	0.0000
4200X-3-CJ-WBDX-021	275	0.0775
4200X-3-CJ-WBDX-022	0	0.0000
4200X-3-CJ-WBDX-023	0	0.0000
4200X-3-CJ-WBDX-023	0	0.0000
4200X-3-CJ-FBDX-024	0	0.0000
4200X-3-CJ-FBDX-025	63	0.0177
4200X-3-CJ-WBDX-026	0	0.0000
4200X-3-CJ-FBDX-027	67	0.0189
4300X-3-CJ-FBDX-001	76	0.0214
4300X-3-CJ-FBDX-002	45	0.0127
4300X-3-CJ-FBDX-003	123	0.0346
4300X-3-CJ-FBDX-004	0	0.0000
4300X-3-CJ-FBDX-005	0	0.0000
4300X-3-CJ-FBDX-006	0	0.0000

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
4300X-3-CJ-WBDX-007	200	0.0563
4300X-3-CJ-WBDX-008	224	0.0631
4300X-3-CJ-WBDX-009	0	0.0000
4300X-3-CJ-FBDX-010	124	0.0349
4300X-3-CJ-FBDX-011	124	0.0349
4300X-3-CJ-WBDX-012	0	0.0000
4300X-3-CJ-FBDX-013	88	0.0248
4300X-3-CJ-FBDX-014	29	0.0082
4300X-3-CJ-FBDX-015	37	0.0104
4300X-3-CJ-WBDX-016	0	0.0000
4300X-3-CJ-WBDX-017	159	0.0448
4300X-3-CJ-WBDX-018	983	0.2769
4300X-3-CJ-WBDX-019	923	0.2600
4300X-3-CJ-FBDX-020	144	0.0406
4400X-3-CJ-FBDX-001	361	0.1017
4400X-3-CJ-FBDX-002	166	0.0468
4400X-3-CJ-FBDX-003	183	0.0515
4400X-3-CJ-FBDX-004	187	0.0527
4400X-3-CJ-FBDX-005	236	0.0665
4400X-3-CJ-FBDX-006	53	0.0149
4400X-3-CJ-FBDX-007	268	0.0755
4400X-3-CJ-FBDX-008	183	0.0515
4400X-3-CJ-FBDX-009	110	0.0310
4400X-3-CJ-FBDX-010	237	0.0668
4400X-3-CJ-FBDX-011	75	0.0211
4400X-3-CJ-FBDX-012	14	0.0039
4400X-3-CJ-FBDX-013	0	0.0000
4400X-3-CJ-FBDX-014	198	0.0558
4400X-3-CJ-WBDX-015	0	0.0000
4400X-3-CJ-WBDX-016	0	0.0000
4400X-3-CJ-WBDX-017	0	0.0000
4400X-3-CJ-WBDX-018	0	0.0000
4400X-3-CJ-RBDX-019	41	0.0115
4400X-3-CJ-RBDX-020	122	0.0344
4500X-3-CJ-FBDX-001	0	0.0000

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
4500X-3-CJ-FBDX-002	42	0.0118
4500X-3-CJ-FBDX-003	0	0.0000
4500X-3-CJ-WBDX-004	0	0.0000
4500X-3-CJ-FBDX-005	0	0.0000
4500X-3-CJ-FBDX-006	0	0.0000
4500X-3-CJ-FBDX-007	45	0.0127
4500X-3-CJ-WBDX-008	0	0.0000
4500X-3-CJ-FBDX-009	0	0.0000
4500X-3-CJ-FBDX-010	53	0.0149
4500X-3-CJ-WBDX-011	24	0.0068
4500X-3-CJ-FBDX-012	1	0.0003
4500X-3-CJ-FBDX-013	0	0.0000
4500X-3-CJ-WBDX-014	0	0.0000
4500X-3-CJ-FBDX-015	29	0.0082
4500X-3-CJ-WBDX-016	0	0.0000
4500X-3-CJ-FBDX-017	53	0.0149
4500X-3-CJ-WBDX-018	0	0.0000
4500X-3-CJ-RBDX-019	0	0.0000
4500X-3-CJ-RBDX-020	57	0.0161
4600X-3-CJ-FBDX-001	37	0.0104
4600X-3-CJ-FBDX-002	0	0.0000
4600X-3-CJ-FBDX-003	0	0.0000
4600X-3-CJ-FBDX-004	0	0.0000
4600X-3-CJ-FBDX-005	0	0.0000
4600X-3-CJ-WBDX-006	0	0.0000
4600X-3-CJ-FBDX-007	106	0.0299
4600X-3-CJ-FBDX-008	0	0.0000
4600X-3-CJ-FBDX-009	85	0.0239
4600X-3-CJ-FBDX-010	0	0.0000
4600X-3-CJ-FBDX-011	62	0.0175
4600X-3-CJ-FBDX-012	0	0.0000
4600X-3-CJ-FBDX-013	102	0.0287
4600X-3-CJ-WBDX-014	0	0.0000
4600X-3-CJ-FBDX-015	249	0.0701
4600X-3-CJ-FBDX-016	29	0.0082

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
4600X-3-CJ-RBDX-017	0	0.0000
4600X-3-CJ-RBDX-018	213	0.0600
4600X-3-CJ-RBDX-019	197	0.0555
4600X-3-CJ-RBDX-020	0	0.0000
4700X-3-CJ-FBDX-001	0	0.0000
4700X-3-CJ-FBDX-002	325	0.0915
4700X-3-CJ-FBDX-003	305	0.0859
4700X-3-CJ-FBDX-004	162	0.0456
4700X-3-CJ-FBDX-005	138	0.0389
4700X-3-CJ-FBDX-006	236	0.0665
4700X-3-CJ-FBDX-007	0	0.0000
4700X-3-CJ-FBDX-008	65	0.0183
4700X-3-CJ-FBDX-009	0	0.0000
4700X-3-CJ-FBDX-010	0	0.0000
4700X-3-CJ-FBDX-011	24	0.0068
4700X-3-CJ-FBDX-012	69	0.0194
4700X-3-CJ-WBDX-013	0	0.0000
4700X-3-CJ-FBDX-014	28	0.0079
4700X-3-CJ-FBDX-015	0	0.0000
4700X-3-CJ-FBDX-016	0	0.0000
4700X-3-CJ-FBDX-017	0	0.0000
4700X-3-CJ-FBDX-018	187	0.0527
4700X-3-CJ-FBDX-019	0	0.0000
4700X-3-CJ-FBDX-020	0	0.0000
4700X-3-CJ-WBDX-021	0	0.0000
4700X-3-CJ-WBDX-022	0	0.0000
4700X-3-CJ-WBDX-023	0	0.0000
4700X-3-CJ-WBDX-024	0	0.0000
4700X-3-CJ-WBDX-025	0	0.0000
4800X-3-CJ-FBDX-001	112	0.0315
4800X-3-CJ-FBDX-002	0	0.0000
4800X-3-CJ-FBDX-003	165	0.0465
4800X-3-CJ-FBDX-004	84	0.0237
4800X-3-CJ-FBDX-005	145	0.0408
4800X-3-CJ-FBDX-006	145	0.0408

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
4800X-3-CJ-FBDX-007	145	0.0408
4800X-3-CJ-FBDX-008	132	0.0372
4800X-3-CJ-FBDX-009	108	0.0304
4800X-3-CJ-FBDX-010	165	0.0465
4800X-3-CJ-WBDX-011	0	0.0000
4800X-3-CJ-WBDX-012	0	0.0000
4800X-3-CJ-WBDX-013	178	0.0501
4800X-3-CJ-WBDX-014	0	0.0000
4800X-3-CJ-WBDX-015	0	0.0000
4800X-3-CJ-WBDX-016	0	0.0000
4800X-3-CJ-FBDX-017	74	0.0208
4800X-3-CJ-WBDX-018	8	0.0023
4800X-3-CJ-RBDX-019	20	0.0056
4800X-3-CJ-RBDX-020	0	0.0000
4900X-3-CJ-FBDX-001	152	0.0428
4900X-3-CJ-FBDX-002	50	0.0141
4900X-3-CJ-FBDX-003	244	0.0687
4900X-3-CJ-FBDX-004	74	0.0208
4900X-3-CJ-FBDX-005	0	0.0000
4900X-3-CJ-FBDX-006	54	0.0152
4900X-3-CJ-FBDX-007	0	0.0000
4900X-3-CJ-FBDX-008	50	0.0141
4900X-3-CJ-WBDX-009	1087	0.3062
4900X-3-CJ-WBDX-010	169	0.0476
4900X-3-CJ-FBDX-011	0	0.0000
4900X-3-CJ-WBDX-012	0	0.0000
4900X-3-CJ-FBDX-013	113	0.0318
4900X-3-CJ-FBDX-014	262	0.0738
4900X-3-CJ-FBDX-015	85	0.0239
4900X-3-CJ-FBDX-016	312	0.0879
4900X-3-CJ-FBDX-017	0	0.0000
4900X-3-CJ-FBDX-018	0	0.0000
4900X-3-CJ-RBDX-019	0	0.0000
4900X-3-CJ-RBDX-020	210	0.0592
4200X-3-CQ-FBDX-001	113	0.0318

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
4200X-3-CQ-WBDX-008	75	0.0211
4400X-3-CQ-RBDX-020	119	0.0335
4500X-3-CQ-FBDX-009	115	0.0324
4500X-3-CQ-WBDX-014	0	0.0000
4600X-3-CQ-FBDX-003	169	0.0476
4700X-3-CQ-FBDX-014	352	0.0992
4700X-3-CQ-FBDX-016	0	0.0000
4800X-3-CQ-FBDX-001	247	0.0696
4900X-3-CQ-FBDX-008	0	0.0000

(a) Action level is equivalent to 50% of the Screening Value for Co-60, or 3,550 dpm/100 cm^2.

Table 2-89 BOP Buildings Inside DA Surface Measuremements Summary Statistics

Random				
Mean ($dpm/100cm^2$)	**Median ($dpm/100cm^2$)**	**Max ($dpm/100cm^2$)**	**Min ($dpm/100cm^2$)**	**Std. Dev. ($dpm/100cm^2$)**
39	0	203	0	64

Judgmental				
Mean ($dpm/100cm^2$)	**Median ($dpm/100cm^2$)**	**Max ($dpm/100cm^2$)**	**Min ($dpm/100cm^2$)**	**Std. Dev. ($dpm/100cm^2$)**
86	29	1087	0	155

Combined				
Mean ($dpm/100cm^2$)	**Median ($dpm/100cm^2$)**	**Max ($dpm/100cm^2$)**	**Min ($dpm/100cm^2$)**	**Std. Dev. ($dpm/100cm^2$)**
83	24	1087	0	150

Total Number of Measurements	
Random	14
Judgmental	159
QC	10

Random	
Fraction >1	0
Maximum Fraction:	0.0642

Judgmental	
Fraction >1	0
Maximum Fraction:	0.3440

Figure 2-61 Administrative Office Building First Floor Surface Measurement Locations

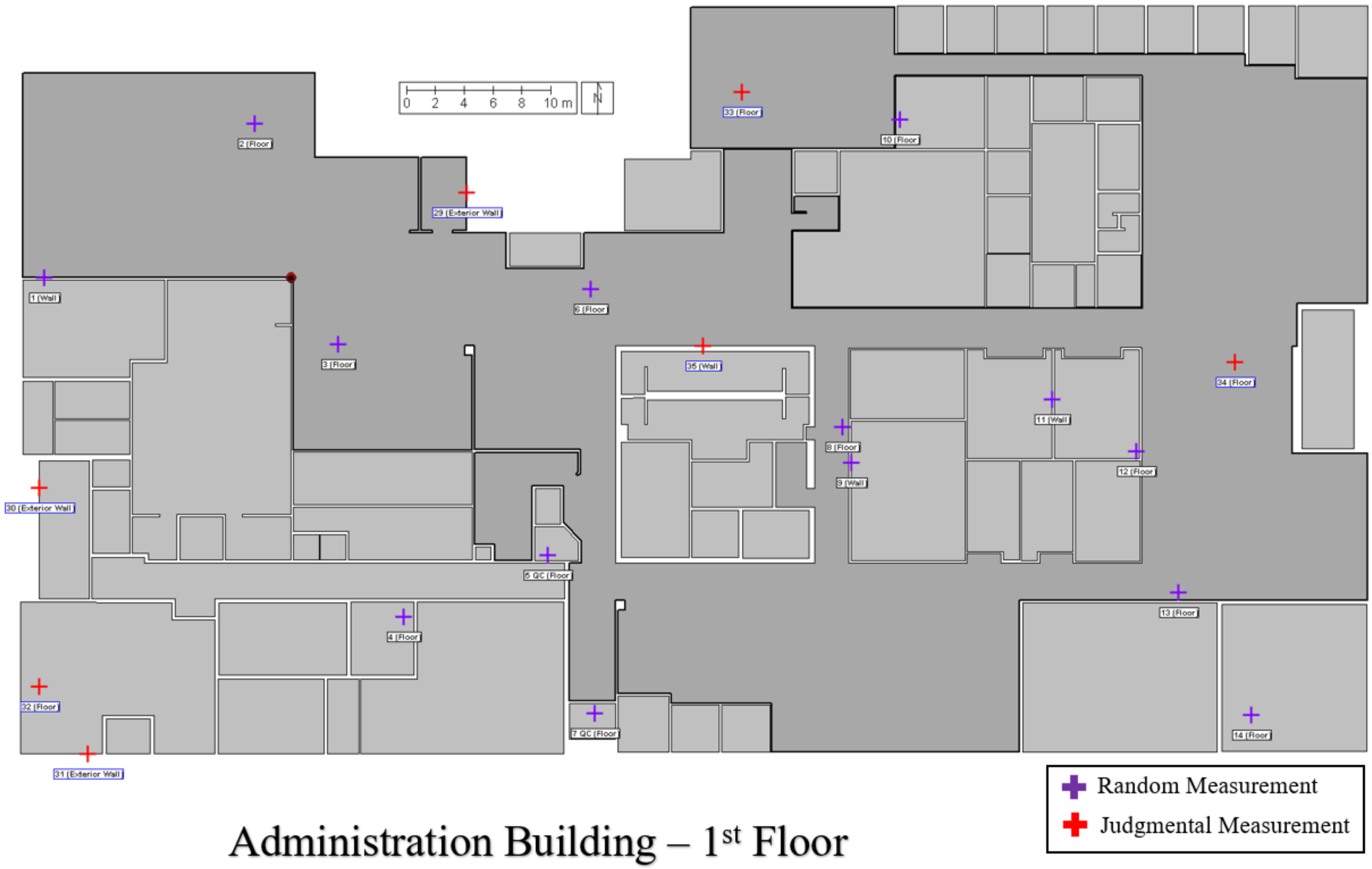


Figure 2-62 Administrative Office Building Second Floor Surface Measurement Locations

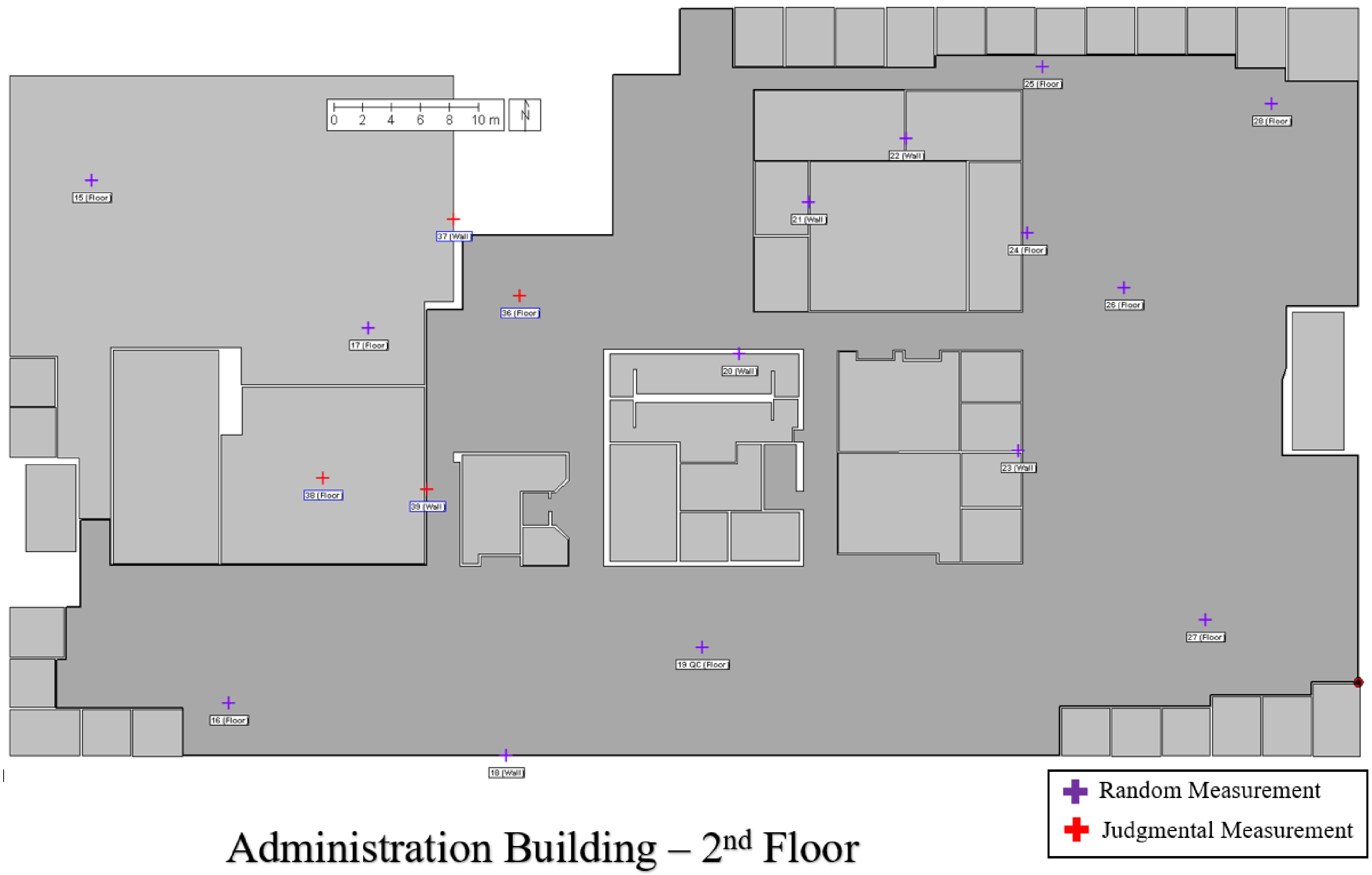


Figure 2-63 Administrative Office Building Roof Surface Measurement Locations

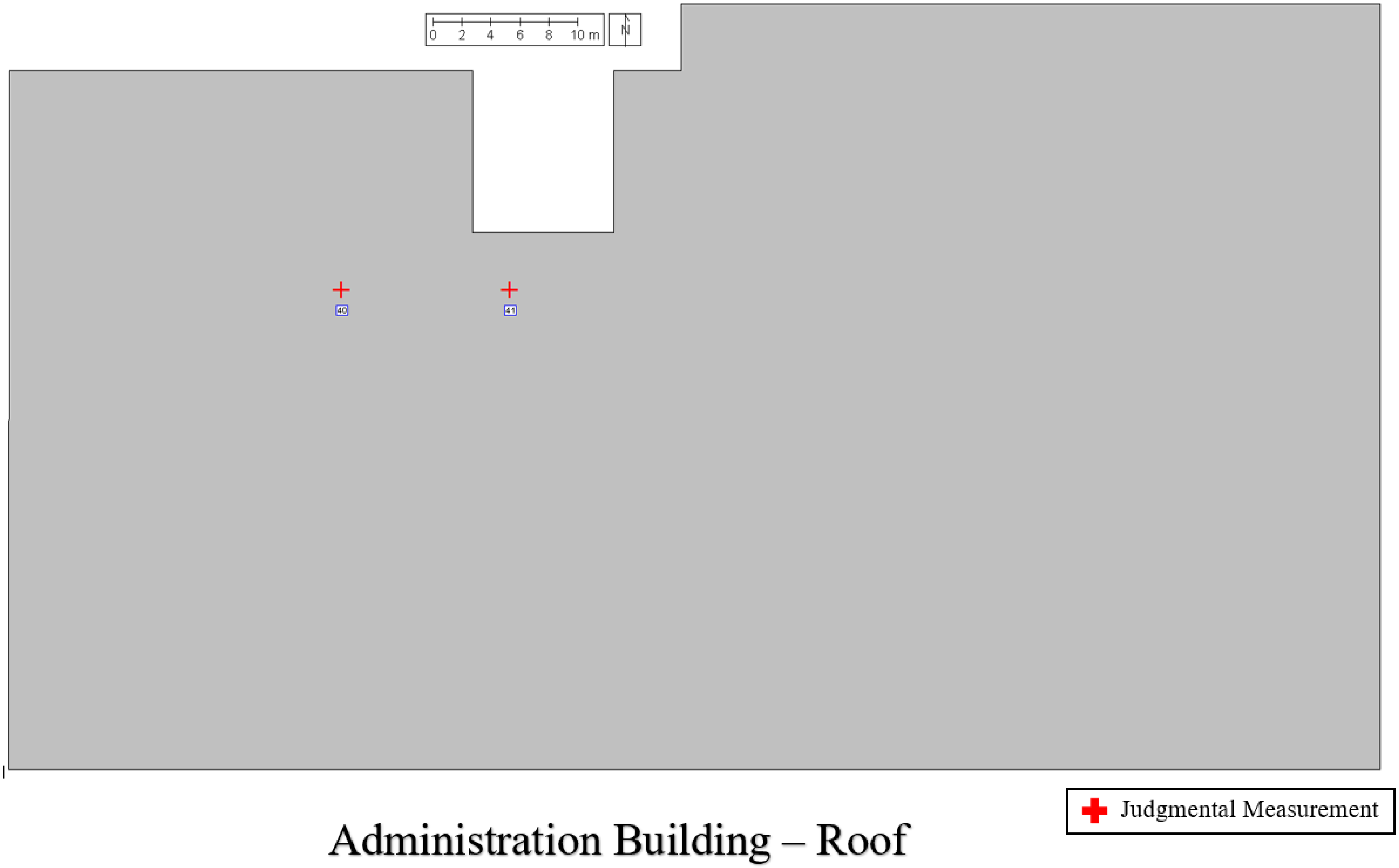


Figure 2-64 Training Center Surface Measurement Locations

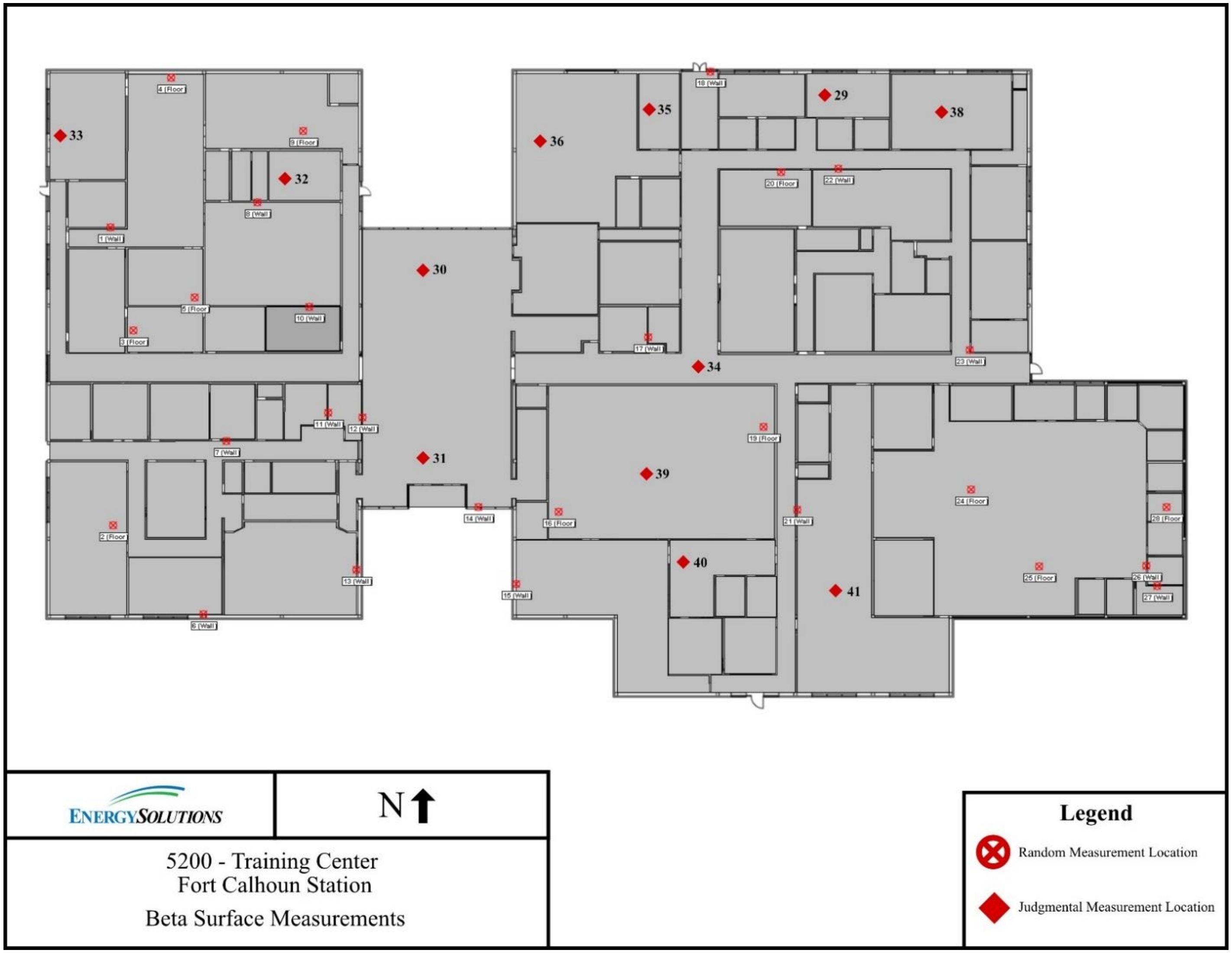


Figure 2-65 Training Center Roof Surface Measurement Locations

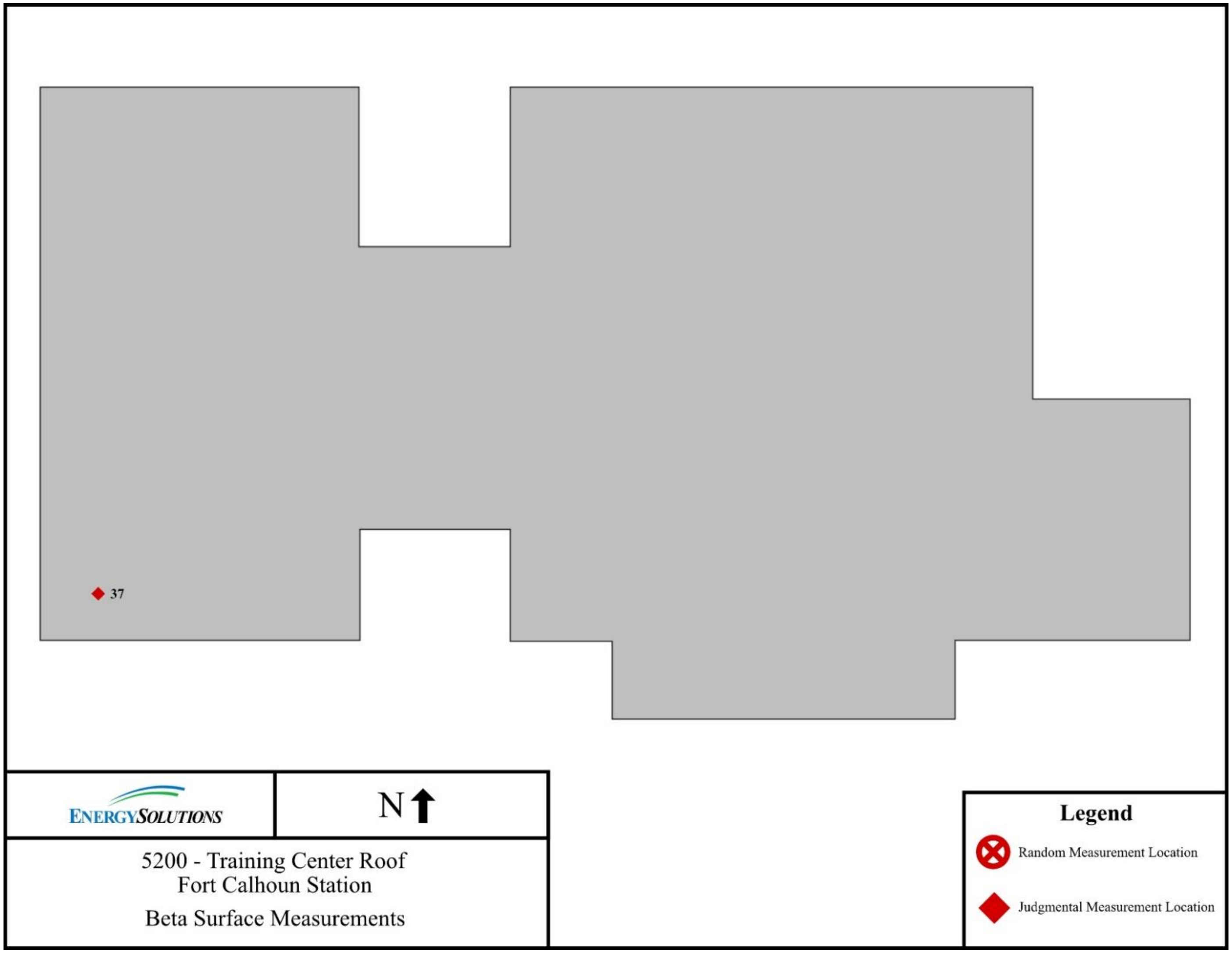


Figure 2-66 FLEX Building Surface Measurement Locations

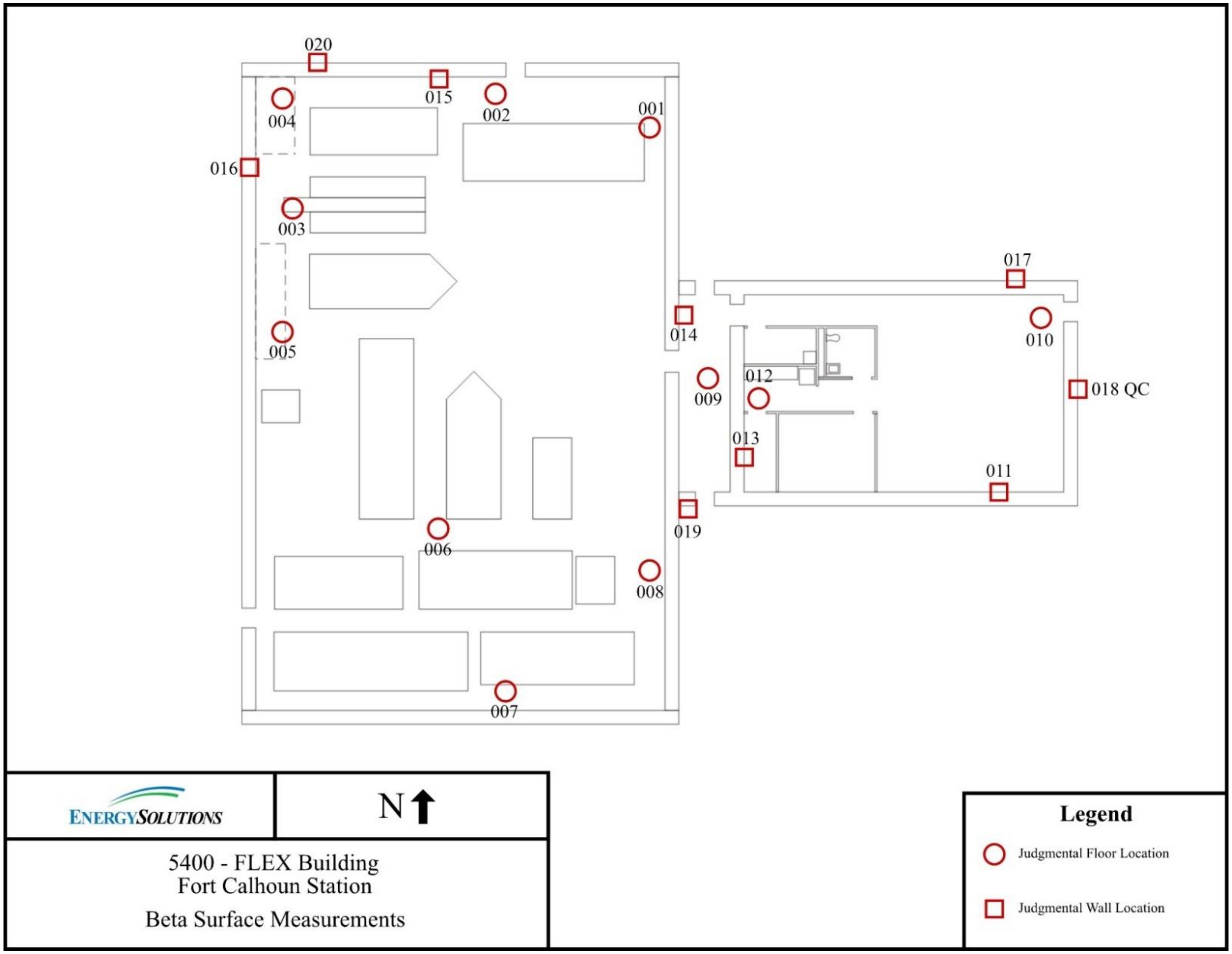


Figure 2-67 New Maintenance Storage Shed Surface Measurement Locations

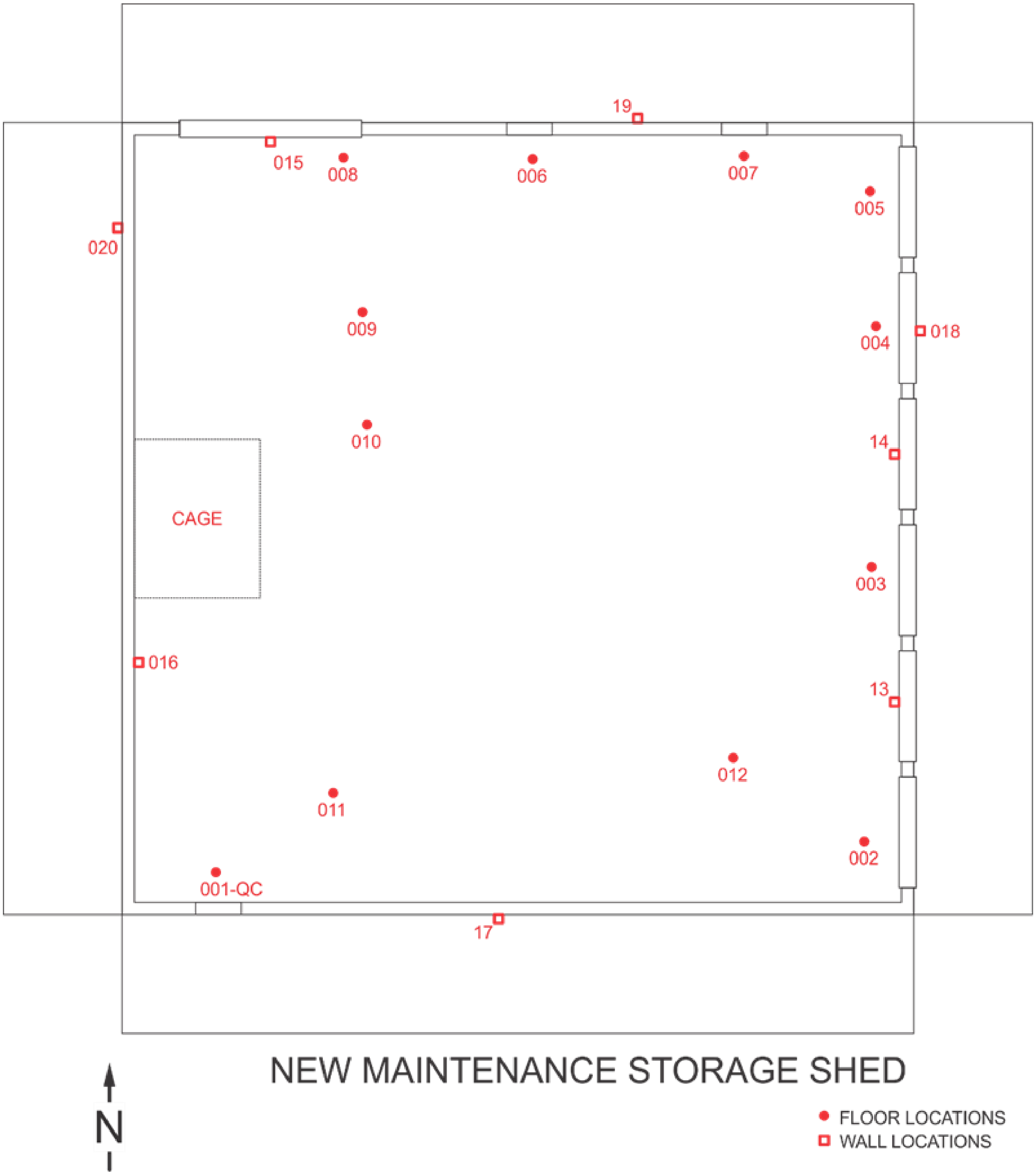


Figure 2-68 Chemical Pump House Surface Measurement Locations

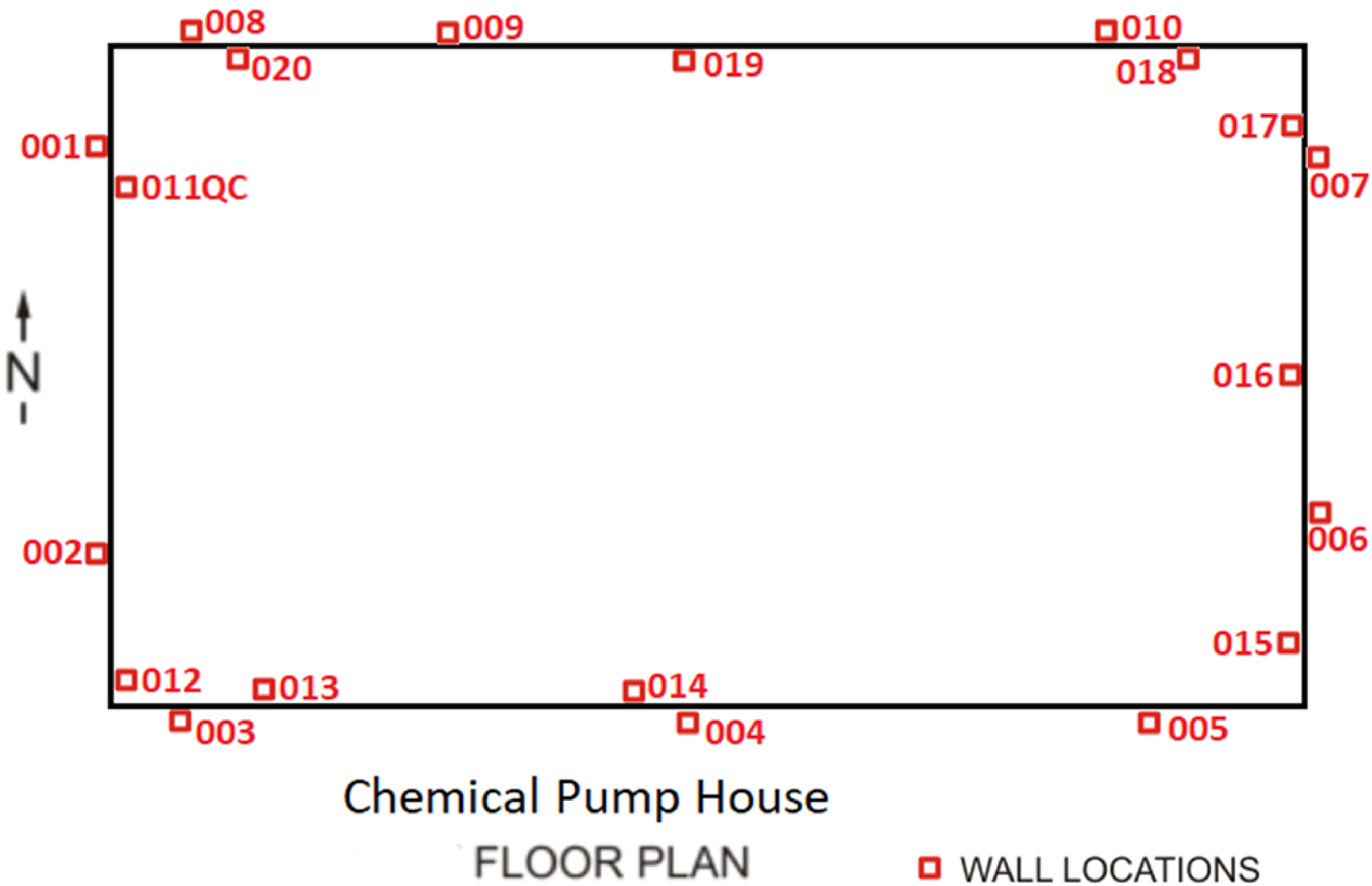


Figure 2-69 Storage Shed Surface Measurement Locations

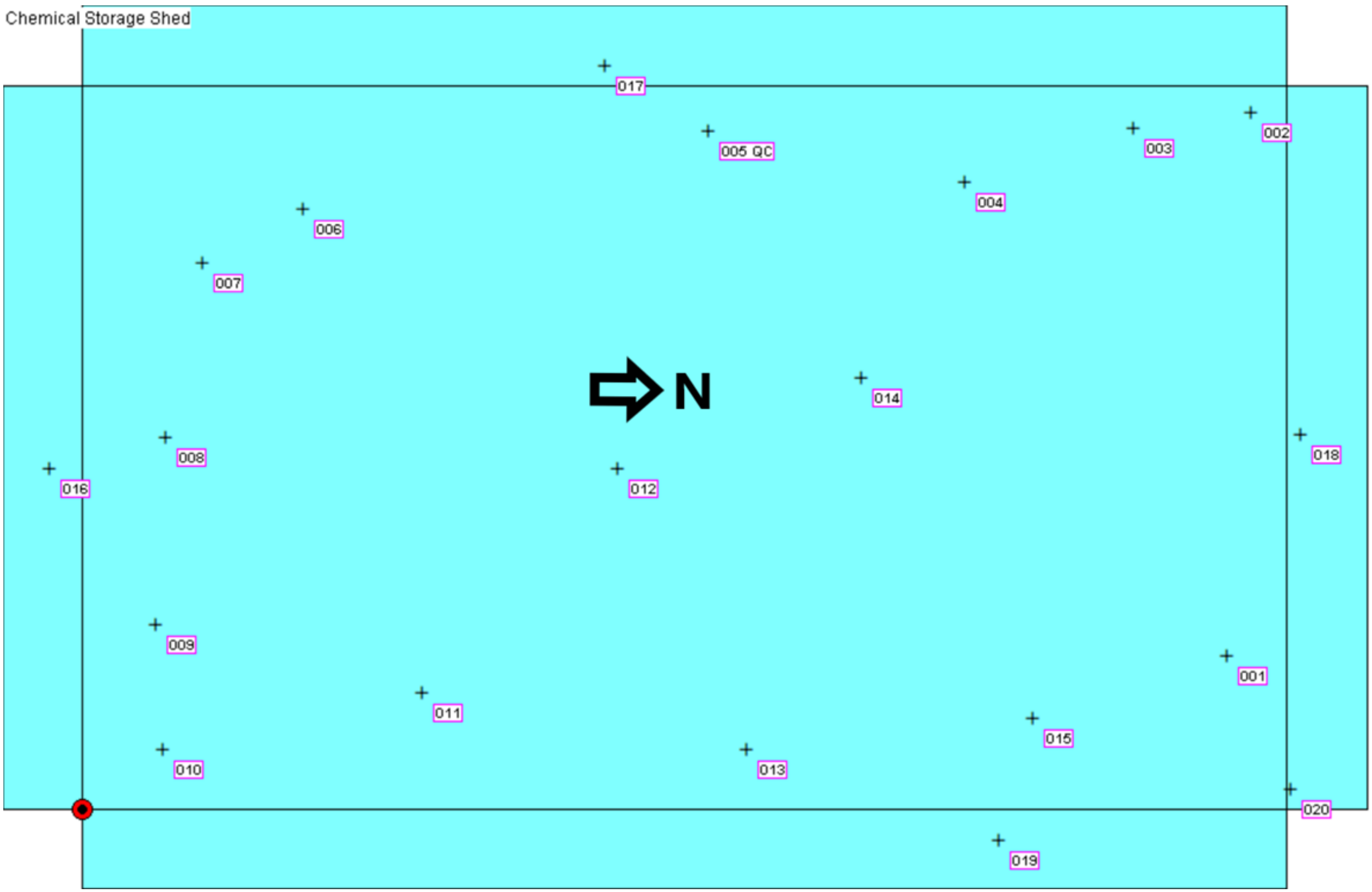


Figure 2-70 Sanitary Lift Stations Surface Measurement Locations

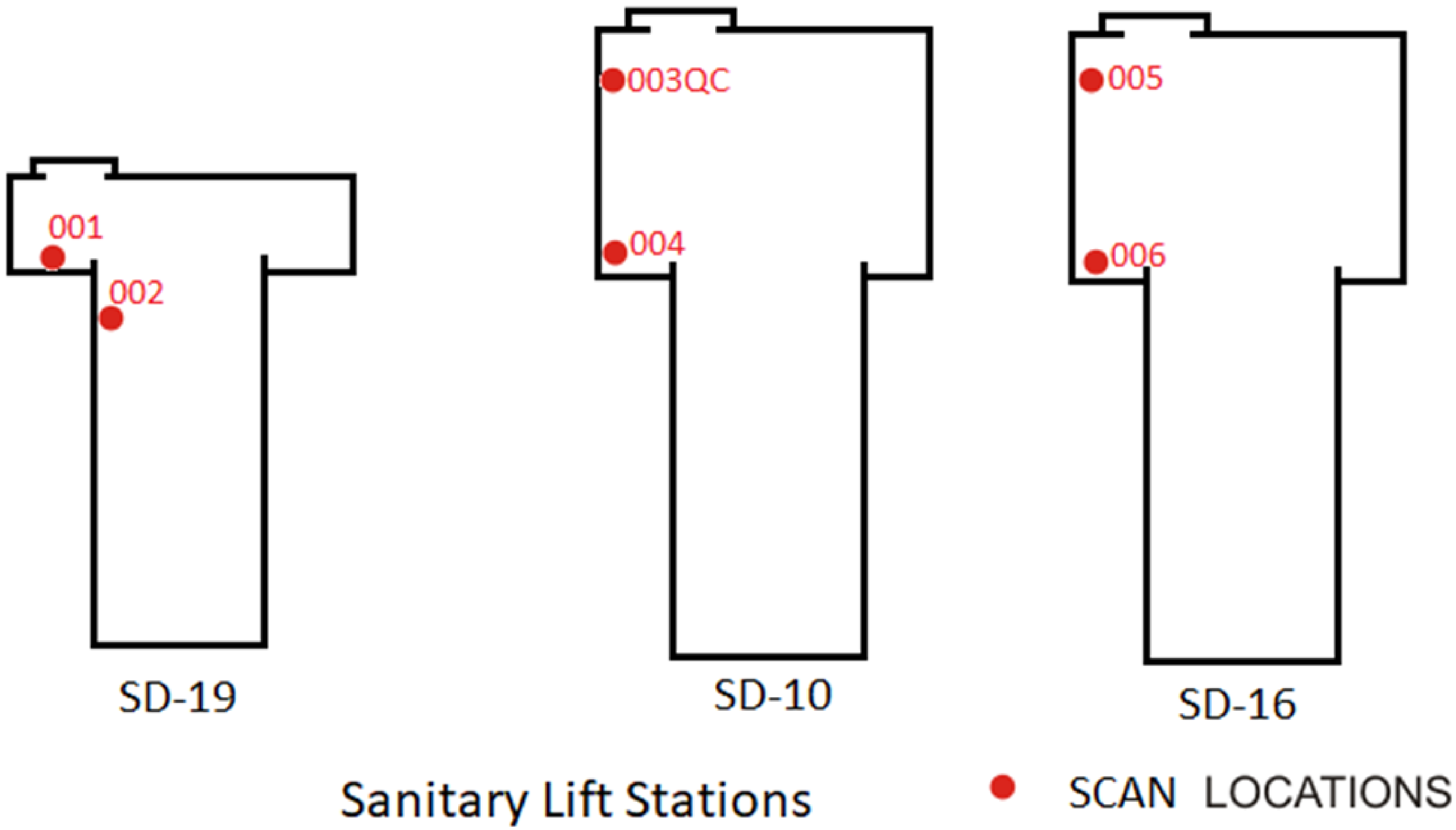


Table 2-90 BOP Buildsings Outside DA Scan Results

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
5700-016	413	352	635	0
5700-017	448	352	635	0
5700-018	550	352	635	0
5700-019	579	352	635	0
5700-020	427	352	635	0
QC 5700-005	551	352	635	0
5700-009	512	340	618	0
5700-010	599	340	618	0
5700-011	606	340	618	0
5700-012	509	340	618	0
5700-013	509	340	618	0
5700-014	497	340	618	0
5700-015	526	340	618	0
5600-001	281	224	449	0
5600-002	291	224	449	0
5600-003	307	224	449	0
5600-004	394	224	449	0
5600-005	301	224	449	0
5600-006	398	224	449	0
5600-007	442	224	449	0
5600-008	253	224	449	0
5600-009	318	224	449	0
5600-010	351	224	449	0
5600-011	429	224	449	0
5600-012	404	224	449	0
5600-013	398	224	449	0
5600-014	429	224	449	0
5600-015	301	224	449	0
5600-016	317	224	449	0
5600-017	390	224	449	0
5600-018	402	224	449	0
5600-019	307	224	449	0
5600-020	312	224	449	0
5700-001	476	371	824	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
5700-002	416	371	824	0
5700-003	420	371	824	0
5700-004	465	371	824	0
5700-005	418	371	824	0
5700-006	429	371	824	0
5700-007	454	371	824	0
5700-008	428	371	824	0
5500-001	567	353	636	0
5500-002	471	353	636	0
5500-003	513	353	636	0
5500-004	577	353	636	0
5500-005	544	353	636	0
5500-006	562	353	636	0
5500-007	623	353	636	0
5500-008	558	353	636	0
5500-009	576	353	636	0
5500-010	550	353	636	0
5500-011	578	353	636	0
5500-012	518	353	636	0
5400-001	537	409	714	0
5400-002	531	409	714	0
5400-003	468	409	714	0
5400-004	505	409	714	0
5400-005	499	409	714	0
5400-006	495	409	714	0
5400-007	467	409	714	0
5400-008	526	409	714	0
5400-009	546	409	714	0
5400-010	495	409	714	0
QC 5400-018	467	409	714	0
5400-011	575	328	601	0
5400-012	585	328	601	0
5400-013	531	328	601	0
5400-014	553	328	601	0
5400-015	502	328	601	0
5400-016	478	328	601	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
5400-017	578	328	601	0
5400-018	515	328	601	0
5400-019	590	328	601	0
5400-020	519	328	601	0
5500-013	324	204	419	0
5500-014	337	204	419	0
5500-015	352	204	419	0
5500-016	379	204	419	0
5500-017	385	204	419	0
5500-018	418	204	419	0
5500-019	361	204	419	0
5500-020	349	204	419	0
QC 5500-001	603	358	644	0
5200-005	435	376	668	0
5200-009	638	376	668	0
5200-010	583	376	668	0
5200-034	637	376	668	0
5200-041	487	269	516	0
5200-040	341	269	516	0
5200-039	461	269	516	0
5200-019	490	269	516	0
5200-016	510	269	516	0
5200-028	424	269	516	0
5200-029	761	461	786	0
5200-035	572	350	633	0
QC 5200-031	479	350	633	0
QC 5800-003	615	360	646	0
5800-001	280	246	483	0
5800-002	306	246	483	0
5800-003	456	368	657	0
5800-004	396	368	657	0
5800-005	480	368	657	0
5800-006	384	368	657	0
5100-008	522	287	543	0
5100-006	453	287	543	0
5100-007	456	287	543	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
5100-013	418	287	543	0
5100-014	470	287	543	0
5100-034	403	287	543	0
5100-012	450	287	543	0
5100-041	335	209	426	0
5100-005	398	209	426	0
5100-004	617	473	802	0
5100-024	646	473	802	0
5100-026	747	473	802	0
5100-017	729	473	802	0
5100-019	419	241	476	0
5100-016	414	241	476	0
5100-015	337	241	476	0
QC 5100-007	421	321	591	0
5100-029	747	448	768	0
5100-030	764	448	768	0
5100-033	409	282	536	0
5100-010	427	282	536	0
QC 5100-019	435	282	536	0
5100-031	642	473	801	0
5200-001	458	270	518	0
5200-003	497	270	518	0
5200-011	433	270	518	0
5200-008	495	270	518	0
5200-017	453	270	518	0
5200-013	403	270	518	0
5200-014	366	270	518	0
5200-015	447	270	518	0
5200-021	380	270	518	0
QC 5200-007	301	270	518	0
5200-027	415	270	518	0
5200-025	260	268	516	0
5200-004	678	425	736	0
5100-009	366	261	505	0
5100-035	356	261	505	0
5100-001	281	261	505	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
5100-011	308	261	505	0
5100-039	328	261	505	0
5100-040	296	199	411	0
5100-021	372	199	411	0
5100-022	309	199	411	0
5100-025	310	199	411	0
5100-028	403	199	411	0
QC 5100-005	435	341	619	0
5100-036	411	341	619	0
5100-027	432	341	619	0
5100-032	642	473	801	0
5100-038	382	293	801	0
5100-020	639	368	657	0
5100-036	614	492	827	0
5100-037	603	492	827	0
5200-030	663	418	727	0
5200-031	671	418	727	0
5200-002	667	418	727	0
5200-032	674	418	727	0
5200-033	716	418	727	0
5200-024	638	418	727	0
5200-012	431	214	434	0
5200-018	374	219	442	0
5200-022	344	219	442	0
5200-023	436	219	442	0
5200-016	369	219	442	0
QC 5200-013	362	219	442	0
5200-007	301	219	442	0
5200-006	289	219	442	0
5200-003	413	219	442	0
5100-018	276	182	386	0
5100-020	246	182	386	0
5100-023	267	182	386	0
5100-037	272	182	386	0
5100-002	615	515	858	0
5100-003	590	515	858	0

Location	Scan Logged Result (cpm)	Avg Background (cpm)	Action Level (cpm)	Scan Alarms
QC 5600-011	340	192	401	0

Table 2-91 BOP Buildings Outside DA Static Results

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
5100X-3-CR-WBDX-001	83	0.0234
5100X-3-CR-FBDX-002	0	0.0000
5100X-3-CR-FBDX-003	0	0.0000
5100X-3-CR-FBDX-004	0	0.0000
5100X-3-CR-FBDX-005	221	0.0623
5100X-3-CR-FBDX-006	244	0.0687
5100X-3-CR-FBDX-007	158	0.0445
5100X-3-CR-FBDX-008	97	0.0273
5100X-3-CR-WBDX-009	19	0.0054
5100X-3-CR-FBDX-010	56	0.0158
5100X-3-CR-WBDX-011	0	0.0000
5100X-3-CR-FBDX-012	24	0.0068
5100X-3-CR-FBDX-013	65	0.0183
5100X-3-CR-FBDX-014	0	0.0000
5100X-3-CR-FBDX-015	0	0.0000
5100X-3-CR-FBDX-016	1	0.0003
5100X-3-CR-FBDX-017	0	0.0000
5100X-3-CR-WBDX-018	190	0.0535
5100X-3-CR-FBDX-019	0	0.0000
5100X-3-CR-WBDX-020	107	0.0301
5100X-3-CR-WBDX-021	168	0.0473
5100X-3-CR-WBDX-022	96	0.0270
5100X-3-CR-WBDX-023	23	0.0065
5100X-3-CR-FBDX-024	0	0.0000
5100X-3-CR-FBDX-025	108	0.0304
5100X-3-CR-FBDX-026	0	0.0000
5100X-3-CR-FBDX-027	0	0.0000
5100X-3-CR-FBDX-028	124	0.0349
5100X-3-CJ-WBDX-029	0	0.0000
5100X-3-CJ-WBDX-030	0	0.0000
5100X-3-CJ-WBDX-031	0	0.0000
5100X-3-CJ-FBDX-032	0	0.0000
5100X-3-CJ-FBDX-033	105	0.0296
5100X-3-CJ-FBDX-034	45	0.0127

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
5100X-3-CJ-WBDX-035	91	0.0256
5100X-3-CJ-FBDX-036	0	0.0000
5100X-3-CJ-WBDX-037	51	0.0144
5100X-3-CJ-WBDX-038	0	0.0000
5100X-3-CJ-WBDX-039	0	0.0000
5100X-3-CJ-RBDX-040	100	0.0282
5100X-3-CJ-RBDX-041	0	0.0000
5200X-3-CR-WBDX-001	93	0.0262
5200X-3-CR-FBDX-002	0	0.0000
5200X-3-CR-FBDX-003	242	0.0682
5200X-3-CR-FBDX-004	0	0.0000
5200X-3-CR-FBDX-005	0	0.0000
5200X-3-CR-WBDX-006	0	0.0000
5200X-3-CR-WBDX-007	0	0.0000
5200X-3-CR-WBDX-008	44	0.0124
5200X-3-CR-FBDX-009	196	0.0552
5200X-3-CR-WBDX-010	0	0.0000
5200X-3-CR-FBDX-011	74	0.0208
5200X-3-CR-WBDX-012	100	0.0282
5200X-3-CR-FBDX-013	0	0.0000
5200X-3-CR-FBDX-014	0	0.0000
5200X-3-CR-FBDX-015	0	0.0000
5200X-3-CR-FBDX-016	0	0.0000
5200X-3-CR-FBDX-017	119	0.0335
5200X-3-CR-WBDX-018	0	0.0000
5200X-3-CR-FBDX-019	0	0.0000
5200X-3-CR-FBDX-020	73	0.0206
5200X-3-CR-FBDX-021	0	0.0000
5200X-3-CR-WBDX-022	16	0.0045
5200X-3-CR-WBDX-023	0	0.0000
5200X-3-CR-FBDX-024	0	0.0000
5200X-3-CR-FBDX-025	0	0.0000
5200X-3-CR-WBDX-026	40	0.0113
5200X-3-CR-WBDX-027	0	0.0000
5200X-3-CR-FBDX-028	172	0.0485

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
5200X-3-CJ-FBDX-029	0	0.0000
5200X-3-CJ-FBDX-030	259	0.0730
5200X-3-CJ-FBDX-031	0	0.0000
5200X-3-CJ-FBDX-032	0	0.0000
5200X-3-CJ-FBDX-033	0	0.0000
5200X-3-CJ-FBDX-034	248	0.0699
5200X-3-CJ-FBDX-035	0	0.0000
5200X-3-CJ-FBDX-036	0	0.0000
5200X-3-CJ-FBDX-037	0	0.0000
5200X-3-CJ-FBDX-038	111	0.0313
5200X-3-CJ-FBDX-039	10	0.0028
5200X-3-CJ-FBDX-040	0	0.0000
5200X-3-CJ-FBDX-041	30	0.0085
5400X-3-CJ-FBDX-001	2	0.0006
5400X-3-CJ-FBDX-002	63	0.0177
5400X-3-CJ-FBDX-003	79	0.0223
5400X-3-CJ-FBDX-004	0	0.0000
5400X-3-CJ-FBDX-005	0	0.0000
5400X-3-CJ-FBDX-006	0	0.0000
5400X-3-CJ-FBDX-007	132	0.0372
5400X-3-CJ-FBDX-008	71	0.0200
5400X-3-CJ-FBDX-009	71	0.0200
5400X-3-CJ-FBDX-010	172	0.0485
5400X-3-CJ-FBDX-011	141	0.0397
5400X-3-CJ-FBDX-012	205	0.0577
5400X-3-CJ-WBDX-013	149	0.0420
5400X-3-CJ-WBDX-014	169	0.0476
5400X-3-CJ-WBDX-015	244	0.0687
5400X-3-CJ-WBDX-016	256	0.0721
5400X-3-CJ-WBDX-017	189	0.0532
5400X-3-CJ-WBDX-018	65	0.0183
5400X-3-CJ-WBDX-019	2	0.0006
5400X-3-CJ-WBDX-020	129	0.0363
5500X-3-CJ-FBDX-001	0	0.0000
5500X-3-CJ-FBDX-002	0	0.0000

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
5500X-3-CJ-FBDX-003	0	0.0000
5500X-3-CJ-FBDX-004	140	0.0394
5500X-3-CJ-FBDX-005	0	0.0000
5500X-3-CJ-FBDX-006	0	0.0000
5500X-3-CJ-FBDX-007	0	0.0000
5500X-3-CJ-FBDX-008	0	0.0000
5500X-3-CJ-FBDX-009	0	0.0000
5500X-3-CJ-FBDX-010	22	0.0062
5500X-3-CJ-FBDX-011	0	0.0000
5500X-3-CJ-FBDX-012	0	0.0000
5500X-3-CJ-WBDX-013	0	0.0000
5500X-3-CJ-WBDX-014	46	0.0130
5500X-3-CJ-WBDX-015	0	0.0000
5500X-3-CJ-WBDX-016	0	0.0000
5500X-3-CJ-WBDX-017	54	0.0152
5500X-3-CJ-WBDX-018	0	0.0000
5500X-3-CJ-FBDX-019	147	0.0414
5500X-3-CJ-FBDX-020	0	0.0000
5600X-3-CJ-WBDX-001	0	0.0000
5600X-3-CJ-WBDX-002	28	0.0079
5600X-3-CJ-WBDX-003	0	0.0000
5600X-3-CJ-WBDX-004	0	0.0000
5600X-3-CJ-WBDX-005	0	0.0000
5600X-3-CJ-WBDX-006	58	0.0163
5600X-3-CJ-WBDX-007	0	0.0000
5600X-3-CJ-WBDX-008	96	0.0270
5600X-3-CJ-WBDX-009	9	0.0025
5600X-3-CJ-WBDX-010	0	0.0000
5600X-3-CJ-WBDX-011	0	0.0000
5600X-3-CJ-WBDX-012	0	0.0000
5600X-3-CJ-WBDX-013	0	0.0000
5600X-3-CJ-WBDX-014	0	0.0000
5600X-3-CJ-WBDX-015	0	0.0000
5600X-3-CJ-WBDX-016	0	0.0000
5600X-3-CJ-WBDX-017	0	0.0000

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
5600X-3-CJ-WBDX-018	0	0.0000
5600X-3-CJ-WBDX-019	0	0.0000
5600X-3-CJ-WBDX-020	0	0.0000
5700X-3-CJ-FBDX-001	220	0.0620
5700X-3-CJ-FBDX-002	0	0.0000
5700X-3-CJ-FBDX-003	0	0.0000
5700X-3-CJ-FBDX-004	0	0.0000
5700X-3-CJ-FBDX-005	60	0.0169
5700X-3-CJ-FBDX-006	0	0.0000
5700X-3-CJ-FBDX-007	0	0.0000
5700X-3-CJ-FBDX-008	21	0.0059
5700X-3-CJ-FBDX-009	4	0.0011
5700X-3-CJ-FBDX-010	119	0.0335
5700X-3-CJ-FBDX-011	0	0.0000
5700X-3-CJ-FBDX-012	0	0.0000
5700X-3-CJ-FBDX-013	19	0.0054
5700X-3-CJ-FBDX-014	23	0.0065
5700X-3-CJ-FBDX-015	123	0.0346
5700X-3-CJ-WBDX-016	0	0.0000
5700X-3-CJ-WBDX-017	0	0.0000
5700X-3-CJ-WBDX-018	0	0.0000
5700X-3-CJ-WBDX-019	0	0.0000
5700X-3-CJ-WBDX-020	0	0.0000
5800X-3-CJ-WBDX-001	12	0.0034
5800X-3-CJ-WBDX-002	28	0.0079
5800X-3-CJ-WBDX-003	0	0.0000
5800X-3-CJ-WBDX-004	0	0.0000
5800X-3-CJ-WBDX-005	0	0.0000
5800X-3-CJ-WBDX-006	0	0.0000
5100X-3-CQ-FBDX-005	0	0.0000
5100X-3-CQ-FBDX-007	1	0.0003
5100X-3-CQ-FBDX-019	0	0.0000
5200X-3-CQ-FBDX-007	0	0.0000
5200X-3-CQ-WBDX-013	28	0.0079
5200X-3-CQ-FBDX-031	48	0.0135

Measurement ID	Gross Beta Activity (dpm/100cm^2)	Fraction of Action Level[a]
5400X-3-CQ-WBDX-018	172	0.0485
5500X-3-CQ-FBDX-001	6	0.0017
5600X-3-CQ-WBDX-011	87	0.0245
5700X-3-CQ-FBDX-005	263	0.0741
5800X-3-CQ-WBDX-003	0	0.0000

Table 2-92 BOP Buildings Outside DA Surface Measurements Summary Statistics

Random				
Mean (dpm/100cm^2)	**Median (dpm/100cm^2)**	**Max (dpm/100cm^2)**	**Min (dpm/100cm^2)**	**Std. Dev. (dpm/100cm^2)**
53	1	244	0	72

Judgmental				
Mean (dpm/100cm^2)	**Median (dpm/100cm^2)**	**Max (dpm/100cm^2)**	**Min (dpm/100cm^2)**	**Std. Dev. (dpm/100cm^2)**
39	0	259	0	68

Combined				
Mean (dpm/100cm^2)	**Median (dpm/100cm^2)**	**Max (dpm/100cm^2)**	**Min (dpm/100cm^2)**	**Std. Dev. (dpm/100cm^2)**
44	0	259	0	69

Total Number of Measurements	
Random	56
Judgmental	112
QC	11

Random	
Fraction >1	0
Maximum Fraction:	0.0687

Judgmental	
Fraction >1	0
Maximum Fraction:	0.0730

FORT CALHOUN STATION DECOMMISSIONING PROJECT
LICENSE TERMINATION PLAN

CHAPTER 3
IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

REVISION 0

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ABBREVIATIONS

ACM	Asbestos Containing Material
ALARA	as low as is reasonably achievable
AMSL	above mean sea level
CARP	Chemistry and Radiation Protection
CNW	Chicago and Northwestern
CPP	calculations and position paper
CVS	contamination verification survey
D&D	dismantlement and decontamination
DA	deconstruction area
DAW	dry active waste
DCGL	derived concentration guideline level
DSAR	Defueled Safety Analysis Report
FCS	Fort Calhoun Station
FSS	final status survey
GTCC	greater than Class C
HBH	hemispheric bottom head
HSA	historical site assessment
HSM	horizontal storage module
IOF	ISFSI Operation Facility
ISFSI	Independent Spent Fuel Storage Installation
LTP	license termination plan
MARSAME	Multi-Agency Radiation Survey and Assessment of Materials and Equipment
NESHAP	National Emissions Standards for Hazardous Air Pollutants
NPDES	National Pollutant Discharge Elimination System
OCA	Owner Controlled Area
ODCM	Off-Site Dose Calculation Manual
OPPD	Omaha Public Power District
PCB	polychlorinated biphenyls
PCP	Process Control Program

QATR	Quality Assurance Topical Report
RA	radiological assessment
RASS	remedial action support survey
RV	Reactor Vessel
RVI	Reactor Vessel Internals
RWP	radiation work permit
SAF	Security Access Facility
SSC	structure, system, and component
URS	unconditional release survey

3 IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1 Introduction

In accordance with 10 CFR 50.82(a)(9)(ii)(B), the license termination plan (LTP) must identify the remaining major dismantlement and decontamination (D&D) activities for the decommissioning at the time of submittal. The information includes those areas and equipment that need further remediation and an assessment of the potential radiological conditions that may be encountered. Estimates of the occupational radiation dose accrued during the performance of the scheduled task and the projected volumes of radioactive waste that will be generated are also included. These activities will be undertaken pursuant to the current 10 CFR Part 50 license, are consistent with the Fort Calhoun Station (FCS) “Defueled Safety Analysis Report” [1] (DSAR), will not be inimical to the common defense and security or to the health and safety of the public pursuant to 10 CFR 50.82(a)(10), and do not depend upon LTP approval to proceed. Information that demonstrates that these activities will not have a significant effect on the quality of the environment is provided in LTP Chapter 8, “Supplement to the Environmental Report.”

This chapter was written following the guidance of NUREG-1700, “Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans” [2], and Regulatory Guide 1.179, “Standard Format and Content of License Termination Plans for Nuclear Power Reactors” [3], and will discuss those dismantlement activities as of July 27, 2021.

Omaha Public Power District’s (OPPD) primary goals are to decommission FCS safely and to maintain the continued safe storage of spent fuel in an Independent Spent Fuel Storage Installation (ISFSI). OPPD will decontaminate and dismantle FCS in accordance with the DECON alternative, as described in NUREG-0586, Supplement 1, Volume 1, “Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities” [4]. Completion of the DECON option is contingent upon continued access to one or more low level waste disposal sites. Currently, OPPD has access to the low-level waste disposal facilities located in Andrews, Texas and Clive, Utah.

Decommissioning activities at FCS will be conducted in accordance with the DSAR, the facility operating license (License No. DPR-40, Docket No. 50-285), all associated technical specifications, and the requirements of 10 CFR 50.82(a)(6) and (a)(7). Currently, the remaining activities do not involve any unreviewed safety questions or changes in the technical specifications for FCS. If an activity requires prior NRC approval under 10 CFR 50.59(c)(2), or a change to the technical specifications or license, a submittal will be made to the NRC for review and approval before implementing the activity in question.

Decommissioning activities are conducted in accordance with the existing OPPD Radiation Protection Program, Industrial Safety Program, Process Control Program (PCP), and Waste Management Program. These programs are well established and frequently inspected by the NRC. Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during operations, especially those during major maintenance and outage evolutions.

The remaining D&D activities that will be performed on structures, systems, and components (SSCs) are described in Section 3.5. These sections provide an overview of each structure that is earmarked for demolition, describe the major remaining components of potentially contaminated plant systems, and, as appropriate, a description of specific equipment remediation considerations. Table 3-1 contains a list of major SSCs that have been or are to be removed, as well as those slated to remain. Table 3-2 contains a list of general project milestones and the approximate date for completion.

3.2 Decommissioning Overview

On October 24, 2016, the FCS reactor was shut down for the final time, thus initiating the decommissioning process. In October 2018, the OPPD Board authorized the move toward decommissioning FCS in accordance with the DECON alternative. On April 1, 2019, OPPD entered into a partnership with Energy*Solutions*, and at that time, OPPD and its partner performed evaluations of major plant SSCs to determine what function, if any, these SSCs would be expected to perform during decommissioning. Each major plant SSC was evaluated to determine if the SSC (in its entirety or any portion thereof) was important for the safe storage of the spent fuel, was important for the monitoring and control of radiological hazards, or was needed to perform a function during the D&D of the plant.

OPPD and Energy*Solutions* administrative procedures specify the standard methods of accomplishing plant activities and processes. They are documents used to implement the requirements of NO-FC-10, “Quality Assurance Topical Report” [5] (QATR). The QATR ensures that FCS complies with the requirements of 10 CFR 50, Appendix B for quality assurance. Examples of administrative processes controlled by procedures include: ALARA (as low as is reasonably achievable) reviews, radiation protection (including airborne and contamination control), effluent and environmental monitoring, radioactive waste processing (including transportation and release requirements), safety programs, control of design basis (modification and work package procedures), and final status surveys (FSS).

A work control process is applied for decommissioning to document work performed and apply plant processes and controls to the activities. If a decommissioning activity requires NRC review pursuant to 10 CFR 50.59, an amendment to the license will be submitted.

3.3 End State at License Termination

The “End State” is defined as the configuration of the remaining below-ground basements, above-ground structures, buried piping, and open land areas at the time of license termination. Chapter 6 of the LTP provides a more detailed discussion of the FCS end state. Figure 3-1 provides the location of the remaining below-ground basements, above-grade buildings, and buried piping that will remain at the time of license termination.

The following above-grade buildings will remain at the time of license termination: Training Building, FLEX Building, Owner Controlled Area (OCA) Entrance Building, 3451 Old Building, 3451 New Building, and the 1251 Control and Switchgear Building. With the exception of the removal of the 345kv and 161kv lines to the station, the Switchyard will remain after decommissioning. All other structures will be removed to a minimum of three feet below grade (approximately 1,001 feet above mean sea level or AMSL). Because there are no dose model implications, if the number and identification of above-grade buildings to remain and be subject to FSS changes, the NRC will be notified.

The ISFSI and the ISFSI Operation Facility (IOF) will remain in the reduced 10 CFR Part 50 license and are not within the scope of this LTP.

The basements of the Turbine Building, Containment Building, Auxiliary Building, and Intake Structure will remain with all interior walls removed, with the exception of the Turbine Building where the turbine pedestal will remain up to three feet below grade. All other structures will be removed in their entirety with the exception of the six above-grade buildings and the Switchyard listed above. Once D&D activities are completed, as well as subsequent FSS, each basement will be backfilled with clean fill material to grade level (approximately 1,004 feet AMSL).

The end state will also include a range of buried pipe, embedded pipe, and penetrations. The inventory of buried pipe, embedded pipe, and penetrations to remain is provided in OPPD calculations and position paper (CPP) FC-21-002, "Description of Embedded Piping, Penetrations, and Buried Pipe to Remain in Fort Calhoun End State" [6].

As discussed in Section 3.4.2, OPPD added three rail lines to the existing rail spur to allow for the direct loading of waste into rail cars. The added rail lines will remain in place after license termination with the exception of the portion of the line that resides within the waste processing structure so that a viable means of final spent fuel transfer is available by rail when an approved permanent storage location is developed by the Department of Energy.

3.4 Completed and Ongoing Decommissioning Activities and Tasks

3.4.1 Independent Spent Fuel Storage Installation

The ISFSI was constructed on an elevated pad and consists of 42 heavily reinforced concrete horizontal storage modules (HSMs). The ISFSI is located in the protected area designated for dry storage of spent fuel and greater than Class C waste (GTCC). The ISFSI is licensed under Subpart K of 10 CFR Part 72. In 2006, and again in 2009, 10 custom-fabricated stainless-steel canisters were loaded with 320 spent fuel assemblies. In mid-October 2019, the dry cask project was continued with the loading of 944 spent fuel assemblies into 30 canisters with the final assembly being placed in a canister on May 8, 2020. The last spent fuel dry cask was secured in its storage module on May 13, 2020.

Two additional dry cask stainless-steel canisters will be loaded with GTCC once the reactor vessel segmentation is completed. These last two canisters are scheduled to be transported to the ISFSI in the first quarter of 2023.

3.4.2 Installation of Temporary Enclosures and Structures

In order to more safely and efficiently remove, load, and transport radioactive waste, as well as providing a means to reduce the potential for cross-contamination, two temporary enclosures, a dedicated haul road, and rail spurs have been or will be constructed.

The Containment Building equipment hatch opening will be enlarged to accommodate the removal of the steam generators, other large components, and to facilitate demolition of Containment Building lower elevation concrete. The opening will be 24 feet in width by 50 feet in height (approximate elevation 1,005 feet to 1,055 feet AMSL). Room 66 and the Diesel Generator portion of the Auxiliary Building were demolished and an 80 feet by 100 feet temporary enclosure will be installed in their place. A rail system with trolley and gantry crane will be installed within the temporary enclosure and extend approximately 13 feet into the Containment Building. Negative pressure ventilation will be maintained on the enclosure such that waste load out activities will not impact radiologically clean areas outside of the enclosure.

A dedicated haul road will be used to transfer radioactive waste from the Containment Building enclosure to another temporary enclosure located at the rail spur in the southern portion of the site. A rail spur from the Chicago and Northwestern (CNW) Railway was constructed to serve the original construction of the FCS. The original CNW tracks and rail spur have since been removed. In 1994, a permanent easement was granted to allow the construction of a new rail spur in the approximate location of the old CNW railway to allow trains to serve the Cargill industrial facility located northwest of FCS. In late 2020, OPPD added three lines (Lines A, B, and C approximately 1,360, 1,595, and 2,230 feet in length, respectively) to the rail spur to allow for the direct loading of waste into rail cars. In order to accommodate the expanded rail spur, large amounts of soil had to be removed from the adjacent hillside. Approximately 131,856 cubic yards of soil was removed and relocated to a nearby interim staging area. Radiological surveys, consisting of gamma scans and soil sampling, were performed on the soil prior to relocating to the interim storage area. Isolation and control measures have been instituted around each of the two soil piles and will remain in place until the soil is used as fill material in basements.

A 100 feet by 400 feet temporary waste processing structure has been erected at the rail spur to allow for the size reduction and loading of radioactive waste in a controlled environment. Negative pressure will be maintained in this enclosure such that waste load out activities will not impact radiologically clean areas outside of the enclosure.

3.4.3 Demolition and Dismantlement of Non-Radiological Structures

As of July 27, 2021, the following non-radiological structures have been demolished:

- Service Building
- Security Building
- Maintenance Shop
- Technical Support Center (slab will be removed at a later date)
- Chemistry and Radiation Protection Facility, or CARP (slab will be removed at a later date)
- New Warehouse
- Ballistic and Blast Rated Enclosures

The following are general decontamination and dismantlement considerations that have been incorporated, as appropriate, into the activities for the demolition of non-radiological structures.

- Prior to demolition, the non-radiological structures were removed from permanent power and placed on deconstruction power (cold and dark).
- An unconditional release survey (URS) was performed on each structure, prior to demolition, in accordance with site procedures that describe the protocols to be used to

design, perform, control, evaluate, and document radiological surveys performed on structures (or portions of structures) to support the unconditional release of demolition debris from the FCS site. These protocols are established using guidance from NUREG-1575, Supplement 1, "Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual" (MARSAME) [7], NRC IE Circular No. 81-07, "Control of Radioactively Contaminated Materials" [8], ISO 7503-1:2016, "Measurement of radioactivity - Measurement and evaluation of surface contamination - Part 1: General principles" [9], NRC HPPOS-071, "Control of Radioactively Contaminated Material" [10], NRC HPPOS-072, "Guide on 'How Hard You Have to Look' as Part of Radioactive Contamination Control Program" [11], NRC HPPOS-073, "Surveys of Wastes from Nuclear Reactor Facilities before Disposal" [12], and NRC Information Notice No. 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities" [13].

- Any structures, or portions of structures, where plant-related radioactivity was identified above background levels, were segregated and disposed of as radioactive waste.
- Remediation and disposal of identified Asbestos Containing Material (ACM), lead, polychlorinated biphenyls (PCB), and other hazardous waste was performed prior to demolition.
- The structures, including footers, foundations and underground utilities, were demolished and removed to a minimum of three-feet below grade (approximately 1,001 feet AMSL). The buildings were demolished using standard construction equipment and techniques, which included dust suppression (by wetting down the SSCs during demolition) and use of stringent access control measures for the work area.
- A radiological assessment (RA), consisting of gamma scans and soil sampling, was performed on the excavated footprint prior to backfilling.
- The excavated footprint was backfilled to the established final grade (approximately 1,004 feet AMSL) using fill material in accordance with the "Fort Calhoun Excavation and Backfill Requirements" [14].
- All groundwater intrusion to buildings and excavations were managed in accordance with the FCS "National Pollutant Discharge Elimination System (NPDES) permit" [15] (NPDES permit no. NE0000418) and CH-ODCM-0001, "Offsite Dose Calculation Manual" (ODCM) [16].
- All debris, including concrete, generated from the demolition of non-radiological structures was disposed of at an appropriate off-site landfill or recycling facility. No concrete generated from the demolition of non-radiological structures will be used as fill material in excavations or basements (i.e., all concrete debris will be removed from site and be disposed of appropriately).

3.4.4 Demolition of Room 66 and Diesel Generator portion of Auxiliary Building

As discussed in Section 3.4.2, Room 66 and the Diesel Generator Building were demolished and an 80 feet by 100 feet temporary enclosure will be installed in their place. The following are general decontamination and dismantlement considerations that have been incorporated into the activities for the demolition of Room 66 and the Diesel Generator portion of Auxiliary Building:

- Prior to demolition, the non-radiological structures were removed from permanent power and placed on deconstruction power.
- Remediation and disposal of identified ACM, lead, PCBs, and other hazardous waste was performed prior to demolition.
- A radiological survey was performed on the structures to prior to demolition. Areas were remediated as necessary or mitigated, and the horizontal surfaces in Room 66 were locked down.
- The structures were demolished to elevations 999 feet AMSL and 1,001 feet AMSL in preparation for a concrete pad to support the rail system and temporary enclosure.
- All buried utilities, tanks and piping within the footprint of the temporary enclosure were removed.
- An RA, consisting of gamma scans and soil sampling, was performed on the excavated footprint (including trenches where utilities and storm drains were removed) prior to backfilling.
- All debris generated from the demolition of Room 66 and the Diesel Generator portion of Auxiliary Building will be disposed of as radioactive waste at a facility in Clive, Utah.

3.4.5 Buried Utilities, Tanks and Piping Removal

Prior to the construction of the Containment Building rail system and temporary enclosure, the following buried utilities, tanks and piping within the footprint of the enclosure were removed:

- Diesel Fuel Tank, FO-1 and associated piping
- Section of the fire protection ring header
- Electrical duct banks
- Section of the storm drain system

Diesel Fuel Tank, FO-1 was emptied of its contents but had not been cleaned. Any tank contents that inadvertently came into contact with soil were cleaned and dispositioned appropriately.

An RA, consisting of gamma scans and soil sampling, was performed on the excavated footprints prior to backfilling.

The excavated footprints were backfilled to the established final grade (approximately 1,004 feet AMSL) using fill material in accordance with the "Fort Calhoun Excavation and Backfill Requirements."

3.5 Future Decommissioning Activities and Tasks

3.5.1 Overview

The remaining D&D activities can be grouped into several classifications, the implementation of which may overlap. The first phase includes the ongoing removal of non-radiological buildings within the deconstruction area (DA) in order to make room for the demolition of the radiologically contaminated buildings. The Turbine Building demolition is part of this phase of work. The current phase also includes the removal of contaminated and non-contaminated systems, decontamination of site buildings, and site remediation. This phase may be implemented on an area-by-area basis. Under this approach, often only a portion of a system will be removed, while the remaining portions await removal in a subsequent phase. The remaining contaminated systems and components will be decontaminated or removed, packaged, and shipped directly to a low-level radioactive waste disposal facility. In some instances, the remaining systems or components may be removed during building demolition as long as they contain residual contamination below the established open-air demolition criteria or adequate contamination control measures have been emplaced.

The next phase of work includes reactor vessel segmentation and reactor internals removal. The removal of some radiological buildings in the DA (e.g., Radwaste Building) will occur earlier in the project. Buried piping will be removed within the DA and outside the DA as it becomes available for excavation. The removal of non-radiological buildings outside the DA will then occur, along with the removal of the remaining radiological buildings within the DA (Auxiliary Building and Containment Buildings). Lastly, the temporary waste processing structure, along with the portion of the rail spur located within the structure footprint, will be removed. The portion of the tracks outside of the structure footprint will remain.

Following the removal or decontamination of contaminated SSCs, a comprehensive final radiological survey will be completed as described in Chapter 5 of this LTP. This survey will verify that residual radioactivity has been reduced to sufficiently low levels, as stipulated in 10 CFR 20.1402, to allow the release of the FCS site for unrestricted use. Upon completion of the FSS for a site area, OPPD will document the results of that FSS and will make them available for NRC inspection.

Detailed schedules for significant decommissioning activities (including FSS activities and building demolition) are routinely communicated to NRC headquarters and NRC Region IV personnel (for example, through periodic decommissioning status meetings) to allow for NRC observation and inspection of these activities. Communication of significant decommissioning activities will be based on activities listed in detailed project projection schedules (or their equivalents).

Decommissioning cost estimates are provided in Chapter 7 of this LTP.

The following sections provide a general description of the remaining decommissioning activities for the FCS site. This information provides the basis for development of programs and procedures for ensuring safe decommissioning and a basis for detailed planning and preparation of D&D activities.

3.5.1.1 Detailed Planning and Engineering Activities

Detailed project plans will continue to be developed in accordance with design control procedures to support the D&D activities. These plans are used to develop work packages, to assess safety and waste issues for each major task, to support ALARA reviews, to aid in estimating labor and resource requirements, and to track decommissioning costs and schedule.

Work packages are used to implement the detailed plans and provide instructions for actual field implementation. The work packages address discrete units of work and include appropriate hold and inspection points. Administrative procedures control work package format and content, as well as the review and approval process.

3.5.1.2 General Decontamination and Dismantlement Considerations

As has been the current practice and in accordance with the FCS DSAR, the following general D&D considerations, as applicable, will continue to be incorporated into decommissioning work packages during the D&D period. Demolition techniques will vary for each building and will depend on building construction, contamination control methods, and end state. It is assumed that all non-radiological hazardous materials will be removed prior to demolition, as appropriate.

Radiological surveys will verify that: (1) residual radioactivity has been reduced to levels that are suitable for open-air demolition; (2) residual radioactivity levels are appropriate for shipment to an off-site licensed radioactive materials handler for processing or disposal; (3) there is no detectable radioactivity above background, and SSCs may be free released; or (4) residual radioactivity has been reduced to sufficiently low levels to release buildings and associated areas for unrestricted use upon their removal from the Part 50 license.

Dismantlement activities are reviewed to ensure they do not impact the safe storage of spent fuel in the ISFSI. Configuration control packages are implemented in accordance with administrative controls and require evaluations in accordance with the requirements of 10 CFR 50.59.

3.5.1.3 General Backfill Considerations

For all excavations created by the removal of buried piping, components or slab-on-grade structures, an RA will be performed prior to backfill. For all sub-grade basements that will remain, a FSS will be performed and, contingent upon the completion of confirmatory surveys and regulatory approval, the basements will be backfilled. If a major sub-grade building has been removed in its entirety, an FSS will be performed on the resultant excavation (using the most-restrictive classification of the building prior to demolition) prior to backfill of the excavation. All void spaces will be backfilled using clean fill material to grade (1,004’ AMSL) in accordance with the “Fort Calhoun Excavation and Backfill Requirements.”

3.5.1.4 Proposed Measures to Ensure that Areas are not Re-contaminated

The capability to isolate or to mitigate the consequences of radioactive release will continue to be maintained during D&D activities. Isolation is the closure or control of penetrations and openings to restrict transport of radioactivity to the environment. However, this consideration does not preclude the removal of penetrations and attachments to the Containment Building or Auxiliary Building.

Work activities are planned to minimize the spread of contamination. Contaminated liquids are contained within existing or supplemental barriers and may be processed by a liquid waste processing system prior to release, if necessary. To minimize the potential for spread of contamination, the following considerations will continue to be evaluated for incorporation into the planning of decommissioning work activities:

- covering of openings in internally contaminated components and systems to confine internal contamination
- use of contamination control barriers, as appropriate, around activities that may result in airborne contamination during cutting and removal processes
- decontamination and dismantlement of contaminated SSCs by decontamination in place, removal and decontamination, or removal and disposal
- removal of contaminated supports in conjunction with equipment removal or decontamination of supports in conjunction with building decontamination
- removal of contaminated systems and components from areas and buildings prior to structural decontamination
- removal or decontamination of embedded contaminated piping, conduit, ducts, plates, channels, anchors, sumps, and sleeves during area and building structural decontamination activities
- use of localized or central processing and cutting stations to facilitate packaging of components removed ion large pieces
- removal of small or compact plant components and parts intact, where feasible, including valves, smaller pumps, some small tanks, and heat exchangers. These components could then be decontaminated in whole or part and disassembled or segmented in preparation for disposal or release.

As described in Section 3.4.2, a temporary waste loadout enclosure will be installed at the Containment Building equipment hatch, and another temporary waste processing enclosure will be installed at the rail spur. These enclosures will provide additional assurance that areas will not be re-contaminated. Routine radiological surveys performed within and adjacent to these enclosures, as well along the haul road route in between the enclosures, will verify the controls are being maintained.

3.5.1.5 Decontamination Methods

Contaminated systems and components are typically removed and sent to a low-level radioactive waste disposal facility or decontaminated on-site and released. Decontamination methods typically include wiping, washing, vacuuming, scabbling, spalling, hydrolasing, and abrasive blasting. Selection of the preferred method is based on the specific situation. Other decontamination technologies may be considered and used as appropriate.

Hand wiping may be used to remove loose surface contamination. Airborne contamination control and waste processing systems are used as necessary to control and monitor releases. If structural surfaces are washed to remove contamination, controls are implemented in accordance with work packages, the ODCM and approved procedures to ensure that radioactive liquid waste is collected for processing by liquid waste processing systems.

Tanks and vessels are evaluated and, if ALARA, are flushed or cleaned to reduce contamination and to remove sediments prior to sectioning or removal. In cases where tanks, vessels, and piping are major contributors of dose rates, the ALARA principle will be applied to determine the removal sequence. Precautions are taken to ensure that liquid that is inadvertently discharged from the tank is captured for processing by a liquid waste processing system. Sediment removed from the tank may be stabilized prior to shipment or may be shipped without stabilization in an approved condition. Wastewater is processed and/or sampled and analyzed in accordance with the NPDES permit (for non-radiological liquid waste) and ODCM (for radioactive liquid waste) before being discharged.

Embedded piping will be cleaned using hydrolasing technology. Radiological surveys will verify the successful remediation and cleanout of the embedded piping. Fiber optics will be used in support of the radiological surveys. Wastewater from hydrolasing activities will be captured in sumps or low spots and will be processed in accordance with the NPDES permit and ODCM.

Concrete that has surface or near-surface contamination may be cleaned, if necessary, to meet applicable derived concentration guideline levels (DCGLs) and ALARA. This is typically achieved by wiping or scabbling. Activated concrete, or concrete with deeper contamination, may be removed as necessary to meet DCGLs (and ALARA considerations) and may be sent to a low-level waste disposal facility or handled by other methods that control the removal depth to minimize the waste volume produced. In some instances, the use of large equipment and hoe rams may be warranted. Vacuum removal of the dust and debris with HEPA filtration of the effluent may be used to minimize the spread of contamination and reliance on respiratory protection measures. Chapter 4 of the LTP provides additional discussion on remediation methods.

3.5.1.6 Contaminated System Dismantlement

Dismantlement methods for contaminated systems can be divided into two basic types: disassembly and cutting. Disassembly generally means removing fasteners and components in an orderly, non-destructive manner (i.e., the reverse order of the original assembly). Cutting methods include flame cutting, abrasive cutting, and cold cutting.

Flame cutting includes the use of oxyacetylene and other gas torches, carbon arc torches, air or oxy arc torches, plasma arc torches, cutting electrodes, or combinations of these. Most of the torches can either be handheld or operated remotely. Abrasive cutting includes the use of grinders, abrasive saw blades, most wire saws, water lasers, grit blast, and other techniques that wear away metal. Cold cutting includes the use of bandsaws, blade saws, mechanical disintegration methods, drilling, machining, shears, and bolt/pipe/tubing cutters. Selection of the preferred method depends on the specific situation. Other dismantlement technologies may be considered and used if appropriate.

Dismantling of systems includes the removal of valves and piping for disposal. Most valves can be removed with the piping. Larger valves and valves with actuators may be removed separately for handling and packaging purposes.

3.5.1.7 Removal Sequence and Material Handling

Removal sequences may be dictated by access and material handling requirements or by personnel exposure considerations. In some cases, a top-down approach may be used. Using this approach, materials and structures at the highest elevations are removed first to allow access to components in lower levels. In other cases, different approaches may prove more efficient.

In some cases, the first items removed are those that are not contaminated, or are only slightly contaminated, to preclude contamination by other equipment removal activities. However, personnel exposure considerations may not always allow this option. The ALARA principle will be applied by removal of hot spot items (e.g., piping with high dose rates) prior to other work. Where non-contaminated equipment or piping is not removed first, covers or other protection methods may be used to prevent cross-contamination.

Where rapid cutting techniques are available, pipes and equipment can be sectioned into pieces that are manageable using light rigging or by manual lifting. Where slow cutting techniques are used, the largest manageable pieces will typically be freed and further reduced in size at a more convenient location or a location with lower dose rates.

The FCS site is equipped with multiple cranes, hoists, and lifting and transport systems. These systems can be used to lift and transport components and equipment to support plant decommissioning activities. Forklifts, mobile cranes, front-end loaders, and other lifting and transport devices can also be used for plant decommissioning activities.

Inspection requirements for these cranes meet the specific requirements of plant procedures. Other rigging equipment will be inspected and verified in accordance with procedures. Smaller rigging equipment (e.g., chain hoists, straps, and slings) will be inspected and verified to be in good working condition prior to use.

Installed cranes and hoists may be used in conjunction with temporary or mobile lifting and transport devices to support decommissioning. The installed plant cranes, hoists, and other lifting devices may be dismantled when they no longer are required to support decommissioning activities.

3.5.1.8 System Isolation/De-Energization

Systems and components will continue to be deactivated prior to decontamination and dismantlement. In general, isolation/de-energization is implemented by mechanical isolation of interfaces with operating plant systems, draining piping/components, and de-energizing electrical supplies. Combustible materials (e.g., charcoal from filters, lube oil) are removed from the deactivated components, where practical. Chemicals used in, or resulting from, decommissioning activities are controlled in accordance with the plant chemical safety program.

Isolation/de-energization of plant systems is controlled by approved procedures. Project plans and plant work orders have been or are being written for system removal, including the removal of temporary power. The design change process is used to remove components, lift electrical leads, install electrical jumpers, cut and cap piping systems, or install blank flanges.

3.5.1.9 Temporary Systems Required to Support Decommissioning

Decontamination and dismantlement of SSCs often require the removal of interferences. Removal of some of these interferences may eliminate power, service air, and other services needed to support decommissioning. Also, use of installed plant systems for decommissioning support may become impractical, due to the risk of encountering energized systems or circuits. Temporary services and systems are being provided by portable air compressors using hoses or temporary air manifolds. Temporary modifications to plant SSCs are controlled by design control procedures.

Portable load centers are powered from motor control centers, plant load centers, or the yard loop. These portable load centers can supply cutting and hoisting equipment, temporary lighting, or other power needs. Service air can be provided by portable air compressors using hoses or temporary air manifolds. Demineralized water is available from portable demineralizer skids or portable tankers brought on-site. Portable hydraulic power centers can be used to power hydraulic equipment.

Temporary liquid and solid waste processing systems are being used during decommissioning for processing plant waste. These systems include filters and/or demineralizers, and may be used at one or more locations in the waste processing path.

Portable radiation monitors and air monitoring equipment provide local radiation monitoring. Localized temporary ventilation equipment and HEPA filtration is being used to supplement building ventilation and minimize the spread of radioactive particulate contamination.

3.5.2 Specific Radiological Decommissioning and Dismantlement Activities

This section presents a summary description of the specific D&D activities for the remaining SSCs at FCS as of July 27, 2021.

3.5.2.1 Reactor Vessel Internal Segmentation

The segmentation effort required to prepare the Reactor Vessel Internals (RVI) for packaging will be performed under water. This will help lower dose rates and aid in contamination control. Mechanical segmentation technologies will be applied which includes use of rotating saw blades with a surface pattern that generates “easy to collect” shaped chips (i.e., there is no coiling). The RVI component will be placed on a volume reduction station turntable where it will be cut to the appropriate size to fit into a basket. The equipment that will be used to perform the RVI segmentation is based upon the design of the equipment used for the Zion Nuclear Power Station and San Onofre Nuclear Generating Station decommissioning projects and incorporates lessons learned from these projects.

A water filtration and chip collection system will be installed to keep dose rates low and visibility high during the project. Segmentation of each of the RVI will be done in accordance with a predetermined segmentation plan designed to maximize the packing factor in the designed containers.

The RVI segmentation process will generate three classifications of radioactive waste that will require different shipping containers for the different final resting places. Class A waste will be loaded primarily into 8-120A casks, gondola rail cars, or articulating bulk cars and shipped to Clive, Utah. Class B/C waste will be loaded into 3-60B casks and shipped to Andrews County, Texas. GTCC waste will be loaded into TN liners/casks and will remain on-site at the ISFSI.

3.5.2.2 Reactor Vessel Segmentation

In preparation of the Reactor Vessel (RV) segmentation, the reactor cavity sealing surface around the RV will be removed to make the gap between the bio shield and RV accessible.

The free-standing thermal insulation will be removed and disposed of as required. The RV nozzles will be cut and capped after the inside of the RV is cleaned and dewatered. Measures to cut and cap in-core instrumentation penetrations under the RV will be taken. If additional obstacles or recesses are present in the gap, they will also be removed. At this point, the vertical cuts on the RV flange section will be performed.

Torch cutting equipment will be installed in the gap between the RV and bio shield, supported by a frame capable of turning on its axis as an adaptation to the shielding plate.

Tenting of the cutting area will be used to add a layer of contamination control, as required. The packaging of segments can be performed in the reactor cavity deep end or on the operating floor, as appropriate, for the exposure rates associated with the segments. Once the RV has been cut and packaged except for the hemispherical bottom head (HBH), the inner part of the shielding plate will be removed and the same torch cutting equipment will be used to cut the RV support skirt after the HBH is attached to the polar crane. The HBH can either be disposed of in one piece or positioned in the refueling pool for manual separation into halves. If dose rates disallow for manual segmentation, a torch guide rack can perform the separation cut while the support skirt provides stability.

The Class A waste from the RV segmentation will be loaded into shield boxes and shipped via truck or rail to Clive, Utah.

3.5.2.3 Containment Building

Inspections were completed to identify any remaining waste streams in the Containment Building. Surveys were performed in accordance with the National Emissions Standards for Hazardous Air Pollutants (NESHAP) to identify any potential ACM, hazardous waste, and PCB waste. Currently, all accessible friable ACM, PCB waste, or other hazardous waste is being removed.

As discussed in Section 3.4.2, the equipment hatch opening will be enlarged to accommodate the removal of the steam generators, other large components, and to facilitate demolition of lower elevation concrete. The opening will be 24 feet in width by 50 feet in height (approximate elevation 1,005 feet to 1,055 feet AMSL). A rail system will be installed and consist of a self-powered flat horizontal moveable platform with a capacity of 203,000 lbs. The overall rail system will be approximately 45 feet long with approximately 13 feet of the rail extending into the Containment Building. The 13-foot section of the rail will enter the hatch at the 1,005 feet AMSL elevation.

The two steam generators will be removed intact via use of the polar crane (130-ton capacity). A special crane may be used if it determined the polar crane does not have the vertical lift necessary to tilt the steam generators while intact. The steam generators will be placed inside the temporary enclosure that is attached to the equipment hatch. The steam generators will be prepared for disposition by performing decontamination and applying a non-permeable coating. Once a release survey has been completed, the steam generators will be removed from the enclosure. Utilizing two top hatches of the enclosure and a crane, the prepped steam generators will be placed on saddles. A Goldhofer trailer will be backed into the enclosure and the steam generators will be placed onto the Goldhofer.

The remaining interior SSCs will be removed, including the reactor head, pressurizer, reactor coolant pumps, reactor coolant tank, safety injection tanks, and other miscellaneous equipment. The reactor head will be segmented and packaged to meet the Waste Acceptance Criteria limits. Some interior SSCs may remain in place for structural demolition such as the containment spray ring header. All piping and sleeves will be removed from penetrations.

All interior concrete will be removed down to the stainless-steel liner, thus removing all embedded piping in the process.

Once the remaining stainless-steel structure located below 3 feet below grade (1,001 feet AMSL) has satisfactorily undergone FSS and compliance with the unrestricted release criteria has been demonstrated, contingent upon the completion of confirmatory surveys and regulatory approval, the Containment Building void will be backfilled using clean fill material to 1,001 feet AMSL. An impermeable membrane will then be installed on top of the clean fill. The material will be thick enough to protect the soils underneath it from contamination. Three feet of sacrificial fill will then be placed on top of the membrane to approximately the 1,004 feet AMSL elevation.

Surface decontamination or application of a fixative will then be performed to support the performance of a Contamination Verification Survey (CVS), which is performed to verify the as-left radiological conditions inside the Containment Building (above the 1,004 feet AMSL elevation), as well as the temporary enclosure and pad, are suitable for open-air demolition.

The rail system will be removed, and the temporary enclosure will be demolished and disposed of as radioactive waste. An RA, consisting of gamma scans and soil sampling, will be performed on the excavated footprint prior to backfilling.

Prior to demolition of the Containment Building exterior shell, a URS will be performed on the exterior concrete surfaces to verify the concrete may be unconditionally released from site as non-radioactive waste. If detectable plant-related radioactivity is identified during the URS of the Containment Building exterior shell, then that concrete will be controlled as radioactive material and disposed of appropriately.

Removal of the pre-stressing tendons has already been accomplished with the exception of four dome tendons. Prior to demolition, the remaining four dome tendons will be removed, along with the remaining galbestos panels which were unable to be removed previously because of plant interferences. The tendons, as well the tendon grease, will be disposed of as non-radioactive waste.

The gradual demolition of the containment shell will then be initiated from grade, using ram-hoes to chip away the concrete along the bottom circumference of the shell and allowing the weight of the remaining structure to slowly demolish the structure to grade. The shell will be demolished to 3 feet below grade or the 1,001 feet AMSL elevation. The sacrificial fill and membrane will be removed and packaged as radioactive waste.

After demolition is complete, an RA, consisting of gamma scans and soil sampling, will be performed on the excavated footprint prior to backfilling.

3.5.2.4 Auxiliary Building

Inspections were completed to identify any remaining waste streams in the Auxiliary Building. Surveys were performed in accordance with the NESHAP to identify any potential ACM, hazardous waste, and PCB waste. All accessible friable ACM, PCB waste, or other hazardous waste is being or will be removed.

Embedded piping in the floors of the 989 feet and 971 feet AMSL basement elevations will be hydrolased and then undergo a radiological survey to verify the as-left contamination levels are below the established DCGLs. The openings to all embedded piping will be temporarily sealed to prevent re-contamination by use of mechanical plugs and grout.

A CVS will be performed on the interior and exterior surfaces of SSCs prior to demolition. Fixative will be applied to internals of systems and components containing potential contamination in excess of the criteria established for open-air demolition. Surgical demolition will be performed on any SSCs that do not meet the open-air demolition criteria.

The Auxiliary Building will be demolished from the top down to the floor slab at the 989' and 971' elevations, and exterior walls at the 1,001 feet AMSL elevation, removing all remaining interior SSCs in the process. The embedded piping will then be un-sealed.

After the Auxiliary Building basement has satisfactorily undergone FSS (including embedded piping) and compliance with the unrestricted release criteria has been demonstrated, contingent upon the completion of confirmatory surveys and regulatory approval, the Auxiliary Building void will be backfilled using clean fill material to the 1,004 feet AMSL elevation (grade).

3.5.2.5 Radwaste Processing Building

Inspections were completed to identify any remaining waste streams in the Radwaste Processing Building. Surveys were performed in accordance with the NESHAP to identify any potential ACM, hazardous waste, and PCB waste. All accessible friable ACM, PCB waste, or other hazardous waste is being or will be removed.

A CVS will be performed on the interior and exterior surfaces of SSCs prior to demolition. Fixative will be applied to internals of systems and components containing potential contamination in excess of the criteria established for open-air demolition. Surgical demolition will be performed on any SSCs that do not meet the open-air demolition criteria.

In Room 505, there is a manhole that provides access to an electrical vault that sits beneath the slab-on-grade. Prior to demolition of the above-grade portion of the Radwaste Processing

Building, the electrical vault will be filled with gravel and temporarily sealed to prevent potential contamination from entering during demolition.

The Radwaste Processing Building will be demolished from the top down to the floor slab at the 1,004 feet AMSL elevation, removing all remaining interior SSCs in the process, but leaving the electrical vault in Room 505. The slab-at-grade will remain to be used to support the demolition of the Auxiliary Building.

After the above-grade portion of demolition has been completed, the temporary seal and gravel will be removed from the electrical vault in Room 505 and an FSS will be performed. Once the electrical vault has satisfactorily undergone FSS and compliance with the unrestricted release criteria has been demonstrated, contingent upon the completion of confirmatory surveys and regulatory approval, the electrical vault void will be backfilled using clean fill material to grade (1,004 feet AMSL).

Following the demolition of the Auxiliary Building, the Radwaste Processing Building slab-on-grade will be demolished along with embedded drains, footings or other structural features, underground drain piping, and any other utilities to 3 feet below grade.

After demolition is complete, an RA, consisting of gamma scans and soil sampling, will be performed on the excavated footprint prior to backfilling.

3.5.2.6 Circulating Water Tunnels, Discharge Piping, and Liquid Waste Discharge Line

The north and south discharge lines will have stop logs in place to prevent flow to the Missouri River. A more robust reinforcement may be necessary to support grouting activities. The remaining water will be pumped down to the permitted discharge point in accordance with the NPDES permit and the ODCM. Starting from the former Turbine Building footprint, the circulating water discharge piping and the liquid waste discharge line will be removed along with all footings, foundations, or other structural features which may extend to 3 feet below grade. The waste will be treated as radioactive unless proven otherwise.

After cleaning out any silt and debris remaining in the Circulating Water Tunnels, an FSS will be performed within the tunnels. Once the tunnels have satisfactorily undergone FSS and compliance with the unrestricted release criteria has been demonstrated, contingent upon the completion of confirmatory surveys and regulatory approval, the Circulating Water Tunnels void will be grouted using flowable fill.

3.5.2.7 Sewage Lagoon Piping and Lift Stations

All sanitary sewage buried piping and associated lift stations will be removed and disposed of as radioactive waste. An RA, consisting of gamma scans and soil sampling, will be performed on the excavated trenches prior to backfilling.

The sewage lagoons may need to be dewatered prior to excavating and removing contaminated soils. Remedial action support surveys (RASSs), consisting of gamma scans and soil sampling, will determine if additional soil beneath the liner must be removed. The excavated soils will have to be dried before being disposed of as radioactive waste. Safety measures will be taken (e.g., treating the sewage with lime) to mitigate the hazardous biological component of this waste.

After excavation is complete, an RA, consisting of gamma scans and soil sampling, will be performed on the excavated footprint prior to backfilling. The void will be backfilled using clean fill material to grade (1,004' AMSL) in accordance with the "Fort Calhoun Excavation and Backfill Requirements."

3.5.3 Specific Non-Radiological Decommissioning and Dismantlement Activities

3.5.3.1 Turbine Building

Initial component removal will include the dismantlement and removal of most of the large components including the turbines, generator, moisture separator reheaters, feedwater heaters, and coolers. The Turbine Building gantry crane, identified as HE-3, is powered by deconstruction power and will be utilized for large component removal.

In parallel with this effort, the URS of materials, equipment, and structural surfaces throughout the building will be performed. All systems, materials, and portions of the structure that are or will be identified by URS as contaminated with detectable plant-derived radioactive material will be removed and dispositioned and properly disposed of as radioactive waste. The remaining structure will then be made "Cold, Dark, and Dry" and turned over to a Demolition Contractor as a non-radiologically controlled structure for demolition as a contracted work scope.

In addition, inspections were completed to identify any remaining waste streams. Surveys were performed in accordance with the NESHAP to identify any potential ACM, hazardous waste, and PCB waste. All accessible friable ACM, PCB waste, or other hazardous waste will be removed prior to demolition.

All remaining commodities, equipment, and components will be removed, and the structure will be demolished to a depth of 3' below grade. All floors and walls that are 3 feet below grade will be removed with the exception of the turbine pedestals. Embedded equipment drains within the basement floor will be removed. Because of concrete fracturing concerns, the deeper embedded floor drains will be cleaned out and remain. All remaining piping embedded in concrete, as well as penetrations, will undergo FSS to demonstrate compliance with the unrestricted release criteria prior to being isolated, abandoned in place, and filled with grout or fill, as appropriate.

Once the remaining concrete structure located below 3 feet below grade (1,001 feet AMSL) has satisfactorily undergone FSS and compliance with the unrestricted release criteria has been demonstrated, contingent upon the completion of confirmatory surveys and regulatory approval, the Turbine Building void will be backfilled using clean fill material to grade (1,004 feet AMSL) in accordance with the "Fort Calhoun Excavation and Backfill Requirements."

3.5.3.2 Security Access Facility

Inspections were completed to identify any remaining waste streams in the Security Access Facility (SAF). Surveys were performed in accordance with the NESHAP to identify any potential ACM, hazardous waste, and PCB waste. All accessible friable ACM, PCB waste, or other hazardous waste is being or will be removed.

A URS will be performed of materials, equipment, and structural surfaces throughout the SAF. All systems, materials, and portions of the structure that are or will be identified by URS as contaminated with detectable plant-derived radioactive material will be removed and dispositioned and properly disposed of as radioactive waste.

All remaining commodities, equipment, and components will be removed and the structure will be demolished to a depth of 3 feet below grade.

After demolition is complete, an RA, consisting of gamma scans and soil sampling, will be performed on the excavated footprint prior to backfilling.

3.5.3.3 Mausoleum

Inspections were completed to identify any remaining waste streams in the Mausoleum. Surveys were performed in accordance with the NESHAP to identify any potential ACM, hazardous waste and PCB waste. All accessible friable ACM, PCB waste, or other hazardous waste is being or will be removed.

An URS will be performed of the walls, ceiling, and roof portions of the structure. The floor slab, sump, and structure to 3 feet below grade will be dispositioned as radioactive waste. All systems, materials, and portions of the structure that are or will be identified by URS as contaminated with detectable plant-derived radioactive material will be removed and dispositioned and properly disposed of as radioactive waste.

All remaining commodities, equipment, and components will be removed and the structure will be demolished to a depth of 3 feet below grade.

After demolition is complete, an RA, consisting of gamma scans and soil sampling, will be performed on the excavated footprint prior to backfilling.

3.5.3.4 Intake Structure

Inspections were completed to identify any remaining waste streams in the Intake Structure. Surveys were performed in accordance with the NESHAP to identify any potential ACM, hazardous waste, and PCB waste. All accessible friable ACM, PCB waste, or other hazardous waste is being or will be removed.

A URS will be performed of materials, equipment, and structural surfaces throughout the Intake Structure. All systems, materials, and portions of the structure that are or will be identified by URS as contaminated with detectable plant-derived radioactive material will be removed and dispositioned and properly disposed of as radioactive waste.

The sluice gates are closed, as are other Intake Structure riverside penetrations. The piping that penetrates the electrical Intake Structure walls have been air gapped and sealed. Stop logs will be used for the discharge lines. A more robust reinforcement may be necessary to support grouting activities. Bulkheads are to be installed as close as is reasonably possible to the river end of the Intake Structure penetrations. In locations where the intake tunnel is less than 3 feet below grade, portions of the structure will be removed to 3 feet below grade in order to meet the end state conditions. Intake and discharge tunnels will be filled with flowable fill and capped after the FSS and subsequent confirmatory surveys have been successfully completed.

All remaining commodities, equipment, and components will be removed and the structure will be demolished to a depth of 3 feet below grade. All embedded piping will be removed from the slab. The circulating water intake line will be removed and the remaining opening will be capped. Once the remaining concrete structure located below 3' below grade has satisfactorily undergone FSS and compliance with the unrestricted release criteria has been demonstrated, contingent upon the completion of confirmatory surveys and regulatory approval, the Intake Structure void will be backfilled using flowable fill to 3 feet below grade. Approximately 3 feet of rip-rap will be placed on top of the fill material to match the river bank contour.

3.5.3.5 Miscellaneous Non-Radiological Structures

The same general process will be applied for the demolition of miscellaneous non-radiological structures (e.g., New Maintenance Storage Shed, Neutralization (Tertiary) Building and Basin, and Storage Shed).

Inspections were completed to identify any remaining waste streams in the miscellaneous structures. All accessible friable ACM, PCB waste, or other hazardous waste is being or will be removed.

A URS will be performed of materials, equipment, and structural surfaces throughout the miscellaneous structures. All systems, materials, and portions of the structure that are or will be identified by URS as contaminated with detectable plant-derived radioactive material will be removed and dispositioned and properly disposed of as radioactive waste.

All remaining commodities, equipment, and components will be removed and the structures will be demolished to a depth of 3 feet below grade.

After demolition is complete, an RA, consisting of gamma scans and soil sampling, will be performed on the excavated footprint prior to backfilling.

3.5.3.6 Waste Processing Enclosure, Haul Roads, and Rail Spur

The waste processing enclosure, haul roads, and portions of the rail spur will be the last features to be removed from site. The haul roads will be removed working from the DA to the west, ending with the removal of the waste processing enclosure, the pad within the enclosure, and a portion of the rail spur that resided in the enclosure.

The haul roads will be 30 feet at their widest and 20 feet at their most narrow crossing and will consist of 16 inches of crushed stone along with two layers of geogrid. The majority of the haul roads are assumed to be radiologically clean and, after radiological surveys verify no plant-related radioactivity, will be dispositioned as clean waste.

The rail spur consists of Tracks A, B, and C. The rail spur in its entirety consists of approximately 2,000 standard timber ties (creosote), 402 wooden switches (up to 17 inches), 80 concrete ties (10 inches) and 136# sized railing. The tracks are built upon 12 inches of sub ballast and 8 inches of ballast designed with a 12-inch shoulder to each side with a 3:1 slope. Tracks A and B are located outside of the waste processing enclosure footprint and will be used to stage empty rail cars. Track C will be the loading track and approximately 400 feet of Track C resides within the waste processing enclosure. Inside the enclosure is a 6-inch asphalt floor on top of a 10-inch layer of crushed stone.

Tracks A and B and the portions of Track C that did not reside within the temporary waste processing enclosure will remain on-site at the time of license termination and will be subject to FSS. The portion of Track C that resides within the waste processing enclosure will be removed and disposed as radioactive waste. The enclosure and the pad will be demolished and disposed as radioactive waste.

After demolition is complete, an RA, consisting of gamma scans and soil sampling, will be performed on the excavated footprint prior to backfilling.

3.5.3.7 Soil Remediation

There is limited potential for contaminated surface or subsurface soil to be present at FCS based on the findings of the "Historical Site Assessment for Fort Calhoun Station" (HSA) [17] and the results of extensive characterization performed in 2020. The results of the open land characterization surveys are detailed in FC-20-2012, "Fort Calhoun Station Decommissioning Project Radiological Characterization Report" [18] and summarized in Chapter 2 of this LTP. According to the HSA, there have been low levels of Sr-90 periodically identified by the groundwater monitoring program at FCS, and the levels will continue to be monitored.

3.6 Radiological Impacts of Decommissioning Activities

The decommissioning activities described are and will be conducted under the provisions of the OPPD Radiation Protection Program and Radioactive Waste Management Program. These programs are and will continue to be implemented as described in the DSAR. The OPPD Radiation Protection Program and written site procedures are intended to provide sufficient information to demonstrate that decommissioning activities will be performed in accordance with 10 CFR 20, "Standards for Protection Against Radiation" and to maintain radiation exposures ALARA. The OPPD Radioactive Waste Management Program controls the generation, characterization, processing, handling, shipping, and disposal of radioactive waste in accordance with the approved OPPD Radiation Protection Program, Process Control Program, and written plant procedures.

The current Radiation Protection Program, Waste Management Program, and ODCM will be used to protect the workers and the public during the various decontamination and decommissioning activities. These well-established programs are routinely inspected by the NRC to ensure that workers, the public, and the environment are protected during facility decommissioning activities. It is also important to note that decommissioning activities involve the same radiation protection and waste management considerations as those encountered during plant operations, maintenance, and outages. As described in the DSAR, the decommissioning will be accomplished with no significant adverse environmental impacts in that:

- no site-specific factors pertaining to the decommissioning of the FCS would alter the conclusions presented in NUREG-0586 (see LTP Chapter 8),
- radiation dose to the public will be minimal, and
- decommissioning is not an imminent health or safety concern and will generally have a positive environmental impact.

Continued application of the current and future Radiation Protection and Radiological Effluent Monitoring Programs at FCS ensures public protection in accordance with 10 CFR 20 and 10 CFR 50, Appendix I. ODCM reports for FCS to date conclude that the public exposure as a result of decommissioning activities is bounded by the evaluation in NUREG-0586, which concludes the impact is minimal.

3.6.1 Occupational Exposure

Detailed exposure estimates and exposure controls for specific activities are developed during detailed planning per OPPD Radiation Protection Program procedures. Table 3-3 provides estimated personnel exposures for various decommissioning and fuel storage activities. These estimates were developed to provide site management ALARA goals. The goals are verified by summation of actual site dose, as determined by appropriate dosimetry. ALARA estimates are a compilation of radiation work permit (RWP) estimates for the period. This information is in addition to information gathered for reporting of yearly site dose. The total radiation exposure impact for decommissioning and spent fuel management is estimated to total approximately 230 person-rem.

3.6.2 Radioactive Waste Projections

The OPPD Radioactive Waste Management Program is used to control the characterization, generation, processing, handling, shipping, and disposal of radioactive waste during decommissioning. Activated and contaminated SSCs represent the largest volume of low-level radioactive waste expected to be generated during decommissioning. Other forms of waste generated during decommissioning include:

- contaminated water,
- used disposable protective clothing,
- expended abrasive and absorbent materials,
- expended resins and filters,
- contamination control materials (e.g., strippable coatings, plastic enclosures), and
- contaminated equipment used in the decommissioning process.

Table 3-4 provides projections of waste classifications and quantities that will be generated by the decommissioning of FCS. As OPPD has elected to institute an approach commonly referred to as “rip & ship” verses performing significant on-site decontamination activities, the total volume of radioactive waste for disposal has been estimated at 3,222,861 cubic feet. Actual waste volumes and classifications may vary. The vast majority of waste will be loaded into 8-120A or 3-60B casks, gondola rail cars, or articulating bulk cars and shipped to Clive, Utah. Other radioactive waste will go to the licensed WCS facility in Andrews County, Texas also by gondola railcar or truck. GTCC waste will remain on-site at the ISFSI.

3.7 References

[1] Omaha Public Power District, Fort Calhoun Station Unit 1, "Defueled Safety Analysis Report".

[2] U.S. Nuclear Regulatory Commission, "NUREG-1700, Revision 2, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," 2018.

[3] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.179, Revision 2, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," 2019.

[4] U.S. Nuclear Regulatory Commission, "NUREG-0586, Supplement 1, Volume 1, Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," 2002.

[5] Omaha Public Power District, "NO-FC-10, Quality Assurance Topical Report".

[6] Omaha Public Power District, "FC-21-002, Description of Embedded Piping, Penetrations, and Buried Pipe to Remain in Fort Calhoun End State".

[7] U.S. Nuclear Regulatory Commission, "NUREG-1575, Supplement 1, Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME)," 2009.

[8] U.S. Nuclear Regulatory Commission, "IE Circular No. 81-07, Control of Radioactively Contaminated Materials," 1981.

[9] International Organization for Standardization, "ISO 7503-1, Part 1, Evaluation of Surface Contamination, Beta-Emitters (maximum beta energy greater than 0.15 MeV) and Alpha-Emitters," 2016.

[10] U.S. Nuclear Regulatory Commission, "HPPOS-071, Control of Radioactively Contaminated Material".

[11] U.S. Nuclear Regulatory Commission, "HPPOS-072, Guide on 'How Hard You Have to Look' as Part of Radioactive Contamination Control Program".

[12] U.S. Nuclear Regulatory Commission, "HPPOS-073, Surveys of Wastes from Nuclear Reactor Facilities before Disposal".

[13] U.S. Nuclear Regulatory Commission, "Information Notice No. 85-92, Surveys of Wastes Before Disposal from Nuclear Reactor Facilities," 1985.

[14] Omaha Public Power District, "Fort Calhoun Excavation and Backfill Requirements".

[15] “National Pollutant Discharge Elimination System (NPDES) permit” No. NE0000418.

[16] Omaha Public Power District, "CH-ODCM-0001, Offsite Dose Calculation Manual".

[17] Radiation Safety and Control Services, "TSD 20-001, Historical Site Assessment for Fort Calhoun Station," 2020.

[18] Omaha Public Power District, "FC-20-012, Fort Calhoun Station Decommissioning Project Radiological Characterization Report".

Table 3-1 Status of Major FCS Systems, Structures, and Components as of 7/27/2021

Structure, System, and Component	Status
Reactor Coolant System	In place - Isolated
Reactor Vessel Internals	In place
Reactor Vessel	In place
Steam Generators	In place - Isolated
Reactor Coolant Pumps	In place - Isolated
Pressurizer/Pressurizer Quench Tank	In place - Isolated
Chemical & Volume Control System	In place – Isolated
Safety Injection System	In place – Isolated
Residual Heat Removal System	In place – Isolated
Containment Spray System	In place – Isolated
Component Cooling Water System	In place – Isolated
Raw Water System	In place – Isolated
Spent Fuel Pool (SFP)	In place - Isolated
Fuel Handling Equipment	Removed
Spent Fuel Pool Cooling and Demineralizer System (SFPI systems)	In place - Isolated
Condensate System	In place – Isolated
Feedwater System	In place – Isolated
Steam Generator Blowdown System	In place – Isolated
Primary Makeup Water System	In place – Isolated
Refueling Water Storage Tank	In place
Plant Effluent Monitoring System	In place – Isolated
Containment Ventilation System	In place – Isolated
Radwaste Building Ventilation System	In place – Isolated
Aux Building Ventilation System	In place
Auxiliary Boiler	Removed
Instrument and Service Air System	In place – Isolated

Table 3-1 (continued)

System or Component	Status
Gaseous Radioactive Waste System	In place – Isolated
Solid Radioactive Waste System	Removed
Liquid Radioactive Waste System	In place - Isolated
Makeup Water Systems	In place - Isolated
Radiation Monitoring System	In place
Process Sampling System	In place – Isolated
Fire Protection System	In place
Electrical Systems	In place – Isolated
Lift Stations and Associated Piping	In place
Storm Drains	In place
Containment Building	In place
Auxiliary Building	Mostly In place – Room 66 Removed
Turbine Building	In place
Service Building	In place
Radwaste Processing Building	In place
Intake Building	In place
Security Building	Removed-Disposed
Security Access Facility (SAF)	In place
Maintenance Shop	Removed-Disposed
Technical Support Center (TSC)	Removed-Disposed except slab
Chemistry and Radiation Protection Facility	Removed-Disposed
New Warehouse	Removed-Disposed
Mausoleum (OSGSF)	In place
New Maintenance Storage Shed	In place
Chemical Storage Shed	In place
Neutralization (Tertiary) Building and Basin	In Place

Table 3-1 (continued)

System or Component	Status
Training Center	To remain at license termination
Administration Building	In place
6 – Bay Garage	In Place
Flex Building	To remain at license termination
Switchyard	To remain at license termination
Switchyard 3451 Old Building	To remain at license termination
Switchyard 3451 New Building	To remain at license termination
1251 Control and Switchgear Building	To remain at license termination
ISFSI Operation Facility (IOF)	To remain in 10 CFR Part 50 license
Fuel Oil Tanks (FO-1, FO-10, FO-27, FO-32, FO-43 A/B, FO-44A/B)	In Place - Isolated
Owner Controlled Area (OCA) Entrance Building	To remain at license termination

Table 3-2 General Project Milestones

Date	Milestone
Q2/2020	Complete Transfer of Spent Nuclear Fuel to ISFSI
Q2/2021	Complete Demolition of New Warehouse and Security Building
Q2/2021	Complete Demolition of Chemistry and RP Building
Q2/2021	Complete Demolition of Maintenance Building
Q3/2021	Complete Demolition of Technical Support Center
Q3/2021	License Termination Plan Submittal to NRC
Q4/2021	Complete Demolition of Service Building and Intake Structure
Q4/2021	Complete Containment Building Access and Rail Installation
Q3/2022	Complete Demolition of Turbine Building
Q1/2023	Reactor Vessel Internals Segmentation Complete
Q2/2023	Complete Demolition of Auxiliary Building
Q2/2023	Complete Demolition of Radwaste Building
Q3/2023	Reactor Vessel Segmentation Complete
Q3/2024	Complete Demolition of Security Access Facility
Q3/2024	Large Component Removal Complete
Q2/2025	Complete Demolition of Mausoleum
Q3/2025	Complete Demolition of Containment Building
Q3/2025	Final Status Survey Complete
Q4/2026	License Termination

Note: Circumstances can change during decommissioning. If OPPD determines that the decommissioning cannot be completed as outlined in this schedule, OPPD will provide an updated schedule to the NRC.

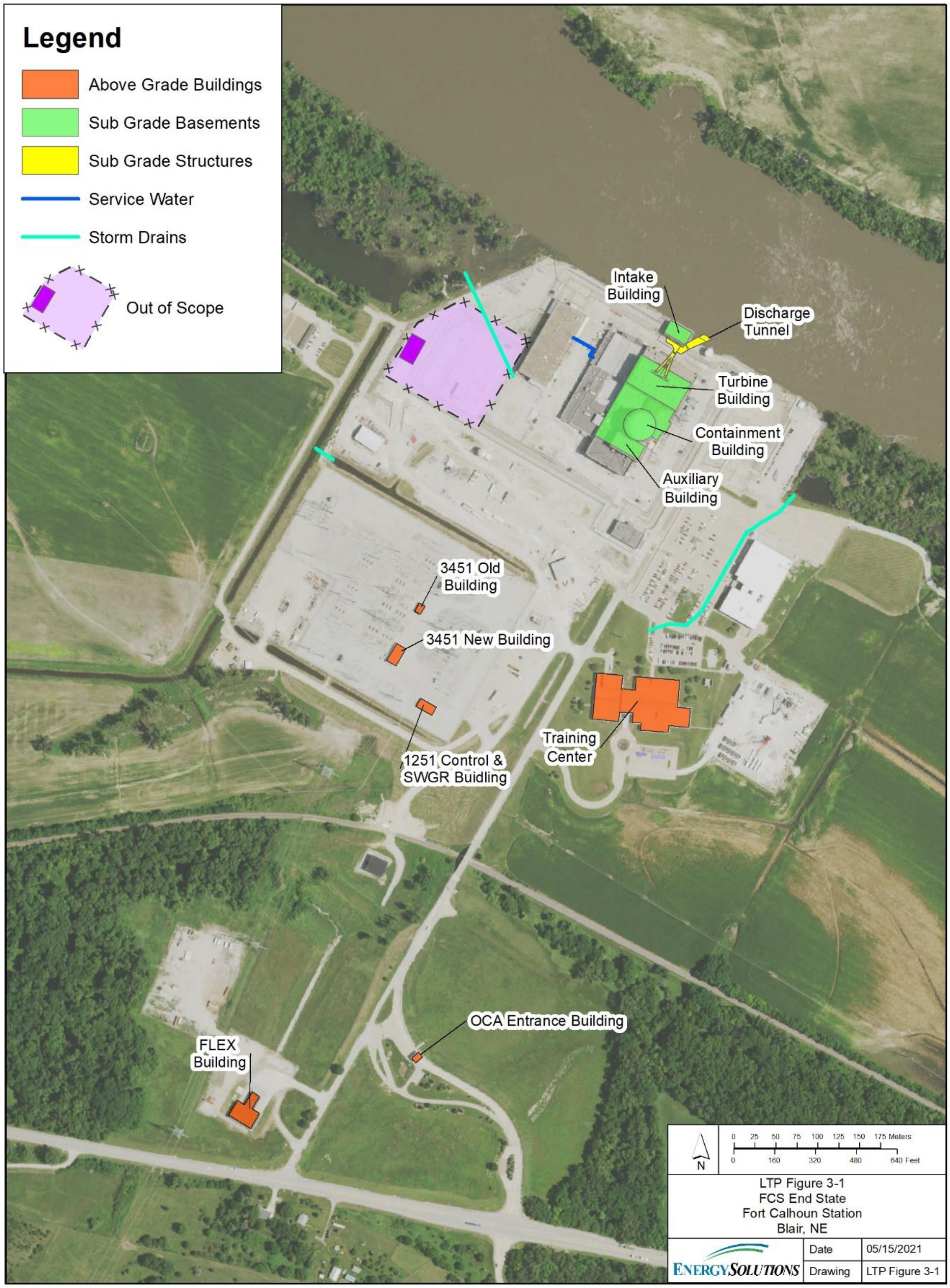
Table 3-3 Radiation Exposure Projections for Decommissioning After 8/1/2021

Activity	Exposure (person-rem)
Reactor Vessel Head Work	2.0
Reactor Vessel/Internals Segmentation	54.290
Steam Generators and Pressurizer	20.0
Reactor Coolant System	7.0
Plant Systems	5.0
Structure Demolition	3.0
Asbestos Removal Activities	0
Waste Operations Activities	42.76
Subtotal Dismantlement Activities	**134.05**
Walkdowns/ Tours/ Security	2
RP Routine Activities	20
Misc. D&D Work Activities	7.5
NRC Activities	0.15
FSS Activities	0.05
ISFSI Activities	0.05
Visitor Activities	0.2
Subtotal Operational Activities	**29.950**
TOTAL Estimate (remaining after 08/01/2021)	**164.0**
FCS Project Radiation Exposure (thru 07/31/2021)	**121 person-rem of the current estimate of 285 person-rem.**

Table 3-4 Projected Waste Quantities

Waste Type	Waste Class	Waste Weight (lbs.)	Packing Density (lbs./cubic feet)	Waste Volume (cubic feet)
Bulk Concrete	A	125,570,000	100	1,256,000
Soils	A	63,300,000	80	792,000
Metal Debris	A	39,938,131	36	1,121,580
Large Components	A	3,690,868	varies	50,696
Asbestos	A	100,000	35	3,460
DAW	A	2,000,000	18	100,440
Ion exchange media, processed wet waste	A	352,021	varies	7,800
HazMat (containerized)	A	100,000	200	500
Highly Radioactive (filters/sludge/debris)	B or C	3,100	6	118
Highly Radioactive (irradiated/non-irradiated hardware)	B or C	177,000	90	1,339
Very Highly Radioactive	>C	98,841	132.49	746
Clean Concrete (local landfill)	-	224,818,806	90	2,497,987
Clean Non-hazardous (local landfill)	-	782,766	15	52,184
Clean Scrap Metal (recycler)	-	56,924,256	20	2,846,213
Totals	**-**	**517,855,789**		**8.731,063**

Figure 3-1 Fort Calhoun Station End State



FORT CALHOUN STATION DECOMMISSIONING PROJECT
LICENSE TERMINATION PLAN

CHAPTER 4
REMEDIATION PLAN

REVISION 0

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ABBREVIATIONS

ALARA	as low as reasonably achievable
AMCG	average member of the critical group
AMSL	above mean sea level
BFM	basement fill model
D&D	decontamination and dismatlement
DCGL	derived concentration guideline level
EMC	elevated measurement comparison
FCS	Fort Calhoun Station
FSS	final status survey
HEPA	high efficiency particulate air
ISFSI	Independent Spent Fuel Storage Installation
LLRW	low-level radioactive waste
LTP	license termination plan
NRC	U.S. Nuclear Regulatory Commission
OPPD	Omaha Public Power District
RP	radiation protection

4 SITE REMEDIATION PLAN

In accordance with 10 CFR 50.82(a)(9)(ii)(C), the license termination plan (LTP) must provide the plans for site remediation. These plans must include the provisions to meet the criteria from Subpart E of 10 CFR 20 before the site may be released for unrestricted use. The two radiological criteria for unrestricted use specified in 10 CFR 20.1402 are: (1) the total effective dose equivalent from residual radioactivity that is distinguishable from background radiation must not be greater than 25 mrem/year to the average member of the critical group (AMCG) and (2) residual radioactivity levels must be as low as is reasonably achievable (ALARA).

Decontamination and dismantlement (D&D) activities will be conducted in accordance with established Radiation Protection (RP), Safety, and Waste Management programs which include approved written procedures. These programs and procedures are frequently audited for technical content and compliance. Revisions have and will continue to be made to these programs and procedures to accommodate the changing work environment inherent to reactor decommissioning. The programs and procedures are documented, processed, and approved in accordance with existing administrative procedures using 10 CFR 50.59 and Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59 Changes, Tests and Experiments" [1] as guidance. Consistent with Regulatory Guide 1.179, "Standard Format and Contents for License Termination Plans for Nuclear Power Reactors" [2], details regarding changes to the RP Program to address remediation and decommissioning activities are not provided in this LTP, but periodic updates to the Fort Calhoun Station (FCS) "Defueled Safety Analysis Report" [3] will provide such details.

This chapter describes the methods that may be used to remediate contaminated systems, components, and structures. The methods for demonstrating compliance with the ALARA criterion in 10 CFR 20.1402 are also described. LTP Chapter 6 provides the methods for demonstrating compliance with the 25 mrem/year dose criterion. LTP Chapter 3 describes in detail the remaining site remediation and dismantlement activities and the order in which they will occur for each structure, system, and component.

This chapter also provides a summary of the radiation protection methods and control procedures that will be employed during site dismantlement and remediation.

4.1 Remediation Actions and ALARA Evaluations

When D&D actions are completed, residual radioactivity may remain on building surfaces and in site soils at concentrations that correspond to the maximum annual dose criterion of 25 mrem/year. The remaining residual radioactivity must also satisfy the ALARA criterion, which requires an evaluation as to whether it is feasible to further reduce residual radioactivity to levels below those necessary to meet the dose criterion (i.e., to levels that are ALARA).

The ALARA evaluation calculates the concentration at which the averted collective radiation dose, converted into dollars, is equal to the costs of continued remediation (e.g., risk of transportation accidents converted into dollars, worker and public doses associated with the remediation action converted into dollars, and the actual costs to perform the remediation activity). If this concentration is below the concentrations that correspond to the maximum annual dose criterion, then further reduction of residual radioactivity is justified by ALARA.

Regardless of the outcome of the quantified cost/benefit calculation provided in this chapter, the final dose from residual radioactivity is expected to be below the dose criterion. The majority of the basement surfaces to be backfilled have minimal contamination. In addition, any areas that are identified as potentially containing activity at levels that could exceed the derived concentration guideline level (DCGL), as measured during final status survey (FSS), will be remediated. Industry standard remediation methods have been shown to remove contamination to levels significantly below the target levels, in this case the DCGL, and this result is expected for any remediation. The combination of low contamination levels over the majority of the basement surfaces combined with remediated areas likely containing activity below the DCGL, ensures that the final dose from residual radioactivity at license termination will be below the 25 mrem/year dose criterion. Based on characterization results, there is limited contamination expected in soil, buried pipe, or end-state structures with a corresponding dose that is also expected to be below 25 mrem/year.

4.2 Remediation Actions

Remediation actions are performed throughout the decommissioning process, and the techniques, methods, and technologies are standard to the commercial nuclear industry. All of the remediation actions described may not necessarily be required but are listed as possible actions that may be taken during the decommissioning of FCS. The appropriate remediation techniques, methods, and technologies that will be employed are dependent on the physical composition and configuration of the contaminated media requiring remediation. At FCS, the principal media that will be subjected to remediation are concrete structural surfaces. Characterization survey results and historical survey data indicate that there is minimal soil and groundwater contamination.

4.2.1 Structures and Piping

The following above-ground buildings will remain at the time of license termination: Training Center, FLEX Building, Owner Controlled Area Entrance Building, 3451 Old Building, 3451 New Building, and the 1251 Control and Switchgear Building. With the exception of the removal of the 345kv and 161kv lines to the station, the Switchyard will remain after decommissioning. All other structures will be removed to a minimum of three feet below grade (approximately 1,001 feet above mean sea level [AMSL]). Because there are no dose model implications, if the number and identification of above-ground buildings to remain and be subject to FSS changes, the U.S. Nuclear Regulatory Commission (NRC) will be notified, but an LTP revision will not be required.

The Independent Spent Fuel Storage Installation (ISFSI) and the ISFSI Operations Facility will remain in the reduced 10 CFR Part 50 license and are not within the scope of the proposed partial site release.

The basements of the Turbine Building, Containment Building, Auxiliary Building, Intake Structure, and Circulating Water Tunnels will remain with all interior walls removed, with the exception of the Turbine Building where the turbine pedestal will remain up to three feet below grade. All other structures will be removed in their entirety with the exception of the six above-ground buildings listed above and the Switchyard. Once D&D activities are completed, as well as subsequent FSS, each basement will be backfilled with fill material to grade level (approximately 1,004 feet AMSL).

Static measurements, In Situ Object Counting System measurements, or the analysis of volumetric samples will be used to calculate the remaining total concentration of residual activity. The concrete walls and floors of the basements will be remediated to levels that will provide high confidence that FSS measurements will not exceed radionuclide-specific DCGLs that represent the annual dose criterion for unrestricted use specified in 10 CFR 20.1402.

Remediation techniques that may be used for the structural surfaces below the 1,001 feet AMSL elevation include washing, wiping, pressure washing, vacuuming, scabbling, chipping, and sponge or abrasive blasting. Cost estimates for these techniques also include the amount of water generated and the cost to process, package, and ship this waste. Concrete removal may include using machines with hydraulic-assisted, remote-operated, articulating tools. These machines have the ability to exchange scabbling, shear, chisel, and other tool heads.

4.2.1.1 Scabbling and Shaving

The principal remediation method expected to be used for removing contaminants from concrete surfaces is scabbling and shaving. Scabbling entails the removal of concrete from a surface by the high-velocity impact of a tool with the concrete surface which transforms the solid surface to a volumetric particulate which can be removed. One method of scabbling is a surface removal process that uses pneumatically operated air pistons with tungsten-carbide tips that fracture the concrete surface to a nominal depth of 0.125 inches at a nominal rate of approximately130 ft^2 or 12.07 m^2 per hour. The scabbling pistons (feet) are contained in a close-capture enclosure that is connected by hoses to a sealed vacuum and collector system. Shaving uses a series of diamond cutting wheels on a spindle and performs at similar rates to scabbling. The wheels are also contained in a close-capture enclosure similar to scabbling equipment. The fractured media and dusts from both methods are deposited into a sealed removable container. The exhaust air passes through both roughing and absolute high efficiency particulate air (HEPA) filtration devices. Dust and debris generated through these remediation processes is collected and controlled during the operation.

4.2.1.2 Needle Guns

The needle gun is a pneumatic, air-operated tool containing a series of tungsten-carbide or hardened steel rods enclosed in a housing. The rods are connected to an air-driven piston to abrade and fracture the media surface. The media removal depth is a function of the residence time of the rods over the surface. Typically, one to two millimeters are removed per pass. Generated debris collection, transport, and dust control are accomplished in the same manner as other scabbling methods. Use of needle guns for removal and chipping of media is usually reserved for areas not accessible to normal scabbling operations. These include inside corners, cracks, joints, and crevices. Needle gunning techniques can also be applied to painted and oxidized surfaces. As with scabbling, dust and debris generated through these remediation processes is collected and controlled during the operation.

4.2.1.3 Chipping

Chipping includes the use of pneumatically operated chisels and similar tools coupled to vacuum-assisted collection devices. Chipping activities are usually reserved for cracks and crevices. This method is also a form of scabbling.

4.2.1.4 Sponge and Abrasive Blasting

Sponge and abrasive blasting are similar techniques that use media or materials coated with abrasive compounds such as silica sands, garnet, aluminum oxide, and walnut hulls. Sponge blasting is less aggressive, incorporating a foam media that, upon impact and compression, absorbs contaminants. The medium is collected by vacuum and the contaminants are washed from the medium so the medium may be reused. Abrasive blasting is more aggressive than sponge blasting but less aggressive than scabbling. Both operations use intermediate air pressures. Sponge and abrasive blasting are intended for the removal of surface films and paints.

4.2.1.5 Pressure Washing

Pressure washing uses a nozzle of intermediate water pressure to direct a jet of pressurized water that removes superficial materials from the suspect surface. A header may be used to minimize over-spray. A wet vacuum system is used to suction the potentially contaminated water into containers for filtration or processing.

4.2.1.6 Washing and Wiping

Washing and wiping decontamination techniques are actions that are typically performed during the course of remediation activities for housekeeping and to minimize the spread of loose surface contamination. FCS will implement good housekeeping throughout decommissioning to ensure ALARA and that loose surface contamination is removed prior to evaluating the surface for acceptable concentrations of residual activity.

Washing and wiping techniques are actions that are normally performed during the course of remediation activities and will not always be evaluated as a separate ALARA action. When washing and wiping techniques are used as the sole means to reduce residual contamination below DCGL levels, ALARA evaluations will be performed. Washing and wiping techniques used as housekeeping or good practice measures will not be evaluated.

4.2.1.7 High-Pressure Water Blasting

One method that may be used to remediate pipe interior surfaces is high-pressure water blasting. A high-pressure, liquid-jetting system has a high pressure water pump capable of producing a water pressure of 10,000 pounds per square inch (psi) to 20,000 psi at an actual flow rate that ranges from 44 gallons per minute at 10,000 psi to 23 gallons per minute at 20,000 psi. A rotating jet-mole tip is used for 360 degree coverage of pipe interiors. The jet-mole is attached to a lance and high-pressure hose. The lance is manually advanced through the interior of the pipe. As the lance is advanced, the high-pressure water abrades the interior surface of the pipe, removing the corrosive layer, internal debris, and radiological contamination. The waste water containing the removed contamination is then collected and stored for processing as liquid radiological waste.

4.2.1.8 Grit Blasting

Another approach that may be used to remediate the surfaces of pipe interiors is grit blasting. Grit blasting uses grit media such as garnet or sand under intermediate air pressure directed through a nozzle that is pulled through the closed piping at a fixed rate. The grit blasting action removes the interior surface layer of the piping. A HEPA vacuum system maintains the sections being cleaned under negative pressure and collects the media for reuse or disposal. The final system pass is performed with clean grit to remove any residual contamination.

4.2.1.9 Removal of Activated and Contaminated Concrete

As previously stated, the principal means of remediating concrete surfaces is scabbling and shaving. If the concrete structure is designated for complete removal, such as interior concrete walls or the containment bioshield, the primary method that will be used to completely remove the concrete is through large scale demolition using hydraulic-operated crushing shears and jack-hammers fitted to large, tracked excavators. Concrete structures will be fractured and crushed by these tools. As the concrete is reduced to rubble, the embedded rebar will be exposed and segregated from the concrete rubble. In situations where a more surgical removal is required, activated or contaminated concrete removal may be accomplished using a machine-mounted, remote-operated articulating arm with interchangeable tooling heads. As concrete is fractured and rebar is exposed, the metal is cut using flame cutting equipment. The concrete rubble and exposed rebar is collected and transferred into containers for later disposal in both techniques. Dusts, fumes, and generated debris are locally collected and, as necessary, controlled using temporary enclosures coupled with close-capture HEPA systems or controlled water misting systems. Bulk concrete such as floors and walls may be removed as intact sections after sawing with blades, wires, or other cutting methods.

4.2.1.10 Additional Remedial Actions

Mechanical abrasive equipment, such as hones, may be used to remove contamination from the surfaces of embedded or buried piping. Chemical removal means may be used, as appropriate, for the removal of certain contaminants.

4.2.2 Soil

The soil $DCGL_W$ that will be used to demonstrate compliance with the dose-based criteria of 10 CFR 20, Subpart E for the unrestricted release of open land survey units are provided in Table 5-6 of Chapter 5. Section 2.5.1.1 of NUREG-1575, “Multi-Agency Radiation Survey and Site Investigation Manual” [4] addresses the concern for the presence of small areas of elevated radioactivity. A simple comparison to an investigation level is used to assess the dose impact of potential elevated areas. This is referred to as the elevated measurement comparison (EMC). The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_W$ modified by an area factor to account for the small area of the elevated radioactivity. Any radiological contamination in soils identified in concentrations greater than the $DCGL_{EMC}$ will be removed and disposed of as radioactive waste.

The site characterization process has established the location and extent of soil contamination at FCS. Characterization survey results and historical survey data indicate that there is minimal residual radioactivity in soil and groundwater. As needed, additional investigations will be performed to ensure that any changing soil radiological contamination profile during the remediation actions is adequately identified and addressed. LTP Chapter 5 discusses soil sampling and survey methods.

Soil remediation equipment may include shovels and back hoe and track hoe excavators. Other equipment including soil dredges and vacuum trucks may also be used. As practical, when the remediation depth approaches the soil interface region between unacceptable and acceptable contamination, a squared edge excavator bucket design or similar technique may be used. This simple methodology minimizes the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket.

Remediation of soils will be performed using established excavation safety and environmental control procedures. Operational constraints and dust control will be addressed in site excavation and soil control procedures. In addition, work package instructions for remediation of soil may include additional constraints and mitigation or control methods to ensure adequate erosion, sediment, and air emission controls during soil remediation.

4.3 Remediation Activities Impact on the Radiation Protection Program

The current approved RP Program is adequate to comply with all federal and state regulatory requirements for the protection of occupational personnel from radiological hazards encountered or expected to be encountered during the decommissioning of a commercial pressurized water reactor facility. In addition, the program ensures the protection of the public from radiological hazards and ensures occupational, effluent, and environmental dose from exposure to radioactive materials is and remains ALARA. To ensure that adequate and proper engineering controls and hazard mitigation techniques are employed, work control programs and procedural requirements allow RP personnel to integrate radiation protection and radiological hazard mitigation measures directly into the work planning and scheduling process. Consequently, the necessary radiological controls are correctly implemented to accommodate each remediation technology as appropriate.

The spread of loose surface contamination is mitigated by the routine remediation of work areas by washing and wiping. Water washing with a detergent is effective in reducing low levels of loose surface contamination over large surface areas. Wiping with detergent soaked or oil-impregnated media is an effective technique to reduce loose surface contamination on small items, overhead spaces and small hand tools. These same techniques are also effective in reducing low levels of surface contamination on structural surfaces.

For intermediate levels of surface contamination, more aggressive methods such as pressure washing, high-pressure water blasting, and grit blasting may be more appropriate. Pipes, surfaces, and drain lines can be cleaned and hot spots removed using these techniques and technologies. Small tools, hoses, and cables can also be pressure washed in a containment to reduce contamination levels. A paint coating may be applied after surface cleaning to prevent surface contamination from drying out and becoming airborne.

To mitigate high levels of fixed surface contamination embedded in concrete, scabbling or other surface removal techniques may be appropriate. A combination of mechanical and flame cutting will be used to section the reactor vessel and its internals.

Concrete cutting or surface scabbling, mechanical cutting, abrasive water jet cutting, hydrolazing, and grit blasting are remediation techniques that have been used at FCS in the past during operations. The current RP Program provides adequate controls for these actions.

A dedicated haul road will be used to transport radioactive waste between the Containment Building enclosure and the waste loadout enclosure. The haul road and transport paths will be surveyed via scanning on a routine basis by the RP Group.

Decommissioning does not present any new challenge to the RP Program above those encountered during normal plant operation and refueling. Decommissioning planning allows RP personnel to focus on each area of the site and plan each activity well before execution of the remediation technique.

The decommissioning organization is experienced in, and capable of applying, these remediation techniques on contaminated systems, structures, or components during decommissioning. The RP Program is adequate to safely control the radiological aspects of this work. Because the activities expected during decommissioning are the same or similar to those encountered during operations, as described above, the approval of any changes to the existing approved RP Program as described in the NRC Docket Number 50-285, Facility Operating License Number DPR-40, is not requested in this LTP.

4.4 ALARA Evaluation

Guidance for conducting ALARA analyses is provided in Appendix N of NUREG-1757, Volume 2, "Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report" [5], which describes acceptable methods for determining when further reduction of residual radioactivity is required to concentrations below the levels necessary to satisfy the 25 mrem/year dose criterion.

Section N.3.6 of NUREG-1757 states, "For residual radioactivity in soil at sites that may have unrestricted release, generic analyses (see NUREG-1496 and the examples in Section N.2.5) show that shipping soil to a licensed low-level waste disposal facility is unlikely to be cost effective for unrestricted release, largely because of the high costs of waste disposal. An ALARA analysis is not needed for soil removal to meet unrestricted release at or below a dose criterion of 0.25 mSv (25 mrem) per year. Therefore, the licensee generally does not have to evaluate shipping soil to a low-level waste disposal to achieve exposure levels at or below the criterion for unrestricted release." To illustrate that this is a reasonable approach and applicable to FCS, a simple ALARA analysis for the excavation and disposal of soils as LLRW is provided in Section 4.4.1.

For above-ground buildings, below-ground basements (including embedded piping and penetrations), and buried piping, surfaces with residual radioactivity concentrations in excess of the $DCGL_W$ will be remediated or removed as radioactive waste. For structures that will remain at license termination, the ALARA analysis will determine whether further remediation is necessary by comparing the desired beneficial effects to the undesired costs. Benefits are the averted collective radiation dose (converted into dollars) following the removal of radioactivity. The costs of remediation include transportation accidents, worker and public dose associated with remedial action, and the actual costs to perform the remediation (converted into dollars). If the costs exceed the benefits, then the dose reduction achieved by further remediation is not ALARA.

The ALARA criterion specified in 10 CFR 20.1402 is not met by solely performing remediation. The ALARA analysis is a planning tool to justify that further remediation is not necessary. When remediation is performed, there is no need to analyze whether the action was necessary to meet the ALARA requirement.

The methods and results of the ALARA evaluation for concrete remediation in structures below the 1,001 feet elevation is provided in Section 4.4.2.

4.4.1 ALARA Analysis of Soil Remediation

In order to determine if additional remedial action is warranted by ALARA analysis, the desired beneficial effects (benefits) and the undesirable effects (costs) must be calculated. If the benefits from remedial action will be greater than the costs, then the remedial action is warranted and should be performed. However, if the costs exceed the benefit, then the remedial action is considered to be not ALARA and should not be performed.

Based upon a simple ALARA analysis, the only benefit of reducing residual radioactivity in soil is the monetary value of the collective averted dose to future occupants of the site. For soils, the averted dose is based upon the "resident farmer" scenario.

4.4.1.1 Calculation of Benefits

The benefit from collective averted dose, B_{AD}, is calculated by determining the present worth of future collective averted dose and multiplying by a factor to convert the dose to a monetary value. In accordance with Appendix N of NUREG-1757, the equation is as follows:

Equation 4-1

$$B_{AD} = V_{AD} \times PW(AD_{Collective})$$

where

B_{AD} = benefit from an averted dose for a remediation action, in US dollars

V_{AD} = value of averted dose, which is a conversion factor for the monetary value of radiation dose (dollars per person-rem)

$PW(AD_{Collective})$ = present worth of a future collective averted dose in person-rem

The present worth of future collective averted dose, $PW(AD_{Collective})$, is then expressed in accordance with the following equation:

Equation 4-2

$$PW(AD_{Collective}) = P_D \times A \times 0.025 \times F \times \frac{Conc}{DCGL_W} \times \frac{1 - e^{-(r+\lambda)N}}{r + \lambda}$$

where P_D = population density for the critical group scenario in people/m^2

A = area being evaluated in m^2

0.025 = annual dose to an AMCG from residual radioactivity at the $DCGL_W$ concentration in rem/year

F = effectiveness, or fraction of the residual radioactivity removed by the remediation action

Conc = average concentration of residual radioactivity in the area being evaluated in units of activity per unit area for buildings or activity per unit volume for soil

$DCGL_W$ = derived concentration guideline equivalent to the average concentration of residual radioactivity that would give a dose of 25 mrem/year to the AMCG, in the same units as *Conc*

r = monetary discount rate in units per year

λ = radiological decay constant for the radionuclide in units per year

N = number of years over which the collective dose will be calculated

4.4.1.2 ALARA Analysis Parameters

In accordance with Table N.2 of NUREG-1757, which is recreated as Table 4-1 below, the acceptable and relevant parameters for use in performing ALARA analyses are as follows:

Table 4-1 Acceptable Parameter Values for Use in ALARA Analyses

Parameter	Value	Reference and Comments
Workplace accident fatality rate, F_W	4.2×10^{-8}/hr	NUREG-1496, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities" and NUREG-1496, Volume 2 [6], Appendix B, Table A.1
Transportation fatal accident rate, F_T	Trucks: 3.8×10^{-8}/hr	NUREG-1496, Volume 2, Appendix B, Table A.1
Value of averted dose, V_{AD}, and value of statistical life, V_{SL}	Values are updated periodically	NUREG/BR-0058 [7], NUREG-1530 [8].
Monetary discount rate, r	0.03/year and 0.07/year discount rates, with special considerations for	NUREG/BR-0058

Parameter	Value	Reference and Comments
	integrationsal consequences (as discussed in Section N.2.2)	
Number of years of exposure, N	Buildings: 70 years Soil: 1,000 years	NUREG-1496, Volume 2, Appendix B, Table A.1
Population density, P_D	Building: 0.09 person/m^2 Land: 0.0004 person/m^2	NUREG-1496, Volume 2, Appendix B, Table A.1
Excavation, monitoring, packaging, and handling soil	Soil: 1.62 person-hours/m^3 of soil	NUREG-1496, Volume 2, Appendix B, Table A.1
Waste shipment volume, V_{SHIP}	Truck: 13.6 m^3/shipment	NUREG-1496, Volume 2, Appendix B, Table A.1

- Value of averted dose (V_{AD}): $5,100.00/person-rem (per NUREG-1530)
- Value of statistical life (V_{SL}): $9,000,000.00 (per NUREG-1530)
- Monetary discount rate (r): 0.00 yr^{-1} for soil

  Note: This variable was established at 0.03 yr^{-1} for soil in Table N.2 of Appendix N of NUREG-1757. The monetary discount for the ALARA analysis was removed from the equation through Federal Register Notice 72 FR 46102 – August 16, 2007. Consequently, the r variable has been conservatively set at 0.00 yr^{-1} for soil (i.e., no monetary discount for soils as well as basements).
- Area (A) used to calculate the population density: 10,000 m^2 (size of reference area that was evaluated)

4.4.1.3 Calculation of Costs

The total cost ($Cost_T$), which is balanced against the benefits, has several components and may be evaluated according to Equation N-3 of NUREG-1757 below:

Equation 4-3

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} + Cost_{other}$$

where $Cost_R$ = monetary cost of the remediation action (may include "mobilization" costs)

$Cost_{WD}$ = monetary cost for transport and disposal of the waste generated by the action

$Cost_{ACC}$ = monetary cost for worker accidents during the remediation action

$Cost_{TF}$ = monetary cost of traffic fatalities during transport of the waste

$Cost_{WDose}$ = monetary cost of dose received by workers performing the remediation action and transporting the waste to the disposal facility

$Cost_{PDose}$ = monetary cost of the dose to the public from excavation, transport, and disposal of the waste

$Cost_{other}$ = other costs as appropriate for the particular situation

4.4.1.4 Calculation of Total Cost for Soil Remediation by Excavation and Disposal

For the analysis of soil excavation and disposal as low-level radioactive waste, the variables for $Cost_R$, $Cost_{ACC}$, $Cost_{WDose}$, and $Cost_{PDose}$ were not calculated for this evaluation based upon their anticipated unlikely impact on the total cost. This is consistent with the guidance provided in NUREG-1757 which states that if one or two of the costs can be shown to exceed the benefit, then the remediation cost is shown to be unnecessary without calculating all of the costs.

4.4.1.4.1 Transport and Disposal of the Waste ($Cost_{WD}$)

The cost of waste transport and disposal ($Cost_{WD}$) was calculated using Equation N-4 of NUREG-1757, which is expressed as follows:

Equation 4-4

$$Cost_{WD} = V_A \times Cost_V$$

where V_A = volume of waste produced, remediated in units of m^3

$Cost_V$ = cost of waste disposal per unit volume, including transportation cost, in units of dollars per m^3

Disposal cost for generated waste was calculated as $2,046.94/$m^3$. This cost includes packaging, transportation, and burial fees. The transportation component of this cost is based on the average transportation cost of using either rail or highway hauling from the FCS site to Clive, Utah (the Energy*Solutions* radioactive waste disposal facility). The details of waste disposal costs are provided in Chapter 7.

The volume of waste produced by remediation (V_A) assumes that the reference area of 10,000 m^2 (A) is remediated to a depth of 0.15 meters. This results in a value for waste volume (V_A) of 1,500 m^3, which produces a value for $Cost_{WD}$ of $3,070,415.28.

4.4.1.4.2 Transportation Risks ($Cost_{TF}$)

The cost of traffic fatalities incurred during the transportation of waste ($Cost_{TF}$) was calculated using Equation N-6 of NUREG-1757, which is expressed as follows:

Equation 4-5

$$Cost_{TF} = V_{SL} \times \frac{V_A}{V_{SHIP}} \times F_T \times D_T$$

where V_{SL} = monetary value of a statistical life (or fatality) equivalent to $5,100/person-rem (per NUREG-1530)

V_A = volume of waste producted in units of m^3

F_T = fatality rate per truck-kilometer traveled in units of fatalities/truck-km

D_T = distance traveled in km

V_{SHIP} = volume of a truck shipment in m^3

For this evaluation, the waste volume (V_A) is assumed to be 1,500 m^3, and the haul volume of an overland truck shipment per NUREG-1757 is assumed to be 13.6 m^3 (V_{SHIP}).

In accordance with NUREG-1496, Appendix B, Table A.1, a value of 3.80E-08/hr was used for F_T.

The Clive, Utah, round trip distance from the FCS site by highway is approximately 2,044 miles (approximately 3,290 km). The distance for rail shipments is further than that for highway shipments because of the route rail shipments must follow, however the difference as it pertains to the calculation is insignificant. The highway shipment distance of 3,290 km (D_T) was used for the calculation of $Cost_{TF}$. For this evaluation, the value for the $Cost_{TF}$ variable is $124,100.74.

4.4.1.4.3 __Total Cost ($Cost_T$)__

The total cost ($Cost_T$) assumed for this evaluation is $3,194,516.02.

4.4.1.5 Residual Radioactivity in Soils that are ALARA

Determination of residual radioactivity in soils that are ALARA is the concentration at which benefit equals or exceeds the costs of removal and waste disposal. When the total cost ($Cost_T$) is set equal to the dose averted, the ratio of the concentration to the $DCGL_W$ is calculated as follows:

Equation 4-6

$$\frac{Conc_{ALARA}}{DCGL_W} = \frac{Cost_T}{V_{AD} \times P_D \times 0.025 \times F \times A} \times \frac{r + \lambda}{1 - e^{-(r+\lambda)N}}$$

The following values are assumed for the remaining variables:

- the default parameter values from Section 4.4.1.2
- a value of one for remediation effectiveness (F), assuming all residual radioactivity is removed during the excavation
- a soil $DCGL_W$ of 12.37 pCi/g for Cs-137

With these values, the ratio of the concentration to the $DCGL_W$ is determined with the the following equation:

Equation 4-7

$$\frac{Conc_{ALARA}}{DCGL_W} = \frac{\$3{,}194{,}516.02}{\$5{,}100 \times 0.0004 \times 0.025 \times 1 \times 10{,}000} \times \frac{0 + 0.02297}{1 - e^{-(0+0.02297)1{,}000}}$$

The ratio of the concentration to the $DCGL_W$ when $Cost_T$ is equal to the dose averted is 143.88.

Assuming a concentration set at 50% of the $DCGL_W$ (based on the investigation level for a Class 3 area), $PW(AD_{Collective})$ can be calculated as follows:

Equation 4-8

$$PW(AD_{Collective}) = 0.0004 \times 10{,}000 \times 0.025 \times 1 \times \frac{6.55}{13.1} \times \frac{1 - e^{-(0+0.02297)1{,}000}}{0 + 0.02297}$$

This calculation results in a $PW(AD_{Collective})$ of 2.18 person-rems. The benefit from collective averted dose (B_{AD}) is then calculated as follows:

Equation 4-9

$$B_{AD} = \$5{,}100 \times 2.18 = \$11{,}101.52$$

This simple analysis confirms the statement in Section N.1.5 of NUREG-1757 that the cost of disposing excavated soil as low-level radioactive waste is clearly greater than the benefit of removing and disposing of soil with residual radioactivity concentrations less than the dose criterion. Because the cost is greater than the benefit, it is not ALARA to excavate and dispose of soils with residual radioactivity concentrations below the $DCGL_W$.

4.4.2 ALARA Analysis for Remediation of Basement Structures

With the exception of some penetrations, embedded piping, and buried piping, all contaminated and non-contaminated systems will be disassembled, removed, packaged, and shipped off-site as a waste stream commodity. Once commodity removal is complete, structural surfaces will be remediated, as necessary. Once remediation is complete, structural surfaces located above the 1,004 feet AMSL elevation and non-load-bearing interior concrete walls below the 1,001 feet AMSL elevation foot elevation will be demolished, reduced in size, packaged, and shipped off-site to a licensed disposal facility.

All concrete inside the liner will be removed from the interior of the Containment Building prior to demolition. This includes all activated and contaminated concrete and any embedded piping. The source term in the Containment Building basement remaining after demolition will consist of low levels of surface contamination on the exposed liner surfaces. There is minimal contamination in the Turbine Building, Intake Structure, and Circulating Water Tunnels at levels that are expected to be below the DCGLs listed in LTP Chapter 5. The only portion of the Radwaste Processing Building that will remain following building demolition is the below-ground electrical vault in Room 505.

In summary, the vast majority of residual radioactivity remaining in the structures after the concrete is removed from the Containment Building basement, will be located in the Auxiliary Building basement. Therefore, the ALARA assessment for the remediation of basement structures will focus on the floors of 971 feet and 989 feet AMSL elevations of the Auxiliary Building, as this is the location where the greatest benefit of concrete remediation could be achieved. An ALARA assessment of the Auxiliary Building basement floor will bound ALARA assessments for the other buildings which would use the same methods (and cost estimate) but remove less contamination.

The Auxiliary Building basement floor concrete is volumetrically contaminated. A total of 261 volumetric concrete samples were collected from 67 locations in the Auxiliary Building during characterization. The sample analysis of these concrete samples indicates that the majority of the radionuclide inventory resides within the first ½-inch of concrete. However, a number of samples showed detectable Cs-137 and Co-60 at depths in excess of 6 inches.

4.4.2.1 ALARA Analysis Equation for Remediation of Basement Structures

The ALARA analysis for the remediation of basement structures uses Equation 4-6 from Section 4.4.1.5 with modifications to account for multiple radionuclides and the three basement dose scenarios (*insitu*, drilling spoils, and excavation). The 0.025 term in Equation 4-6 is replaced by the radionuclide mixture fraction (f_i) for each radionuclide multiplied by the averted dose for each radionuclide within each dose scenario ($DOSE_{AMCGi,j}$) as shown in Equation 4-10. Using Equation 4-10, individual $Conc_{ALARA}/DCGL_W$ values are calculated for each radionuclide and dose scenario pair.

Equation 4-10

$$\frac{Conc_{ALARA}}{DCGL_W}_{i,j} = \frac{Cost_T}{V_{AD} \times P_D \times f_i \times DOSE_{AMCGi,j} \times F \times A} \times \frac{r+\lambda}{1-e^{-(r+\lambda)N}}$$

where $\frac{Conc_{ALARA}}{DCGL_W}_{i,j}$ = individual ratio of concentration to $DCGL_W$ for radionuclide *i* and scenario *j*

f_i = normalized radionuclide fraction for the Auxiliary Building for radionuclide *i*

$DOSE_{AMCGi,j}$ = averted dose to AMCG (rem) for radionuclide *i* and scenario *j*

Equation 4-11 is then used to calculate the final ratio of concentration to $DCGL_W$ that accounts for all radionuclides/scenario pairs.

Equation 4-11

$$\frac{Conc_{ALARA}}{DCGL_W} = \frac{1}{\sum \frac{1}{\left(\frac{Conc_{ALARA}}{DCGL_W}_{i,j}\right)}}$$

The total cost for the remedial action when divided by the total benefit of averted dose determines the cost effectiveness of the remedial action. Values greater than unity demonstrate that no further remediation is necessary beyond that required to meet the 25 mrem/year dose criterion and are ALARA. Values less than one provide the fraction of the 25 mrem/year dose criterion where it is necessary to remediate to achieve ALARA. Section 4.4.2.9.2 provides the conclusion of the ALARA analysis for basement structures.

4.4.2.2 Remedial Action Costs

The only structures that will remain as potential candidate surfaces for remediation are the Auxiliary Building, Turbine Building, Intake Structure, Circulating Water Tunnels, and the electrical vault beneath the Radwaste Processing Building slab. As discussed above, the vast majority of contamination to remain after the removal of Containment Building concrete will be in the floor of the Auxiliary Building basement.

The remediation techniques most likely to be implemented to perform this work are vacuuming, pressure washing and hand-wiping, concrete scabbling, or concrete shaving. As these efforts will occur prior to evaluating the remaining structural surfaces for acceptable concentrations of residual activity, this remediation action will not be evaluated for ALARA.

The remediation action evaluated for the ALARA analysis for the remediation of basement structures is scabbling the concrete surface of the floor of the Auxiliary Building basement. Volumetric concrete samples indicate that the majority of the radionuclide source inventory in the 971 feet and 989 feet AMSL elevations concrete floor resides within the first half-inch of concrete. For the purposes of the ALARA evaluation, it is conservatively assumed that 100% of the contamination resides in the first half-inch. In accordance with the guidance in Section G.3.1 of NUREG/CR-5884, Volume 2, “Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station” [9], one pass of scabbling is assumed to remove 0.125 inches (0.635 cm) of concrete. In accordance with calculations and position paper FC-20-006, “Fort Calhoun Station Building End State Concrete Surface Areas and Volumes” [10], the basement floor of the Auxiliary Building has a surface area of 3,134 m^2. This is the surface area which will be evaluated for the remediation cost determination.

4.4.2.2.1 Remediation Activity Rates

The remediation activity rates that were used for this evaluation were based on previous experience, from published literature, or from groups or vendors currently performing these or similar activities. Current project labor costs and past operational experience were also used in developing these rates.

In accordance with NUREG/CR-5884, an assumed crew size for performing concrete scabbling or shaving activities is three full-time laborers, a supervisor at a ¼-time involvement and an RP Technician, also at a ¼-time involvement. Using the current project labor rates for these positions of $66.78 per hour for a laborer, $90.00 per hour for a supervisor, and $55.59 per hour for an RPT, the hourly unit rate that will be used for the evaluation is $236.74.

Using the guidance found in NUREG/CR-5884, it is assumed that the concrete scabbling or shaving activity will remove approximately 0.125 inches of concrete per pass and the effective nominal removal rate is approximately 12.07 m^2 per hour. The ALARA evaluation assumes that 100% of the radioactive contamination resides within the first half-inch. Consequently, removing one half-inch of concrete over an assumed reference area of 3,134 m^2, scabbling at a nominal rate of 12.07 m^2 per hour to a depth of 0.125 inch per pass, equates to approximately 1,038.6 man-hours of work.

In accordance with NUREG/CR-5884, it is assumed that the actual remediation time in a typical eight-hour shift is 5.33 hours. To account for non-remediation work hours for work preparation, donning and removing protective clothing, and work breaks, the total man-hours were increased by a factor of 33% which equates to 1,381.3 man-hours. In addition, a contingency of 25% was added to the manpower hours. This equates to a total of 1,726.7 man-hours, which is multiplied times the hourly unit rate of $236.74 to equal the labor cost for this evaluation of $408,771.33.

4.4.2.2.2 __Equipment Costs__

Using the guidance found in NUREG/CR-5884, equipment costs are based on the rental of commercially available scabbing equipment, a compressor, a vacuum unit and consumables such as cutting bits, vacuum filters, and waste drums for containing waste debris. At 40-hours per work week, 1,726.7 man-hours equates to approximately 43 work-weeks. This evaluation assumes that two different commercially available concrete removal units will be procured, the Pentek Squirrel Scabbler & Vacuum System with a nominal rental rate of $685.00 per week and a Pentek Moose Scabbler & Vacuum System with a nominal rental rate of $950.00 per week. The compressor required for pneumatic equipment operation can be rented at a nominal rate of $115.00 per week. The cutting bits for the units are assumed to be replaced every 80 hours of operation, for an equivalent cost of about $13.00 per hour of operation. Additional costs include filter replacements at about $2.50 per hour of operation and waste drums for the collected debris. A 55-gallon drum holds approximately 7 ft^3 of waste and cost approximately $100.00 per drum. As it is assumed that the scabbing activity will generate approximately 39.8 m^3 of concrete waste, this will require the procurement of approximately 201 drums at a total cost of $20,100.00. The mobilization and demobilization costs associated with procuring this equipment would be approximately $2,200.00 per piece of equipment for a total of approximately $6,600.00. The total equipment costs assumed for this evaluation is approximately $128,610.00.

4.4.2.2.3 __Total Remediation Action Cost ($Cost_R$)__

For the evaluation of the remediation activity of concrete scabbling or shaving, the sum of the labor cost of $408,771.33 plus the equipment cost of $128,610.00 results in a total remediation action cost ($Cost_R$) for this activity of $537,381.33.

4.4.2.3 Transport and Disposal of the Waste ($Cost_{WD}$)

The cost of waste transport and disposal ($Cost_{WD}$) is expressed by Equation 4-4.

Disposal costs for generated waste were based on an average total disposal cost of \$2,046.94/m^3. Based upon an assumed waste volume of 39.8 m^3, a value of \$81,472.04 is calculated for the $Cost_{WD}$ variable.

4.4.2.4 Non-Radiological Risks ($Cost_{ACC}$)

The cost of non-radiological workplace accidents ($Cost_{ACC}$) was calculated using Equation N-5 of NUREG-1757, which is expressed as follows:

Equation 4-12

$$Cost_{ACC} = V_{SL} \times F_W \times T_A$$

where V_{SL} = monetary value of a fatality

F_W = workplace fatality rate in fatalities/hours worked

T_A = worker time required for remediation in units of worker-hours

In accordance with NUREG-1496, Appendix B, a value of 4.20E-08/hr is used for F_W. For T_A, in accordance with NUREG-1757, the same hours that were determined for labor cost (1,726.7 man-hours) were used for worker accident cost. In accordance with NUREG-1530, a value of \$9,000,000.00 is selected for V_{SL}. Subsequently, a value of \$652.69 is calculated for the $Cost_{ACC}$ variable.

4.4.2.5 Transportation Risks ($Cost_{TF}$)

The cost of traffic fatalities incurred during the transportation of waste ($Cost_{TF}$) is expressed by Equation 4-5.

For this evaluation, the waste volume (V_A) is assumed to be 39.8 m^3, and the haul volume of an overland truck shipment per NUREG-1757 is assumed to be 13.6 m^3 (V_{SHIP}). In accordance with NUREG-1496, a value of 3.80E-08/hr is used for F_T. In accordance with NUREG-1530, a value of \$9,000,000.00 is selected for V_{SL}. 3,290 km is used for D_T.

For this evaluation, the value for the $Cost_{TF}$ variable is \$3,292.81.

4.4.2.6 Worker Dose Estimates ($Cost_{WDose}$)

The cost of remediation worker dose ($Cost_{WDose}$) was calculated using Equation N-7 of NUREG-1757, which is expressed as follows:

Equation 4-13

$$Cost_{WDose} = V_{AD} \times D_R \times T$$

where V_{AD} = value of incurred dose, which is a conversion factor for the monetary value of radiation dose (dollars per person-rem)

D_R = total effective dose equivalent rate to remediation workers in units of rem/hour

T = time worked (site labor) to remediate the area in units of person-hour

Costs associated with worker dose are a function of the hours worked and the workers' radiation exposure for the task. A value of 3 mrem per man-hour was used for D_R. This assumes that a majority of the source inventory will be removed prior to performing the concrete scabbling or shaving activity. The time worked to remediate the area in units of person-hour calculated for this activity (T) was 1,726.7 man-hours. For this evaluation, the value for the $Cost_{WDose}$ variable is $26,418.30.

4.4.2.7 Monetary Cost of Dose to the Public ($Cost_{PDose}$)

The cost of remediation worker dose ($Cost_{PDose}$) was calculated using Equation N-7 of NUREG-1757, which is expressed as follows:

Equation 4-14

$$Cost_{PDose} = V_{AD} \times D_R \times T$$

where V_{AD} = value of incurred dose, which is a conversion factor for the monetary value of radiation dose (dollars per person-rem)

D_R = total effective dose equivalent rate to public in units of rem/hour

T = time spent near waste shipments in parking lots in units of person-hour

For this equation, a worst-case value of 0.5 mrem/hr was used for D_R. This assumes that the shipment is classified as Limited Specific Activity in accordance with 49 CFR 173.427 and that the package meets the specific administrative limit of 0.5 mrem/hr on the exterior of the shipment. The exposure time (T) used for this calculation is based upon a transit time of 15 hours driving from FCS to the disposal site in Clive, Utah, times three shipments, for a total of 45 hours. For this evaluation, the value for the $Cost_{PDose}$ variable is $1,147.50.

4.4.2.8 Total Cost ($Cost_T$)

The total cost ($Cost_T$) assumed for this evaluation is $650,364.65.

4.4.2.9 Residual Radioactivity in Basement Structures that are ALARA

The following parameters were used for performing the ALARA calculation using the equation from NUREG-1757 presented in Section 4.4.2.1:

- Population density (P_D) for the critical group: 0.0004 person/m^2 (per NUREG-1496)
- Fraction of residual radioactivity removed by the remedial action (*F*): 1 (Removal of desired concrete volume is assumed 100% effective)
- Area (*A*) used to calculate the population density:
    - 10,000 m^2 for basement fill model (BFM) *insitu* scenario in order to calculate the family of four assumed in the calculation of well water usage (for the resident farmer critical group)
    - 0.566 m^2 for BFM drilling spoils scenario
    - 1,584 m^2 for BFM excavation scenario (the volume of concrete in the Auxiliary Building basement walls spread over a 1 meter lift)
- Monetary discount rate (*r*): 0.00 yr^{-1}
- Number of years (*N*) over which the collective averted dose is calculated: 1,000 years (per NUREG-1496)

4.4.2.9.1 Radionuclides Considered for ALARA Analysis

The radionuclide mixture for contaminated concrete developed in technical support document 21-043, "Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan" [11] was used for the ALARA analysis. The DCGLs for the Auxiliary Building for each individual ROC from Table 5-4 in Chapter 5 were used for the calculation of f_i. DCGLs, in units of pCi/m^2, of basement surface areas, are presented in Chapter 6 for the BFM *insitu*, BFM drilling spoils, and BFM excavation scenarios individually. The DCGLs for the Auxiliary Building, adjusted for insignificant dose contributors, are presented in Table 4-2. The values for half-life, radiological decay constants (λ), and the radionuclide mixture fractions for the Auxiliary Building are presented in Table 4-3.

Table 4-2 Basement DCGLs for the Auxiliary Building

Radionuclide	BFM Walls/Floors DCGL (pCi/m^2)	*insitu* Scenario DCGL (pCi/m^2)	Drilling Spoils Scenario DCGL (pCi/m^2)	Excavation Scenario DCGL (pCi/m^2)
C-14	8.20E+06	2.11E+07	1.43E+14	1.23E+07
Co-60	3.40E+06	2.58E+07	3.02E+08	3.73E+06
Cs-137	6.90E+06	2.28E+07	1.22E+09	9.24E+06
Eu-152	9.62E+06	8.26E+08	6.40E+08	9.35E+06
Sr-90	8.23E+05	1.24E+06	6.36E+10	2.09E+06

Table 4-3 Radionuclide Half-Lifes, Decay Contants, and Mixture Fractions

Radionuclide	Half-Life (years)	λ (yr^{-1})	Mixture Fraction (Auxiliary Building)
C-14	5.70E+03	1.22E-04	8.27E-02
Co-60	5.27E+00	1.31E-01	1.85E-02
Cs-137	3.02E+01	2.29E-02	8.94E-01
Eu-152	1.35E+01	5.13E-02	1.15E-03
Sr-90	2.88E+01	2.41E-02	4.04E-03

The ALARA calculation was performed in three parts, each representing one of the three BFM dose scenarios. Three dose values were required to accurately calculate the averted dose, as opposed to using the summed DCGL (BFM Walls/Floors), because each scenario is applicable to a different area, as detailed in Section 4.4.2.9.

The actual dose from each scenario, assuming a summation of the dose from each scenario equals 25 mrem/year, is presented in Table 4-4. Therefore, the dose values for each ROC from Table 4-4 were used to derive the $DOSE_{AMCG}$ variable in Equation 4-10 for each scenario.

Table 4-4 Dose for Individual Scenarios ($DOSE_{AMCG}$)

Radionuclide	*insitu* (rem/yr)	Drilling Spoils (rem/yr)	Excavation (rem/yr)
C-14	0.0097	0.0000	0.0167
Co-60	0.0033	0.0003	0.0228
Cs-137	0.0076	0.0001	0.0187
Eu-152	0.0003	0.0004	0.0257
Ni-63	0.0165	0.0000	0.0098
Sr-90	0.0097	0.0000	0.0167

4.4.2.9.2 <u>ALARA Calculation and Conclusion</u>

The ALARA calculations performed to evaluate the concrete scabbling or shaving remediation activity is presented in Table 4-5 for the Auxiliary Building basement floor. The ALARA analysis shows that further remediation of concrete beyond that required to demonstrate compliance with the 25 mrem/year dose criterion is not justified.

Table 4-5 ALARA Analysis for the Auxiliary Building Basement

insitu scenario

A	=	10000 m^2
V_{AD}	=	$ 5,100.00
r	=	0 yr^{-1}
N	=	1000 years
P_D	=	0.0004 person/m^2
F	=	1
$Cost_T$	=	$ 650,364.65

Nuclide	Half Life (yrs)	λ	$1-e^{-(r+\lambda)N}$	f_i	$DOSE_{AMCG}$	Individual Ratio	1/Individual Ratio
C-14	5.70E+03	1.22E-04	1.14E-01	8.27E-02	9.73E-03	4.21E+01	2.38E-02
Co-60	5.27E+00	1.31E-01	1.00E+00	1.85E-02	3.29E-03	6.88E+04	1.45E-05
Cs-137	3.02E+01	2.29E-02	1.00E+00	8.94E-01	7.57E-03	1.08E+02	9.25E-03
Eu-152	1.35E+01	5.13E-02	1.00E+00	1.15E-03	2.91E-04	4.89E+06	2.05E-07
Sr-90	2.88E+01	2.41E-02	1.00E+00	4.04E-03	1.65E-02	1.15E+04	8.71E-05

drilling spoils scenario

A	=	0.566 m^2
V_{AD}	=	$ 5,100.00
r	=	0 yr^{-1}
N	=	1000 years
P_D	=	0.0004 person/m^2
F	=	1
$Cost_T$	=	$ 650,364.65

Nuclide	Half Life (yrs)	λ	$1-e^{-(r+\lambda)N}$	f_i	$DOSE_{AMCG}$	Individual Ratio	1/Individual Ratio
C-14	5.70E+03	1.22E-04	1.14E-01	8.27E-02	1.44E-09	5.04E+12	1.98E-13
Co-60	5.27E+00	1.31E-01	1.00E+00	1.85E-02	2.81E-04	1.42E+10	7.02E-11
Cs-137	3.02E+01	2.29E-02	1.00E+00	8.94E-01	1.41E-04	1.03E+08	9.74E-09
Eu-152	1.35E+01	5.13E-02	1.00E+00	1.15E-03	3.76E-04	6.69E+10	1.50E-11
Sr-90	2.88E+01	2.41E-02	1.00E+00	4.04E-03	3.24E-07	1.04E+13	9.65E-14

excavation scenario

A	=	1584 m^2
V_{AD}	=	$ 5,100.00
r	=	0 yr^{-1}
N	=	1000 years
P_D	=	0.0004 person/m^2
F	=	1
$Cost_T$	=	$ 650,364.65

Nuclide	Half Life (yrs)	λ	$1-e^{-(r+\lambda)N}$	f_i	$DOSE_{AMCG}$	Individual Ratio	1/Individual Ratio
C-14	5.70E+03	1.22E-04	1.14E-01	8.27E-02	1.67E-02	1.55E+02	6.45E-03
Co-60	5.27E+00	1.31E-01	1.00E+00	1.85E-02	2.28E-02	6.27E+04	1.59E-05
Cs-137	3.02E+01	2.29E-02	1.00E+00	8.94E-01	1.87E-02	2.77E+02	3.61E-03
Eu-152	1.35E+01	5.13E-02	1.00E+00	1.15E-03	2.57E-02	3.49E+05	2.86E-06
Sr-90	2.88E+01	2.41E-02	1.00E+00	4.04E-03	9.85E-03	1.22E+05	8.22E-06

$Conc_{ALARA}/DCGL_W$ = 23.14

4.5 References

[1] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.187, Revision 2, Guidance for Implementation of 10 CFR 50.59 Changes, Tests and Experiments," 2020.

[2] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.179, Revision 2, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," 2019.

[3] Omaha Public Power District, Fort Calhoun Station Unit 1, “Defueled Safety Analysis Report”.

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

[5] U.S. Nuclear Regulatory Commission, "NUREG-1757, Volume 2, Revision 2, Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria - Draft Report for Comment," 2020.

[6] U.S. Nuclear Regulatory Commission, "NUREG-1496, Volume 2, Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities," 1997.

[7] U.S. Nuclear Regulatory Commission, "NUREG/BR-0058, Revision 4, Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," 2004.

[8] U.S. Nuclear Regulatory Commission, "NUREG-1530, Revision 1, Reassessment of NRC's Dollar Per Person-Rem Conversion Factor Policy - Draft Report for Comment," 2015.

[9] U.S. Nuclear Regulatory Commission, "NUREG/CR-5884, Volume 2, Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station," 1995.

[10] Omaha Public Power District, "FC-20-006, Fort Calhoun Station Building End State Concrete Surface Areas and Volumes".

[11] Radiation Safety and Control Services, "TSD 21-043, Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan," 2021.

FORT CALHOUN STATION DECOMMISSIONING PROJECT
LICENSE TERMINATION PLAN

CHAPTER 5
FINAL STATUS SURVEY PLAN

REVISION 0

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ABBREVIATIONS

AB	Auxiliary Building
AF	area factor
ALARA	as low as is reasonably achievable
AMCG	average member of the critical group
AMSL	above mean sea level
BFM	basement fill model
BOP	balance of plant
CoC	chain-of-custody
CPP	calculations and position paper
CsI	cesium iodide
DA	deconstruction area
DCGL	derived concentration guideline level
DQA	data quality assessment
DQO	data quality objectives
DRP	discrete radioactive particle
DSR	dose to source ratio
EMC	elevated measurement comparison
ETD	easy-to-detect
FCS	Fort Calhoun Station
FSS	final status survey
GPS	Global Positioning System
HPGe	high-purity germanium
HSA	Historical Site Assessment
HTD	hard-to-detect
IC	insignificant contributor
ISFSI	Independent Spent Fuel Storage Installation
ISOCS	In Situ Object Counting System
LBGR	lower bound of the gray region
LT	license termination
LTP	license termination plan

MARLAP	Multi-Agency Radiological Laboratory Analytical Protocols Manual
MARSAME	Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
MDCR	minimum detectable count rate
NAD	North American Datum
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
OCA	Owner Controlled Area
OPPD	Omaha Public Power District
QA	quality assurance
QC	quality control
RA	radiological assessment
RASS	remedial action support survey
RESRAD	RESidual RADioactive
ROC	radionuclides of concern
RWPB	Radioactive Waste Processing Building
SOF	sum of fractions
SFP	Spent Fuel Pool
TB	Turbine Building
TEDE	total effective dose equivalent
TSD	technical support document
UBGR	upper bound of the gray region
VSP	Visual Sample Plan

5 FINAL STATUS SURVEY PLAN

5.1 Introduction

5.1.1 Purpose

The purpose of the final status survey (FSS) Plan is to describe the methods to be used in planning, designing, implementing, and evaluating the FSS at the Fort Calhoun Station (FCS) decommissioning project. The FSS Plan describes the final radiation survey process used to demonstrate that the FCS site complies with the radiological criteria for unrestricted use specified in 10 CFR 20.1402. U.S. Nuclear Regulatory Commission (NRC) regulations applicable to FSS are found in 10 CFR 50.82(a)(9)(ii)(D) and 10 CFR 20.1501(a) and (b).

The two radiological criteria for unrestricted use specified in 10 CFR 20.1402 are: 1) the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent (TEDE) to an average member of the critical group (AMCG) that does not exceed 25 millirem/year (mrem/year), including that from groundwater sources of drinking water, and 2) the residual radioactivity has been reduced to levels that are as low as is reasonably achievable (ALARA).

5.1.2 Regulatory Requirements and Industry Guidance

This FSS Plan has been developed using the guidance contained in the following documents:

- NUREG-1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM) [1]
- NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys" [2]
- NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions" [3]
- NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans" [4]
- NUREG-1757, Volume 2, Revision 2, "Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria - Draft Report for Comment" [5]
- Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" [6]
- NUREG-1575, Supplement 1, "Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual" (MARSAME) [7]
- NUREG-1576, "Multi-Agency Radiological Laboratory Analytical Protocols Manual" (MARLAP) [8]

5.1.3 Scope

The FSS Plan includes the radiological assessment of impacted structures, systems and land areas that will remain following decommissioning (the "End State"). Impacted areas are defined as areas with a reasonable possibility of containing residual radioactivity in excess of natural background or fallout levels. It is Omaha Public Power District's (OPPD) intention to release for unrestricted use the impacted open land areas, remaining below-ground basements, above-ground buildings, and buried piping from the 10 CFR 50 license. The adjacent areas that were classified as non-impacted and previously released from the 10 CFR Part 50 license, will not be subject to FSS. The Independent Spent Fuel Storage Installation (ISFSI), including the ISFSI Operations Facility, which will still remain a licensed area, will also not be subject to FSS. In addition, Section 5.4.1.10 addresses the plan for the assessment of groundwater.

The end state is defined as the configuration of basements (including embedded piping and penetrations), above-ground buildings, buried piping, and open land areas that will remain at the time of license termination. Chapter 3 of the License Termination Plan (LTP), Figure 3-1, provides the location of the remaining basements, above-ground buildings, and buried piping that will remain at the time of license termination. Chapters 3 and 6 of the LTP provide a more detailed discussion of the FCS end state.

The following above-ground buildings will remain at the time of license termination: Training Building, FLEX Building, Owner Controlled Area (OCA) Entrance Building, Switchyard 3451 Old Building, Switchyard 3451 New Building, and the 1251 Control and Switchgear Building. With the exception of the removal of the 345kv and 161kv lines to the station, the Switchyard will remain after decommissioning. All other structures will be removed to a minimum of three feet below grade (approximately 1,001 feet above mean sea level or AMSL). Because there are no dose model implications, if the number and identification of above-ground buildings to remain and be subject to FSS changes, the NRC will be notified but an LTP revision will not be required.

The basements of the Turbine Building (TB) (including the Circulating Water Discharge Tunnel), Containment Building, Auxiliary Building (AB), and Intake Structure (including the Circulating Water Intake Tunnel) will remain with all interior walls removed, with the exception of the Turbine Building where the turbine pedestals will remain up to three feet below grade. All other structures will be removed in their entirety with the exception of the above-ground buildings and the Switchyard. Once decontamination and dismantlement activities are completed, as well as subsequent FSS, each basement will be backfilled with fill material to grade level (approximately 1,004 feet AMSL). The concrete surface areas, volumes and void spaces for the basements are provided in OPPD calculations and position paper (CPP) FC-21-006, "Fort Calhoun Station End State Concrete Surface Areas and Volumes" [9].

The end state will also include a range of buried pipe, embedded pipe, and penetrations. Buried piping is defined as pipe that runs through soil. An embedded pipe is defined as a pipe that runs vertically through a concrete wall or horizontally through a concrete floor and is contained within a given building. A penetration is defined as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end. The inventory of buried pipe, embedded pipe, and penetrations to remain is provided in OPPD CPP FC-21-002, "Description of Embedded Piping, Penetrations, and Buried Pipe to Remain in Fort Calhoun End State" [10]. The list of end-state embedded pipe, buried pipe, and penetrations presented in FC-21-002 is intended to be a bounding end-state condition. No embedded pipe, buried pipe, or penetrations that are not listed in FC-21-002 will be added to the end-state condition; however, embedded pipe, buried pipe, or penetrations can be removed from the list and disposed of as waste without a revision to this LTP because the derived concentration guideline levels (DCGLs) will not increase. If embedded pipe, buried pipe, or penetrations are removed from the list, the NRC will be notified and the CPP will be revised.

5.1.4 Summary of FSS Process

The FSS Plan contained in this chapter will be used as the basis for developing FSS procedures and applying existing procedures to the FSS process. All processes associated with FSS will be conducted in accordance with approved procedures.

The primary objectives of the FSS are:

- verify survey unit classification
- demonstrate that the potential dose from residual radioactivity in each survey unit is below the release criterion
- demonstrate that the potential dose from small areas of elevated activity, when combined with other residual activity in a survey unit, is below the release criterion

The FSS process consists of four principal elements:

- planning
- design
- implementation
- assessment

The data quality objective (DQO) and data quality assessment (DQA) processes are applied to these four principal elements. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions (as is the case in FSS). The DQA process is an evaluation method used during the assessment phase of the FSS to ensure the validity of survey results and demonstrate achievement of the sampling plan objectives (e.g., to demonstrate compliance with the release criterion in a survey unit).

5.1.4.1 Summary of Survey Planning

Survey planning includes a review of the "Historical Site Assessment for Fort Calhoun Station" [11] (HSA) and other pertinent characterization information to establish the radionuclides of concern (ROC) and the survey unit classifications. Survey units are fundamental elements for which final status surveys are designed and implemented. Chapter 2 of the LTP provides a list of all survey units that will undergo FSS, as well as figures that delineate the survey unit locations and boundaries.

Before the survey process can proceed to the design phase, concentration levels that represent the maximum annual dose criterion of 10 CFR 20.1402 must be established. These concentrations are established for either surface contamination or volumetric contamination. They are used in the survey design process to establish the minimum sensitivities required for the survey instruments and techniques, and in some cases, the spacing of fixed measurements or samples to be made within the survey unit. Surface or volumetric concentrations corresponding to the dose criterion are referred to as DCGLs. A DCGL for the average residual radioactivity in a survey unit is called a $DCGL_W$. Values of the $DCGL_W$ may then be increased through the use of area factors (AF) to obtain a DCGL that represents the same dose to an individual for residual radioactivity over a smaller area within a survey unit. This scaled value is called the $DCGL_{EMC}$, where EMC stands for elevated measurement comparison.

In areas where remediation is required, a remedial action support survey (RASS) will be performed to confirm that remediation was successful prior to initiating FSS activities. Radiological assessments (RA) or turnover surveys for areas not requiring remediation, may be performed to verify the area is suitable for FSS. The results of RASS and RA will be documented in the applicable FSS release records.

Prior to implementation of FSS, isolation and control methods are established in survey units to ensure that radioactive material is not reintroduced into the area from ongoing decommissioning activities and to maintain the “as-left” radiological and physical conditions of the area. An inspection of the survey unit is performed to ensure the area is suitable for turnover to perform FSS, and to identify any safety hazards. Survey planning is further discussed in Section 5.2.

5.1.4.2 Summary of Survey Design

The survey design process establishes the methods and performance criteria used to conduct the FSS. Survey design assumptions are documented in sample plans in accordance with approved procedures. The survey design selects the appropriate survey instruments and techniques to provide adequate coverage of the survey unit through a combination of scans, fixed measurements, and sampling. This process ensures that data of sufficient quantity and quality are obtained to make decisions regarding the suitability of the survey design assumptions and whether the survey unit meets the release criterion. Approved procedures will direct this process to ensure consistent implementation and adherence to applicable requirements.

The general approach prescribed by MARSSIM for FSS requires that at least a minimum number of measurements or samples be taken within a survey unit so that the nonparametric statistical tests used for data assessment can be applied with adequate confidence. The Sign test is the most appropriate test for FSS at FCS, as background is expected to constitute a small fraction of the $DCGL_W$ based on the results of characterization surveys. Consequently, the Sign test will be applied when demonstrating compliance with the unrestricted release criterion. Section 5.2.4 provides additional discussion on media-specific background and area ambient background.

The level of survey effort required for a given survey unit is determined by the potential for contamination as indicated by its classification. The percent coverage for scan surveys is determined in accordance with Section 5.3.4. The number and location of surface measurements and volumetric samples are established in accordance with Sections 5.3.1 and 5.3.5. Investigation levels are established in accordance with Section 5.5.5.

A survey map is prepared for each survey unit that depicts the locations of scan grids, as well as random, systematic or judgmental/investigative measurement and sample locations.

The appropriate instruments and detectors, instrument operating modes, and survey methods used to collect and analyze data are also specified.

Quality control (QC) measures will be specified to obtain quantitative information to demonstrate that measurement and sample results have the required precision and are sufficiently free of errors to accurately represent the area being investigated. The minimum requirements for QC measurements/samples are established in accordance with Section 5.6.

5.1.4.3 Summary of Survey Implementation

After preparation of the sample plan, the FSS data are collected. Trained and qualified personnel will perform the necessary measurements using calibrated instruments in accordance with approved procedures. Section 5.4.1 addresses data collection requirements. The FSS process and instrumentation described in this plan adhere to the guidance of MARSSIM. However, advanced survey technologies may be used to conduct radiological surveys that can scan the surface and record the results. This plan allows for the use of these advanced technologies, where survey quality and efficiency can be increased, as long as the survey results are at least equivalent, in terms of their statistical significance, to those that would have been obtained using the nonparametric sampling methods of MARSSIM. In cases where advanced survey technologies are to be used, a technical evaluation will be developed to describe the technology to be used and to demonstrate how the technology meets the objectives of the survey. These technical evaluations will be referenced, as appropriate, in FSS reports and will be available for NRC review. Notification will be made to the NRC prior to the use of advanced instruments or technology.

Prior to implementing field activities, a pre-job briefing is held with the survey and support team during which the survey instructions are reviewed and additional survey unit considerations are discussed (e.g., safety hazards and mitigations).

The survey team gather instruments and equipment as indicated and perform surveys in accordance with the appropriate procedures and the instructions provided in the sample plan. Daily pre- and post-use source checks are performed on all instruments when used. Technicians are responsible for documenting survey results and maintaining custody of samples and instrumentation. At the completion of surveys, technicians download survey data from the instruments and prepare samples for analysis.

5.1.4.4 Summary of Data Assessment

Survey results are converted to appropriate units (i.e., either dpm/100 cm^2, pCi/g, or pCi/m^2) and compared to investigation levels to determine appropriate follow-up action, as necessary.

The DQA approach is applied to FSS results to ensure their validity and to demonstrate that the objectives of the FSS are met. Data assessment includes data verification and validation, review of survey design basis and data analysis. For a given survey unit, the survey data are evaluated to determine if the residual activity levels meet the applicable release criterion and if any areas of elevated activity exist. Measurements exceeding investigation levels will be verified and investigated and, following confirmatory measurements, the affected area may be remediated and/or re-classified and a re-survey performed consistent with the guidance in MARSSIM and commensurate with the classification and extent of contamination. Section 5.5 provides a detailed discussion of data assessment.

5.2 Final Status Survey Planning

5.2.1 Data Quality Objectives

The DQO process will be incorporated as an integral component of the data life cycle, and is used in the planning phase for characterization, remediation and FSS plan development using a graded approach. Survey plans that are complex or that have a higher level of risk associated with an incorrect decision (such as FSS) require significantly more effort to develop the plan than, for example, a survey plan used to obtain characterization data. The DQO process includes a series of planning steps found to be effective in establishing criteria for data quality and developing survey plans. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and alternate actions defined. Furthermore, the DQO process is flexible in that the level of effort associated with planning a survey is based on the complexity of the survey and nature of the hazards. The DQO process is iterative, allowing the survey planning team to incorporate new information and modify the output of previous steps to act as input to subsequent steps. The appropriate design for a given survey will be developed using the DQO process as outlined in Appendix D of MARSSIM. The seven steps of the DQO process are outlined in the following sections.

5.2.1.1 State the Problem

The first step of the planning process consists of defining the problem. This step provides a clear description of the problem, identification of planning team members (especially the decision-makers), a conceptual model of the hazard to be investigated and the estimated resources. The problem associated with FSS is to determine whether a given survey unit meets the radiological release criterion of 10 CFR 20.1402.

5.2.1.2 Identify the Decision

This step of the DQO process consists of developing a decision statement based on a principal study question (i.e., the stated problem) and determining alternative actions that may be taken based on the answer to the principal study question. Alternative actions identify those measures to resolve the problem. The decision statement combines the principal study question and alternative actions into an expression of choice among multiple actions. For the FSS, the principal study question is “does residual radioactive contamination that is present in the survey unit exceed the established $DCGL_W$ values?” The alternative actions can include no action, investigation, resurvey, remediation, and reclassification.

Based on the principal study question and alternative actions listed above, the decision statement for the FSS is to determine whether or not the average radioactivity concentration for a survey unit results in a sum of fractions (SOF) less than unity.

5.2.1.3 Identify Inputs to the Decision

The information required depends on the type of media under consideration (e.g., soil, structure) and whether existing data are sufficient or new data are needed to make the decision. If the decision can be based on existing data, then the source or sources will be documented and evaluated to ensure reasonable confidence that the data are acceptable. If new data are needed, then the type of measurement (e.g., scan, direct measurement and sampling) will need to be determined.

Sampling methods, sample quantity, sample matrix, types of analyses and analytic and measurement process performance criteria, including detection limits, are established to ensure adequate sensitivity relative to the release criteria.

The following information will be used to support the decision:

- ROC
- DCGLs
- use of surrogate relationships to infer hard-to-detect (HTD) ROC
- minimum detectable concentrations (MDC)
- measurement and sampling results

5.2.1.4 Define the Study Boundaries

This step of the DQO process includes identification of the target population of interest, the spatial and temporal features of the population pertinent to the decision, time frame for collecting the data, practical constraints and the scale of decision making. In FSS, the target population is the set of samples or direct measurements that constitute an area of interest (i.e., the survey unit). The medium of interest (e.g., soil, structure) is specified during the planning process. The spatial boundaries include the entire area of interest including soil depth, area dimensions, contained water bodies and natural boundaries, as needed. Temporal boundaries include those activities impacted by time-related events including weather conditions, seasons, operation of equipment under different environmental conditions, resource loading and work schedule.

5.2.1.5 Develop a Decision Rule

This step of the DQO process develops the binary statement that defines a logical process for choosing among alternative actions. The decision rule is a clear statement using the “If...then...” format and includes action level conditions and the statistical parameter of interest (e.g., mean of data). Decision statements can become complex depending on the objectives of the survey and the radiological characteristics of the affected area. For FSS, the decision rule will be based on the question pertaining to whether or not the radioactivity concentration of residual radioactivity in a survey unit exceeds the applicable DCGL value.

- If the SOF is less than unity (1), then no additional investigation will be performed and the survey unit meets the criteria for unrestricted release.
- If the SOF is greater than or equal to unity (1), then the survey unit does not meet the criteria for unrestricted release. Additional remediation followed by FSS redesign and resurvey will be performed.

5.2.1.6 Specify Limits on Decision Errors

This step of the DQO process incorporates hypothesis testing and probabilistic sampling distributions to control decision errors during data analysis. Hypothesis testing is a process based on the scientific method that compares a baseline condition to an alternate condition. The baseline condition is technically known as the null hypothesis. Hypothesis testing rests on the premise that the null hypothesis is true and that sufficient evidence must be provided for rejection. The primary consideration during FSS will be demonstrating compliance with the release criterion. For FSS, the null hypothesis is expressed as “the survey unit exceeds the criteria for unrestricted release.”

Decision errors occur when the data set leads the decision-maker to make false rejections or false acceptances during hypothesis testing. For the design of FSS at FCS, the α error (Type I error) will always be set at 0.05 (5 percent) unless prior NRC approval is granted for using a less restrictive value. The β error (Type II error) will also be initially set at 0.05 (5 percent). However, the Type II error may be adjusted with the concurrence of the License Termination (LT)/FSS Manager, after weighing the resulting change in the number of required sample or measurement locations against the risk of unnecessarily investigating and/or remediating survey units that are truly below the release criterion.

Another output of this step is assigning probability limits to points above and below the gray region where the consequences of decision errors are considered acceptable. The upper bound corresponds to the release criteria. The lower bound of the gray region (LBGR) is determined as another limit on decision error. LBGR is influenced by a parameter known as the relative shift. The relative shift is the $DCGL_W$ minus the LBGR (i.e., the width of the gray region) divided by the standard deviation of the data set used to design the survey. In accordance with NUREG-1757, Appendix A, the LBGR should be set at the mean concentration of residual radioactivity that is estimated to be present in the survey unit. However, if no other information is available regarding the survey unit, the LBGR may be initially set equal to 0.5 times the applicable $DCGL_W$. If the relative shift exceeds a value of 3, then the LBGR should be adjusted until the relative shift value is equal to 3. The adjustment of decision errors is discussed in more detail in Section 5.3.1.

Sample uncertainty is controlled by collecting a small frequency of additional samples from each survey unit. Analytical uncertainty is controlled by using appropriate instrumentation, methods, techniques, training, and QC. The MDC values for individual radionuclides using specific analytical methods will be established. Uncertainty in the decision to release areas for unrestricted use is controlled by the number of samples and/or measurement points in each survey unit and the uncertainty in the estimate of the mean radionuclide or gross radioactivity concentrations. The specific types of instruments that can be used for the FSS of FCS and their respective MDC values are presented in Section 5.4.2, Table 5-21 and Table 5-22.

Graphing the probability that a survey unit does not meet the release criteria may be used during FSS. This graph, known as a power curve, can be performed retrospectively (i.e., after FSS) using actual measurement data. This retrospective power curve is a tool that can be used to demonstrate that the DQOs are met when the null hypothesis is not rejected (i.e., the survey unit does not meet the release criteria).

5.2.1.7 Optimize Design for Obtaining Data

The first six steps of the DQO process develop the performance goals of the survey. This final step in the DQO process leads to the development of an adequate survey design.

By using an on-site analytical laboratory, sampling and analyses processes are designed to provide near real-time data assessment during implementation of field activities and FSS. Gamma scans provide information on soil areas that have residual radioactivity greater than background and allow appropriate selection of biased sampling and measurement locations. This data will be evaluated and used to refine the scope of field activities to optimize implementation of the FSS design and ensure the DQOs are met.

5.2.2 Classification of Survey Units

The adequacy of the FSS process rests upon partitioning the site into properly classified survey units of appropriate physical area. Chapter 2 of the LTP discusses in detail the HSA for the site and the classification assigned to all of the site structures, piping and land areas. Because characterization is an ongoing effort throughout the decommissioning process, survey unit classifications may be modified on the basis of new characterization information or impacts from decommissioning activities. The process described in LTP Section 1.6 will be used to evaluate such modifications in order to determine whether notification to the NRC is required. In general, if a modification decreases a survey unit area classification (i.e., impacted to non-impacted, Class 1 to Class 2, Class 1 to Class 3, or Class 2 to Class 3), the NRC must be notified a minimum of 14 days prior to implementing the change in classification.

A survey unit is a geographical area consisting of structures, land areas, or buried piping of specified size and shape for which a separate decision will be made whether the survey unit meets the radiological release criterion. Survey units are contiguous site areas (with minor exceptions) with a similar operational history and the same classification of contamination potential. Survey units are established to facilitate the survey process and the statistical analysis of survey data.

Impacted areas are areas that may have been impacted by past site operations and have a reasonable possibility of containing residual radioactivity in excess of natural background or fallout levels. For the purposes of license termination, all land areas at FCS, along with the structures, systems and components within them, are considered impacted. Based on the levels of residual radioactivity present, impacted areas have been divided into Class 1, Class 2, or Class 3 designations using the following guidance from MARSSIM:

- Class 1 areas are impacted areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys) above the $DCGL_W$, and there is insufficient evidence to support reclassification as Class 2 or Class 3.
- Class 2 areas are impacted areas that have, or had prior to remediation, a potential for contamination or known contamination, but are not expected to exceed the $DCGL_W$ and little or no potential for small areas of elevated activity.
- Class 3 areas are not expected to contain residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$, based on site operating history and previous radiological survey, and have little or no potential for small areas of elevated activity. Previous remediation would preclude a survey unit from having a Class 3 designation.

Survey units are limited in size based on classification, exposure pathway modeling assumptions, and site-specific conditions. The surface area limits, used in establishing the initial set of survey units, are provided in Table 5-1 for structures and land areas. The area limits for above-ground structures refer to the floor area, and not the total surface area (floors plus walls and ceilings). This is consistent with the guidance in Table A.1 of Appendix A to NUREG-1757 and MARSSIM. The surface area limits given in Table 5-1 will also be used should the need arise to establish any new survey units beyond the initial set given in Chapter 2.

Table 5-1 FCS Survey Unit Surface Area Limits

Survey Unit Type/Classification	Suggested Surface Area Limit
Class 1:	
Above-ground Structures (floor area)	≤ 100 m^2
Land Areas	≤ 2,000 m^2
Basements (all surfaces combined)	no limit
Class 2:	
Above-ground Structures (floor area)	100 to 1,000 m^2
Land Areas	2,000 m^2 to 10,000 m^2
Basements (all surfaces combined)	no limit
Class 3:	
Above-ground Structures (floor area)	no limit
Land Areas	no limit
Basements (all surfaces combined)	no limit

As indicated in Chapter 2, impacted areas of the FCS site have been divided into survey units to facilitate survey design and to manage the survey. The survey units designated for structures below 1,001 feet AMSL were based on screening values and source term assumptions that are significantly different from the basement fill model (BFM) and are therefore not applicable. Structure surfaces below 1,001 feet AMSL may be combined into one single survey unit for each building basement.

5.2.3 Area Preparation

5.2.3.1 Isolation and Control

Before FSS activities can begin in an area, a transition must occur where planned decommissioning activities are completed and the survey unit is subsequently assessed to verify it is suitable for FSS. The determination of readiness for controls and the preparation for FSS will be based on the results of characterization, RAs, and/or RASS that indicate residual radioactivity is unlikely to exceed the applicable DCGLs in the respective survey unit. Following the conclusion of remediation activities and prior to initiating FSS, isolation and control measures will be implemented. Isolation and control methods are established in survey units to ensure that radioactive material is not reintroduced into the area from ongoing decommissioning activities and to maintain the as-left radiological and physical conditions of the area.

Prior to transitioning a survey unit from decommissioning activities to isolation and control, a walkdown will be performed to identify access requirements and to specify the required isolation and control measures. The physical condition of the area will also be assessed, with any conditions that could interfere with FSS activities identified and addressed. If any support equipment needed for FSS activities, such as ladders or scaffolding, are in place, it will be evaluated to ensure that it does not pose the potential for introducing radioactive material into the area. Industrial safety and work practice issues, such as access to high areas or confined spaces, will also be identified during the pre-survey evaluation. In some instances, turnover surveys (in the form of an RA) may be performed to verify that an area is ready for FSS.

The following criteria must be met for an area to be deemed ready for isolation and control:

- Planned decommissioning activities in the area are complete.
- Planned decommissioning activities in areas either adjacent to the area to be isolated or that could otherwise affect it, are either complete or are deemed not to have any reasonable potential to spread plant-related radioactive material to the area.
- Tools and equipment that are not needed to support FSS activities and could interfere with FSS activities, are removed.
- Equipment to be used in support of FSS activities is evaluated to ensure it does not pose the potential for introducing plant-related radioactive material into the area.
- Where practical, transit paths to or through the area, except those required to support FSS activities, are eliminated or re-routed.

These measures will consist of both physical and administrative controls. Examples of the physical controls include rope boundaries and postings indicating that access is restricted to only those persons authorized to enter by the LT/FSS Group, and that the movement or storage of radioactive materials is prohibited. Administrative controls include approved procedures and personnel training on the limitations and requirements for access to areas under these controls. Site notices are provided to make all personnel aware of the isolation and control measures in place. An administrative process will be used to evaluate, approve (or deny), and document activities conducted in these areas during and following FSS. In the event that additional remediation is required in an area following the implementation of isolation and control measures, local contamination control measures will be employed as appropriate.

Open land areas, access roads and boundaries will be posted (as well as informational notices) with signs instructing individuals to contact LT/FSS Group personnel prior to conducting work activities in the area. For open land areas that do not have positive access control (i.e., areas that have passed FSS but are not surrounded by a fence), the area will be inspected periodically and any material or equipment that has been introduced into the area since the last inspection will be investigated (i.e., scanned and/or sampled).

5.2.3.2 Area Surveillance Following Final Status Survey

Isolation and control measures will be implemented through approved procedures and will remain in force throughout FSS activities and until there is minimal risk of recontamination from decommissioning or the survey unit has been released from the license. In the event that isolation and control measures established for a given survey unit are compromised, evaluations will be performed and documented to confirm that no radioactive material was introduced into the area that would affect the results of the FSS.

To provide additional assurance that survey units that have successfully undergone FSS remain unchanged until final site release, documented routine surveillances of the completed survey units, or other control measures, will be performed on a semi-annual basis. The surveillances will be performed in areas following FSS completion to monitor for indications of recontamination and verification of postings and access control measures. These routine surveillances will consist of:

- a review of access control entries since the performance of FSS or the last surveillance,
- a walkdown of the areas to check for proper barricades and postings, as applicable, and
- a check for materials introduced into the area or any disturbance that could change the as-left radiological or physical conditions as those encountered during FSS, including the potential for contamination from adjacent decommissioning activities.

If evidence is found of materials that have been introduced into the survey unit or any disturbance that could change the as-left radiological or physical conditions as those encountered during FSS, then radiological survey, consisting of judgmental scan surveys and sampling, will be performed and documented. The survey will focus on access and egress points, any areas of disturbance and any other areas of concern. If the results of the survey indicate that any direct measurement (in the instance of buildings or piping) or sample result (for land survey units) is statistically greater than the initial FSS results (that is, the result is > 2 standard deviations from the initial FSS mean), then an investigation survey will be conducted of the area. The investigation survey (in the form of an RA) will include a larger physical area than the initial survey. If the results of the RA are statistically different than the original FSS results (that is, the result is > 2 standard deviations from the initial FSS mean), then a full FSS of the affected survey unit will be performed in accordance with the LTP.

5.2.4 Reference Areas and Materials

The DQO process will be used to prepare an FSS sample plan to determine whether media specific backgrounds, ambient area background or no background will be applied to a survey unit. The approach used for a specific survey unit will be based on the type of survey unit and type of measurements or samples used to demonstrate compliance.

If applied, media specific backgrounds will be determined via measurements made in one or more reference areas and on various materials selected to represent the baseline radiological conditions for the site. This determination of media specific background will be controlled with a documented survey plan, which will include the DQO process. These data will be evaluated in a technical support document and available for inspection by the NRC. This process will ensure that the data collected will meet the needs of the FSS. The collected data is typically used as the reference area data set when using the Wilcoxon Rank Sum test, but at FCS the Sign test will be used, as background is expected to constitute a small fraction of the $DCGL_W$ based on the results of characterization surveys. For survey units with multiple materials, background data from reference areas may be subtracted from survey unit measurements (using paired observations) when the Sign test is applied.

Depending on the values of the DCGLs, an alternative method to using material specific backgrounds may be used during FSS of above-ground buildings and buried piping. This alternative method will involve the determination of the ambient area background in the survey unit and will only be applicable to beta-gamma or solely gamma detecting instruments. This determination will be made prior to performing an FSS at a location within a survey area that is of sufficient distance (or attenuation) from the surface to eliminate beta particles originating from the surfaces from reaching the detector. At such a location, the ambient background radiation will be due only to ambient gamma radiation and will be a background component of surface measurements. The average background determined at this location can be used as a conservative estimate since it is expected to be less than the material specific background for the material in the area. This is because the average background does not fully account for the naturally occurring radioactivity in the materials. Using this lower ambient background will result in conservative calculated residual radioactivity levels. If the average background reading exceeds a predetermined value, the survey would be terminated and an investigation performed to determine and eliminate the reason for the elevated reading. Each of the survey unit readings would subtract this average background value and the Sign test applied.

Whether or not they are radionuclide-specific, background measurements should account for both spatial variability over the area being assessed and the precision of the instrument or method being used to make the measurements. Thus, the same materials or areas may require more than one background assessment to provide the requisite background information for the various survey instruments or methods expected to be used for FSS. The result of these background assessments will provide the basis for determining the mean and its associated standard deviation.

5.2.5 Radionuclides of Concern and Mixture Fractions

Technical support document (TSD) 21-043, "Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan" [12] establishes the basis for an initial suite of potential ROC for decommissioning. Industry guidance was reviewed as well as the analytical results from the sampling of various media from past plant operations. An initial suite of radionuclides was prepared after the elimination of some of the theoretical neutron activation products, noble gases, and radionuclides with a half-life less than two years (with the exception of Ce-144). The initial suite is listed in Table 5-2.

Table 5-2 Initial Suite of Radionuclides

Radionuclide	Half Life (Years)	Radionuclide	Half Life (Years)
Am-241	4.32E+02	Fe-55	2.74E+00
C-14	5.70E+03	H-3	1.23E+01
Ce-144	0.78E+00	Ni-59	1.01E+05
Cm-243	2.85E+01	Ni-63	1.00E+02
Cm-244	2.85E+01	Np-237	2.14E+06
Co-58	0.19E+00	Pu-238	8.77E+01
Co-60	5.27E+00	Pu-239	2.41E+04
Cs-134	2.06E+00	Pu-240	6.60E+03
Cs-137	3.02E+01	Pu-241	1.44E+01
Eu-152	1.35E+01	Sb-125	2.76E+01
Eu-154	8.80E+00	Sr-90	2.88E+01
Eu-155	4.76E+00	Tc-99	2.11E+05

LTP Chapter 2 provides detailed characterization data that describes current contamination levels in the basements, above-ground buildings, and soils. The survey data for basements was based on concrete samples obtained from the walls and floors of the Containment Building, Auxiliary Building, and Turbine Building. The survey data for above-ground buildings that are slated to be removed from site was based on concrete samples obtained from balance of plant (BOP) buildings inside the deconstruction area (DA), including data from the Radioactive Waste Processing Building (RWPB). The concrete samples were obtained at biased locations with elevated contact dose rates, contamination levels, and/or evidense of leaks/spills. Surface soil samples were obtained in all impacted areas of the site. In addition, subsurface soil samples were obtained within the DA (including beneath and adjacent to basements) and analyzed for the presence of plant-derived radionuclides. TSD 21-043 evaluates the results of the concrete sample analysis data and refines the initial suite of potential ROC by evaluating the dose significance of each radionuclide.

Insignificant dose contributers were determined consistent with the guidance contained in Section 3.3 of NUREG-1757. LTP Chapter 6 discusses the process used to derive the dose significant ROC including the elimination of insignificant contributor (IC) radionuclides from the initial suite. After eliminating the IC radionuclides in all soil, concrete, embedded pipe, and buried pipe scenarios the ROC were selected. For basement walls/floors and embedded pipe, the ROCs were found to be Cs-137, Co-60, Eu-152, Sr-90, and C-14. For soil, buried pipe, and basement fill material, the ROC were found to be Cs-137, Co-60, Eu-152, and C-14. The IC radionuclides are eliminated from further detailed evaluation. Table 5-3 provides the ROC for the decommissioning of FCS and the normalized mixture fractions based on the radionuclide distribution from TSD 21-043.

Table 5-3 Dose Significant Radionuclides and Renormalized Mixture Fractions

Radionuclide	Containment Building Mix Fraction	AB/TB/RWPB Mix Fraction
C-14	9.27E-01	8.27E-02
Co-60	1.87E-03	1.85E-02
Cs-137	6.74E-02	8.94E-01
Eu-152	3.31E-03	1.15E-03
Sr-90	2.91E-04	4.04E-03
Sum	1.0	1.0

The Containment Building mixture fraction applies to the Containment Building walls and floors. The AB/TB/RWPB mix fraction applies to all basement walls and floors other than the Containment Building as well as embedded pipe which is found in the Auxiliary Building and Turbine Building basements. Sufficient characterization samples have been taken from concrete in the Containment Building, Auxiliary Building, and Turbine Building, to derive the radionuclide mixture and assess the dose impact of HTD radionuclides in walls/floors. The concrete-based mixtures are also considered to be a reasonable estimate of the radionuclide mixture in embedded pipe. However, additional sampling in embedded pipe will be performed during continuing characterization to validate or revise the embedded pipe radionuclide mixture.

There is no indication from the characterization data or operational history that contamination is present in FCS soils. However, if contamination were present, but not identified, the source during plant operation is considered more likely to have been from the Auxiliary Building as opposed to the Containment Building. Therefore, the AB/TB/RWPB radionuclide mixture is assumed to apply to soil, buried pipe and basement fill. Note that due to the expectation of very low concentrations of soil and piping contamination, any uncertainties in the application of the AB/TB/RWPB mixture to soil and buried piping would be unlikely to cause significant dose variability in relation to the 25 mrem/year dose criterion.

The previously inaccessible areas identified in LTP Chapter 2 will be characterized during the continuing characterization process. In order to verify that the IC dose does not change prior to implementing the FSS, and to verify the HTD to surrogate radionuclide ratios used for the surrogate calculation are still valid, OPPD will obtain and analyze concrete samples and soil samples during continuing characterization (including radiological assessments) as described below.

For continuing characterization, 10 percent of all media samples collected in a survey unit during continuing characterization will be analyzed for initial suite of radionuclides from Table 5-2, with a minimum of one sample analyzed, whichever is greater. All samples will first be analyzed by the on-site gamma spectroscopy system. The samples with the highest concentrations of gamma emitters will be sent to an accredited off-site laboratory for the analyses of the full suite of radionuclides from Table 5-2. In the absence of detectable gamma activity, locations will be selected based on the potential for the presence of activity using HSA information or other process knowledge data for sample selection.

The actual IC dose will be calculated for each individual sample result using the initial suite DCGLs (no IC dose adjustment) from LTP Chapter 6, Table 6-7 for soils and Table 6-18 for basements. If the IC dose calculated is less than the IC dose assigned for DCGL adjustment (i.e., 1.25 mrem/year for soil and basement walls/floors), then no further action will be taken. If the actual IC dose calculated from the sample result is greater than the IC dose assigned for DCGL adjustment, then a minimum of five additional investigation samples will be taken around the original sample location. Each investigation sample will be analyzed by the on-site gamma spectroscopy system and sent for analysis of the full initial suite of radionuclides from Table 5-2. As with the original sample, the actual IC dose will be calculated for each investigation sample. If the maximum IC dose exceeds 5 percent, then the additional radionuclide(s) that were the cause of the IC dose exceeding 5 percent will be added as additional ROC for that survey unit during the design and performance of FSS. The surrogate ratio for the added HTD ROC will be based on the continuing characterization data using the methods from TSD 21-043. The ROC for the given survey unit and the survey data serving as the basis for the HTD ROC addition and surrogate calculation will be documented in the release record for the survey unit.

Analyses for HTD radionuclides during continuing characterization include the full initial suite. If the analysis indicates positive results (greater than MDC) for both an HTD ROC (C-14 for concrete or soil, or Sr-90 for concrete) and the corresponding surrogate radionuclide (Cs-137), then the HTD to surrogate ratio will be calculated. If the calculated HTD to surrogate ratio is less than the applicable HTD to surrogate ratio from Table 5-16, then no further action is required. If the HTD to surrogate ratio exceeds the applicable ratio from Table 5-16, then a minimum of five additional investigation samples will be taken around the original sample location. Each investigation sample will be analyzed by the on-site gamma spectroscopy system and then sent to an off-site laboratory for HTD analysis. As with the original sample, the HTD to surrogate ratio will be calculated for each investigation sample. The actual maximum HTD to surrogate ratio observed in any individual sample will be used to infer HTD radionuclide concentrations in the survey units shown to be impacted by the investigation. The survey unit-specific HTD to surrogate ratio and the survey data serving as the basis for the ratio will be documented in the release record for the survey unit(s).

RAs are a form of continuing characterization and will be performed in currently inaccessible soil areas that are exposed after removal of asphalt or concrete roadways and parking lots, rail lines, or building foundation pads (slab-on-grade). A limited number of soil samples are typically collected as a part of the RA. Ten percent of any soil samples collected during an RA in a survey unit, with a minimum of one sample taken, will be analyzed for the full initial suite of radionuclides from Table 5-2.

Mixture fractions are also confirmed during FSS by checking surrogates as discussed in Section 5.2.6.2.

5.2.6 Release Criteria

Before the FSS process can proceed, the DCGLs (referred to in this Chapter as Base Case DCGLs (BcDCGL)) that are used to demonstrate compliance with the 25 mrem/year unrestricted release criterion must be established. The BcDCGLs are calculated by analysis of various pathways (direct radiation, inhalation, ingestion, etc.), media (concrete, soils, pipe, and groundwater), and scenarios through which exposures could occur. Chapter 6 of this LTP describes in detail the approach, modeling parameters, and assumptions used to develop the BcDCGLs.

Each radionuclide-specific BcDCGL is equivalent to the level of residual radioactivity (above background levels) that could, when considered independently, result in a TEDE of 25 mrem/year to the AMCG from a given media. To ensure that the summation of dose from each source term, or media, is 25 mrem/year or less after all FSS is completed, the BcDCGLs are reduced based on a projected, or *a priori*, fraction of the 25 mrem/year dose limit from each media. See Section 5.2.6.8 for the dose summation methods and equations. The *a priori* fraction is based on the results of site characterization, process knowledge, and the extent of planned remediation. The *a priori* dose fractions are listed in Table 5-4.

Table 5-4 *a priori* Dose Fractions for End State Media

Media	*a priori* Fraction
Basement Floors/Walls	0.15
Basement Embedded Pipe	0.3
Soil	0.28
Buried Pipe	0.14
Above-ground Building	0.09
Basement Fill	0.02
Existing Groundwater	0.02
Sum	1

The reduced DCGLs, or Operational DCGLs (OpDCGL), are calculated using Equation 5-1.

Equation 5-1

$$OpDCGL_{m,i} = BcDCGL_{m,i}F_{a\ priori,m}$$

where:

OpDCGL = Operational DCGL for media *m* and radionuclide *i*
BcDCGL = Base Case DCGL for media *m* and radionuclide *i*
$F_{a\ priori}$ = *a priori* dose fraction for media *m*

The OpDCGL is then used for the FSS design of the survey unit (calculation of surrogate DCGLs, investigation levels, etc.).

At FCS, compliance is demonstrated through the summation of dose from seven distinct media in the end-state (basement floor/walls (includes steel liner in Containment and penetrations), basement embedded pipe, basement fill, above-ground buildings, soils, buried pipe and existing groundwater). When applied to backfilled basement floor/walls, embedded pipe and penetrations, the DCGLs are expressed in units of activity per unit of area (pCi/m^2). When applied to soil and basement fill, the DCGLs are expressed in units of activity per unit of mass (pCi/g). For buried piping and above-ground buildings, DCGLs are calculated and expressed in units of activity per surface area (dpm/100 cm^2).

The dose contribution from each ROC, in each media, is accounted for using the SOF method. The SOF is calculated for each media using the BcDCGL adjusted for the insignificant contributor dose. A BcDCGL that is established for the average residual radioactivity in a survey unit is equivalent to a $DCGL_W$. For soils, the $DCGL_W$ can be multiplied by AFs to obtain a BcDCGL that represents the same dose to an individual for residual radioactivity over a smaller area within a survey unit. The scaled value is defined as the $DCGL_{EMC}$. The $DCGL_{EMC}$ will only be applied to Class 1 open land (soil) survey units.

5.2.6.1 Derived Concentration Guideline Levels

5.2.6.1.1 Base Case Derived Concentration Guideline Levels for Basement Floor/Walls

The Basement floor/walls DCGLs ($DCGL_{wf}$) apply to the steel-reinforced concrete walls and floors of the backfilled Containment Building, Auxiliary Building, Turbine Building, Intake Structure, and Circulating Water Tunnels, the steel liner in the Containment Building, and penetrations below 1,001 feet AMSL. The $BcDCGL_{wf}$ are calculated in LTP Chapter 6, Section 6.16. The $BcDCGL_{wf}$ values, after adjusting for the IC dose, are listed in Table 6-31 of this LTP and are reproduced in Table 5-5.

Table 5-5 Base Case DCGLs for Basement Floor/Walls ($BcDCGL_{wf}$) Adjusted for IC Dose

ROC	Containment Building (pCi/m^2)	Auxiliary Building, Turbine Building, Intake Structure, Circulating Water Tunnels (pCi/m^2)
C-14	8.13E+06	8.20E+06
Co-60	8.21E+05	3.40E+06
Cs-137	2.61E+06	6.90E+06
Eu-152	1.88E+06	9.62E+06
Sr-90	8.23E+05	8.23E+05

The $BcDCGL_{wf}$ values in Table 5-5 correspond to summation of the total dose from the three BFM wall/floor scenarios *(in situ*, drilling spoils, and excavation). Section 6.11 of LTP Chapter 6 also calculates DCGLs for each of the three scenarios separately.

5.2.6.1.2 <u>Operational Derived Concentration Guideline Levels for Basement Floor/Walls</u>

The $OpDCGL_{wf}$ values (calculated using Equation 5-1) are used for FSS of basement floors, walls and penetrations and are shown in Table 5-6.

Table 5-6 Operational DCGLs for Basement Floor/Walls ($OpDCGL_{wf}$) Adjusted for IC Dose

ROC	Containment Building (pCi/m^2)	Auxiliary Building, Turbine Building, Intake Structure, Circulating Water Tunnels (pCi/m^2)
C-14	1.22E+06	1.23E+06
Co-60	1.23E+05	5.10E+05
Cs-137	3.92E+05	1.04E+06
Eu-152	2.83E+05	1.44E+06
Sr-90	1.27E+05	1.23E+05

5.2.6.1.3 <u>Base Case Derived Concentration Guideline Levels for Soil</u>

The results of surface and subsurface soil characterization in the impacted areas at FCS show that there is minimal residual radioactivity in soil. At this time, based on the characterization survey results to date, OPPD does not anticipate the presence of significant concentrations of soil contamination.

Surface soil is defined as soil residing in the first 0.15 m layer of soil. A subsurface soil category, which is defined as a layer of soil beginning at the surface but extending to a depth of 1 meter is also assessed to allow for flexibility in compliance demonstration if contamination deeper than 0.15 m is encountered. Site-specific DCGLs for soil were calculated for both the 0.15 m and 1 m thicknesses. Based on characterization data and historical information, there are no expectations of encountering a source term geometry that is comprised of a clean surface layer of soil over a contaminated subsurface soil layer. LTP Section 6.9 provides the exposure scenarios and modeling parameters that were used to calculate the site-specific DCGLs for soils (referred to as Base Case Soil DCGLs in this Chapter). The surface and subsurface soil BcDCGLs for the unrestricted release of open land survey units are provided in Table 5-7. The IC dose percentage of 10 percent was used to adjust the DCGLs in Table 5-7 to account for the dose from the eliminated IC radionuclides.

Table 5-7 Base Case DCGLs for Surface Soil ($BcDCGL_{SS}$) and Subsurface Soil ($BcDCGL_{SB}$) Adjusted for IC Dose

ROC	Surface Soil (0.15 m thickness) (pCi/g)	Subsurface Soil (1.0 m thickness) (pCi/g)
C-14	5.70E+01	9.68E+00
Co-60	3.77E+00	2.93E+00
Cs-137	1.31E+01	7.27E+00
Eu-152	8.41E+00	7.36E+00

5.2.6.1.4 <u>Operational Derived Concentration Guideline Levels for Soil</u>

The OpDCGLs for FSS of surface and subsurface soils are presented in Table 5-8.

Table 5-8 Operational DCGLs for Surface Soil ($OpDCGL_{SS}$) and Subsurface Soil ($OpDCGL_{SB}$) Adjusted for IC Dose

ROC	Surface Soil (0.15 m) (pCi/g)	Subsurface Soil (1.0 m) (pCi/g)
C-14	1.59E+01	2.71E+00
Co-60	1.06E-00	8.21E-01
Cs-137	3.65E+00	2.04E-00
Eu-152	2.36E+00	2.06E-00

5.2.6.1.5 <u>Base Case Derived Concentration Guideline Levels for Buried Piping</u>

The residual radioactivity in buried piping located below 1,001 feet AMSL that will remain and be subjected to FSS is discussed in CPP FC-21-002. The dose assessment methods and resulting DCGLs for buried piping are described in detail in LTP Chapter 6, Section 6.14. Table 5-9 presents the BcDCGLs for buried piping.

Table 5-9 Base Case DCGLs for Buried Piping ($BcDCGL_{BP}$) Adjusted for IC Dose

ROC	Buried Piping DCGL (dpm/100 cm^2)
C-14	2.43E+06
Co-60	1.98E+04
Cs-137	7.60E+04
Eu-152	4.38E+04

5.2.6.1.6 Operational Derived Concentration Guideline Levels for Buried Piping

The OpDCGLs for the FSS of buried piping are presented in Table 5-10.

Table 5-10 Operational DCGLs for Buried Piping ($OpDCGL_{BP}$) Adjusted for IC Dose

ROC	Buried Piping DCGL (dpm/100 cm^2)
C-14	3.40E+05
Co-60	2.77E+03
Cs-137	1.06E+04
Eu-152	6.13E+03

5.2.6.1.7 Base Case Derived Concentration Guideline Levels for Embedded Piping

The BFM groundwater source term transport and dose assessment pathways applicable to embedded pipe are the same as those assumed for concrete (i.e., the activity in the pipe is assumed to be released and mixed with the water in the interstitial spaces of the fill material with the water then used for drinking and irrigation). Note that the DCGLs calculated for embedded pipe assume the instant release of all activity into the basement fill.

$BcDCGL_{EP}$ were calculated for each of the embedded pipe survey units. The $BcDCGL_{EP}$ values from LTP Chapter 6, Section 6.16 are reproduced in Table 5-11. The assumed IC dose percentage of 10 percent was used to adjust the $BcDCGL_{EP}$ values in Table 5-11 to account for the dose from the eliminated IC radionuclides.

Table 5-11 Base Case DCGLs for Embedded Pipe ($OpDCGL_{EP}$) Adjusted for IC Dose

ROC	Auxiliary Floor 971' (dpm/100 m^2)	Auxiliary Floor 989' (dpm/100 m^2)	Turbine Floor 990' (dpm/100 m^2)
C-14	1.41E+09	1.10E+09	7.57E+08
Co-60	4.83E+09	3.77E+09	2.59E+09
Cs-137	7.53E+09	5.88E+09	4.04E+09
Eu-152	8.54E+10	6.67E+10	4.58E+10
Sr-90	2.04E+08	1.59E+08	1.09E+08

5.2.6.1.8 Operational Derived Concentration Guideline Levels for Embedded Piping

The operational DCGLs for the FSS of embedded pipe are presented in Table 5-12.

Table 5-12 Operational DCGLs for Embedded Pipe ($OpDCGL_{EP}$) Adjusted for IC Dose

ROC	Auxiliary Floor 971' (dpm/100 m^2)	Auxiliary Floor 989' (dpm/100 m^2)	Turbine Floor 990' (dpm/100 m^2)
C-14	4.23E+08	3.31E+08	2.27E+08
Co-60	1.45E+09	1.13E+09	7.77E+08
Cs-137	2.26E+09	1.76E+09	1.21E+09
Eu-152	2.56E+10	2.00E+10	1.37E+10
Sr-90	6.11E+07	4.77E+07	3.28E+07

5.2.6.1.9 Base Case Derived Concentration Guideline Levels for Above-Ground Buildings

The BcDCGLs for above-ground buildings, shown below in Table 5-13, are the screening values from NUREG-1757, Vol. 2, Rev. 2, "Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria," Table H.1, "Acceptable Screening Levels for Unrestricted Release." Table H-1 does not include a screening value for Eu-152. The Eu-152 screening value applied is the "Pcrit 0.90" value from Table 5.19 of NUREG/CR-5512, Volume 3 [13].

Table 5-13 Base Case DCGLs for Above-Ground Buildings ($BcDCGL_{AGB}$) Adjusted for IC Dose

ROC	Above-Ground Building DCGL (dpm/100 cm^2)
C-14	3.52E+06
Co-60	6.75E+03
Cs-137	2.66E+04
Eu-152	1.21E+04

5.2.6.1.10 Operational Derived Concentration Guideline Levels for Above-Ground Buildings

The OpDCGLs for the FSS of above-ground buildings are presented in Table 5-14.

Table 5-14 Operational DCGLs for Above-Ground Buildings ($OpDCGL_{AGB}$) Adjusted for IC Dose

ROC	Above-Ground Building DCGL (dpm/100 cm^2)
C-14	3.16E+05
Co-60	6.07E+02
Cs-137	2.39E+03
Eu-152	1.09E+03

5.2.6.1.11 <u>Dose Conversion Factors for Existing Groundwater</u>

The dose from existing groundwater is calculated using dose conversion factors (mrem/year per pCi/L) as opposed to DCGLs. Only positively detected ROC groundwater monitoring results are included in the dose calculation. If a given ROC is not positively detected in any groundwater sample, then the dose from the ROC is assumed to be zero. The dose conversion factors are calculated in LTP Chapter 6, Section 6.18, and reported in Table 6-35. The dose conversion factors in Table 6-35 are reproduced in Table 5-15.

Table 5-15 Dose Conversion Factors for Existing Groundwater

ROC	Dose Conversion Factor for Existing Groundwater (mrem/year per pCi/L)
C-14	2.68E-03
Co-60	2.52E-02
Cs-137	6.86E-02
Eu-152	3.63E-03
Sr-90	1.10E-01

5.2.6.2 Surrogate Radionuclides

The instrumentation and methods used for FSS will be based on the measurement of beta-gamma emitting radionuclides by either gamma spectroscopy or gross counting. As described in more detail below, direct analysis will be performed to determine C-14 concentrations in soil during FSS. The option is available to use gross beta measurements for survey of piping, but this approach is not currently planned. Assuming gamma measurements are used for the survey, the concentrations of the HTD radionuclides will be based on known ratios of the HTD radionuclides to beta-gamma radionuclides when demonstrating compliance with the release criteria. This is accomplished through the application of a surrogate relationship. One exception is the analysis for C-14 in soil which will be performed by direct measurement as opposed to the application of a surrogate.

As a general rule, surrogate ratio DCGLs are developed and applied to materials with residual radioactivity where fairly constant radionuclide concentration ratios can be demonstrated to exist. They are in most cases derived using pre-remediation site characterization data collected prior to the FSS. A surrogate DCGL allows the DCGLs specific to HTD radionuclides in a mixture to be expressed in terms of a single radionuclide that is more readily measured or easy-to-detect (ETD). The ETD or measured radionuclide is called the surrogate radionuclide. The surrogate approach will be applied to basement walls/floors, embedded pipe, buried pipe, and above ground buildings.

The single HTD radionuclide in soil, i.e., C-14, is assessed by direct measurement by an off-site analytical laboratory as opposed to using a surrogate approach. The application of a surrogate approach to soil resulted in the surrogate radionuclide being too low to manage during FSS, particularly when the contingency for cross contamination from the Containment Building decommissioning and demolition are considered. The approach to addressing C-14 in soil is somewhat more complicated due to the high relative concentration of C-14 in the Containment Building and Auxiliary Building basement concrete. The AB/TB/RWPB mix fraction is considered the most likely source of past soil contamination although there is no indication of current soil contamination. The AB/TB/RWPB mix fraction is therefore applied to soil, buried pipe, and fill for the IC dose calculation, ROC selection, and the calculation of gross gamma/beta DCGLs. Direct measurement of C-14 eliminates uncertainty in the C-14 mixture fraction now and after demolition and addresses the contingency of potential cross-contamination from the Containment Building.

The ROC for the basements and embedded pipe are Co-60, Cs-137, and Eu-152, which are gamma emitters and C-14 and Sr-90, which are HTD radionuclides. The surrogate ratios were calculated in TSD 21-043. To account for uncertainty in the data, the 95th percentile values were selected for the final surrogate ratios. Using the 95th percentile and the MDC values will provide reasonable confidence that the projected concentrations of the HTD radionuclides using the surrogate will not be underestimated during FSS. Each HTD radionuclide was ratioed to both Cs-137 and Co-60 in TSD 21-043 in order to determine the most representative gamma emitter for use as a surrogate. Cs-137 was selected as the most appropriate surrogate for C-14 and Sr-90. These final surrogate ratios are summarized in Table 5-16.

Table 5-16 Surrogate Ratios

Radionuclide	Containment Building Mix Surrogate	AB/TB/RWPB Mix Surrogate
C-14	3.83E+01	1.11E+00
Sr-90	1.25E-02	1.97E-01

The maximum ratios from Table 5-16 will be used to infer HTD ROC concentrations unless specific survey information supports the use of a surrogate ratio that is specific to the area. In addition, the actual measured concentrations of HTD ROC may be used in lieu of inference if the quantity of samples is sufficient to quantify the actual HTD ROC concentrations. In these cases, if the survey unit-specific surrogate ratios or the actual measured concentrations are less than the maximum HTD to surrogate ratio from Table 5-16, then the survey unit-specific radiological data and the derived surrogate ratios will be submitted to the NRC for approval. In these cases, the area-specific ratios as determined by actual survey data will be used in lieu of the maximum ratios. The area-specific ratios used and the survey data serving as the basis for the ratios will be documented in the release record for the survey unit.

Using the appropriate scaling factors, the DCGL of the measured radionuclide is modified to account for the represented radionuclide, according to the following equation from Section 4.3.2 of MARSSIM.

Equation 5-2

$$DCGL_{SUR} = DCGL_{ETD} \times \frac{DCGL_{HTD}}{\left[\left({Conc_{HTD}}/{Conc_{ETD}}\right)(DCGL_{ETD})\right] + DCGL_{HTD}}$$

where:

$DCGL_{SUR}$	=	modified DCGL (or Basement Dose Factor) for surrogate ratio
$DCGL_{ETD}$	=	DCGL for easy-to-detect radionuclide
$DCGL_{HTD}$	=	DCGL for the hard-to-detect radionuclide
$Conc_{HTD}$	=	Concentration of the HTD or represented radionuclide
$Conc_{ETD}$	=	Concentration of the ETD or surrogate radionuclide

5.2.6.3 Adjusted Gross DCGLs

For the FSS of above ground buildings, adjusted gross DCGLs ($DCGL_{AG}$) are calculated. This is done because radionuclide-specific data is not acquired with static measurements. The $DCGL_{AG}$ is calculated using Equation 5-3. The surrogate DCGL calculated by Equation 5-2 is used in Equation 5-3.

Equation 5-3

$$DCGL_{AG} = \frac{1}{\left[\left(\frac{f_1}{DCGL_1}\right) + \left(\frac{f_2}{DCGL_2}\right) + \cdots \left(\frac{f_i}{DCGL_i}\right)\right]}$$

Where:

$DCGL_{AG}$	=	Adjusted Gross DCGL in units of dpm/100 cm^2
$DCGL_i$	=	DCGL for detectable radionuclide in units of dpm/100 cm^2
f_i	=	Mixture fraction of detectable radionuclides

5.2.6.4 Sum of Fractions

The SOF or "unity rule" is applied to the data used for the survey planning, data evaluation and statistical tests for the analyses of media since multiple radionuclide-specific measurements will be performed or the concentrations inferred based on known relationships. The application of the unity rule serves to normalize the data to allow for an accurate comparison of the various data measurements to the release criteria. When the unity rule is applied, the $DCGL_W$ (used for the nonparametric statistical test) becomes one. The use and application of the unity rule will be performed in accordance with Section 4.3.3 of MARSSIM.

5.2.6.5 Soil Area Factors

Section 2.5.1.1 and Section 5.5.2.4 of MARSSIM address the concern of small areas of elevated radioactivity in a survey unit. Rather than using statistical methods, a simple comparison to an investigation level is used to assess the impact of potential elevated areas. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_W$ modified by an AF to account for the small area of the elevated radioactivity. The area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_W$. At FCS, AFs and corresponding $DCGL_{EMC}$ are calculated for soil only and are not applicable to any other media.

The $DCGL_{EMC}$ is also referred to as the required MDC for soil scanning, as shown in Equation 5-3 of MARSSIM. The $DCGL_{EMC}$ is calculated using Equation 5-4.

Equation 5-4

$$DCGL_{EMC} = AF \times DCGL_w$$

AFs are calculated using RESidual RADioactive Materials (RESRAD) for each ROC and for source area sizes ranging from 1 m^2 to 100 m^2. The AFs for soils were calculated in LTP Chapter 6, Section 6.20 and are provided in Table 5-17 and Table 5-18.

Table 5-17 Soil Area Factors for ROC 0.15 m Thickness

Radionuclide	Soil Area Factors					
	1m²	2m²	5m²	10m²	100m²	143m²
C-14	3.42E+05	1.43E+05	4.22E+04	1.62E+04	5.89E+02	3.48E+02
Co-60	1.23E+01	6.98E+00	3.76E+00	2.48E+00	1.29E+00	1.24E+00
Cs-137	1.44E+01	8.25E+00	4.47E+00	2.94E+00	1.56E+00	1.50E+00
Eu-152	1.17E+01	6.64E+00	3.59E+00	2.36E+00	1.25E+00	1.20E+00

Table 5-18 Soil Area Factors for ROC 1.0 m Thickness

Radionuclide	Soil Area Factors					
	1m²	2m²	5m²	10m²	100m²	143m²
C-14	1.69E+05	7.57E+04	2.49E+04	1.03E+04	4.52E+02	2.72E+02
Co-60	1.11E+01	6.55E+00	3.61E+00	2.39E+00	1.39E+00	1.34E+00
Cs-137	1.95E+01	1.16E+01	6.44E+00	4.28E+00	2.45E+00	2.35E+00
Eu-152	9.70E+00	5.74E+00	3.17E+00	2.11E+00	1.23E+00	1.20E+00

5.2.6.6 Dose from Groundwater

Based upon the results of groundwater monitoring performed on the FCS site since October 2016, when the reactor was permanently shut down through the current period of active decommissioning, the dose from existing residual radioactivity in groundwater is expected to be low. However, if groundwater contamination is determined to be present at the time of license termination, the dose will be calculated using the existing groundwater Dose Conversion Factors presented in LTP Chapter 6, Section 6.18. An *a priori* fraction of 0.02 is assigned to existing groundwater (see

Table **5-4**) to reduce the OpDCGLs for all other media and ensure a sufficient margin exists to account for the *existing groundwater* dose. The final dose from existing groundwater may exceed 2 percent of the 25 mrem/year dose criterion so long as the compliance dose from Equation 5-6 is less than 25 mrem/year.

5.2.6.7 Dose from Fill Material

The material to be used as basement fill is the soil that was excavated as part of the rail spur expansion project. Approximately 132,000 cubic yards of spoils produced from the excavation of this area are planned to be used on-site as fill material in basement structures after FSS of the structure surfaces and embedded pipe.

The dose from the fill soil that is above the 1001 feet AMSL demolition elevation (3 feet below grade) is the dose attributable to the Class 3 open land soil represented by the fill and is calculated using surface soil DCGLs. This dose is accounted for as an open land survey unit along with all other open land survey units and is not added to the "fill" dose.

The dose from fill below 1001 feet AMSL is calculated using the *in-situ* fill DCGLs as described in Chapter 6 of the LTP, Section 6.19 and Table 6-36 (adjusted for IC dose). This dose is added to the dose from all other site media in the compliance dose calculation. The *in-situ* fill BcDCGLs from Table 6-36 are reproduced in Table 5-19.

Table 5-19 Base Case DCGLs for *in situ* Fill ($BcDCGL_F$) Adjusted for IC Dose

ROC	*in situ* Fill DCGL (pCi/g)
C-14	1.29E+01
Co-60	1.59E+01
Cs-137	1.36E+01
Eu-152	5.50E+02
Sr-90	7.97E-01

The FSS data for the Class 3 open land area represented by the fill material are used to calculate the dose from the *in-situ* fill. Any ROC that is not positively detected will be assigned a concentration of zero in the dose calculation using Equation 5-11. There are no operational DCGLs assigned to the *in-situ* fill. However, an *a priori* fraction of 0.02 is assigned to *in situ* fill (see

Table 5-4) to reduce the OpDCGLs for all other media and ensure a sufficient margin exists to account for the *in-situ* fill dose. The final dose from fill material may exceed 2 percent of the 25 mrem/year dose criterion so long as the compliance dose from Equation 5-6 is less than 25 mrem/year.

5.2.6.8 Demonstrating Compliance with Dose Criterion

The BcDCGLs for backfilled basements, soil, buried piping, embedded piping, above-ground buildings and basement fill for each ROC are presented in Table 5-5, Table 5-7, Table 5-9, Table 5-11, Table 5-13, and Table 5-19, respectively. These values are equivalent to the level of residual radioactivity in the media (above background) that could, when considered independently for each ROC, result in a TEDE of 25 mrem/year to the AMCG.

For all media other than existing groundwater, the dose from the residual radioactivity in each media can be generally expressed by Equation 5-5.

Equation 5-5

$$Dose_m = \sum_{i=1}^{n} 25 \frac{C_{i,m}}{BcDCGL_{i,m}}$$

where:

$Dose_m$ = dose from media *m* (mrem/year)
25 = 25 mrem/year dose criterion
$C_{i,m}$ = concentration of radionuclide *i* in media *m*
$BcDCGL_{i,m}$ = base case DCGL for radionuclide i in media *m*

The dose for existing groundwater is calculated by multiplying the dose conversion factor (mrem/year per pCi/L) by the groundwater concentrations identified through the groundwater monitoring network established during decommissioning. The maximum detectable groundwater concentration(s) for each ROC will be used. If there are no positive results for a given ROC the dose assigned to the ROC is zero.

The results of the FSS performed for each FSS unit (or results of groundwater monitoring for existing groundwater) will be reviewed to determine the maximum dose from each of the seven media (e.g., basement floor/walls, basement embedded pipe, basement fill, soil, buried pipe, above-ground building, and existing groundwater, if applicable). The mean SOF of FSS systematic results and the SOF of any identified elevated areas are used to calculate the dose. For all media except soils, areas of elevated activity are defined in this context as any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL.

The SOF (when using the Operational DCGL) for a single systematic or judgmental measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign test and the mean SOF (when using the Operational DCGL) for the survey unit does not exceed one. For all media except soils, if the SOF for a sample/measurement(s) exceeds one when using Base Case DCGLs, then remediation is required. For soils, the EMC as described in Section 5.5.3.3 of this chapter will apply.

Compliance is demonstrated independently through the FSS of each basement floor/wall, basement embedded pipe, basement fill, soil, buried pipe, and above-ground buildings (and the groundwater monitoring program for existing groundwater). The maximum mean concentrations from the FSS and the maximum positive groundwater monitoring results will be used to calculate dose for each media which are then summed as generally shown in Equation 5-6. The mean values from FSS will include the results of judgmental samples based on an area-weighted average approach.

Equation 5-6

$$D_c = D_{b,wf} + D_s + D_{b,ep} + D_{b,f} + D_{bp} + D_{agb} + D_{egw}$$

where:

D_c = compliance dose (mem/year)
$D_{b,wf}$ = maximum survey unit dose from basement walls and floors (mrem/year)
D_s = maximum survey unit dose from soil (mrem/year)
$D_{b,ep}$ = maximum survey unit dose from basement embedded pipe (mrem/year)
$D_{b,f}$ = maximum survey unit dose from basement fill (mrem/year)
D_{bp} = maximum survey unit dose from buried pipe (mrem/year)
D_{agb} = maximum survey unit dose from above-ground buildings (mrem/year)
D_{egw} = maximum dose from existing groundwater (mrem/year)

Equation 5-7 to Equation 5-14 provide specific calculations for determining the dose for each of the terms in Equation 5-6.

Equation 5-7

$$D_c = 25\ (\text{Max BcSOF}_{b,wf} + \text{Max BcSOF}_s + \text{Max BcSOF}_{b,ep} + \text{BcSOF}_{b,f} + \text{Max BcSOF}_{bp} + \text{BcSOF}_{agb} + \text{Max SOF}_{egw})$$

where:

D_c = compliance dose (mrem/year)
25 = 25 mrem/year dose criterion
Max $BcSOF_{b,wf}$ = Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) for basement wall/floor survey units ($D_{b,wf}$ from Equation 5-6),
Max $BcSOF_s$ = Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) for open land survey units (D_s from Equation 5-6),
Max $BcSOF_{b,ep}$ = Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) for basement embedded pipe survey units ($D_{b,ep}$ from Equation 5-6),
$BcSOF_{b,f}$ = BcSOF (mean of FSS systematic results) for basement fill material open land area survey unit ($D_{b,f}$ from Equation 5-6),
Max $BcSOF_{bp}$ = Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) from buried piping survey units (D_{bp} from Equation 5-6),

Max $BcSOF_{agb}$	= Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) from above-ground building survey units (D_{agb} from Equation 5-6),
Max SOF_{egw}	= Maximum SOF from existing groundwater (D_{egw} from Equation 5-6)

The equations defining the dose calculations for each of the terms in Equation 5-7 are provided below.

Equation 5-8

$$BcSOF_{b,wf} = \sum_{i=1}^{n} \frac{Mean\ Conc_{b,wf,i}}{BcDCGL_{b,wf,i}} + \frac{\left(Elev\ Conc_{b,wf,i} - Mean\ Conc_{b,wf,i}\right)}{\left[BcDCGL_{b,wf,i} \times \left(\frac{SA_{SU}}{SA_{Elev}}\right)\right]}$$

where:

$BcSOF_{b,wf}$	= SOF for basement wall/floor survey unit using Base Case DCGLs
$Mean\ Conc_{b,wf,i}$	= Mean concentration for the systematic measurements taken during the FSS of basement wall/floor survey unit for ROC *i*
$Base\ Case\ DCGL_{b,wf,i}$	= Base Case DCGL for basement wall/floor ($DCGL_B$) for ROC *i*
$Elev\ Conc_{b,wf,i}$	= Concentration of ROC *i* in elevated area
SA_{Elev}	= surface area of the elevated area
SA_{SU}	= surface area of FSS unit

Equation 5-9

$$BcSOF_s = \sum_{i=1}^{n} \frac{Mean\ Conc_{s,i}}{Base\ Case\ Soil\ DCGL_{s,i}} + \frac{\left(Elev\ Conc_{s,i} - Mean\ Conc_{s,i}\right)}{\left[Base\ Case\ DCGL_{s,i} \times (AF_i)\right]}$$

where:

$BcSOF_s$	= SOF for open land survey unit using BcDCGL for ROC *i*
$Mean\ Conc_{s,i}$	= Mean concentration for the systematic measurements taken during the FSS of soils in survey unit for ROC *i*
$Base\ Case\ DCGL_{s,i}$	= Base Case DCGL for soils (surface soils [$DCGL_{SS}$] or subsurface soils [$DCGL_{SB}$] as applicable) for ROC *i*
$Elev\ Conc_{s,i}$	= Concentration for ROC *i* in elevated area
AF_i	= Area Factor for ROC *i*

Equation 5-10

$$BcSOF_{b,ep} = \sum_{i=1}^{n} \frac{Mean\ Conc_{b,ep,i}}{BcDCGL_{b,ep,i}} + \frac{\left(Elev\ Conc_{b,ep,i} - Mean\ Conc_{b,ep,i}\right)}{\left[BcDCGL_{b,ep,i} \times \left(\frac{SA_{SU}}{SA_{Elev}}\right)\right]}$$

where:

$BcSOF_{b,ep}$	= SOF for basement embedded pipe survey unit using Base Case DCGLs
$Mean\ Conc_{b,ep,i}$	= Mean concentration for the systematic measurements taken during the FSS of basement embedded pipe survey unit for ROC *i*
$Base\ Case\ DCGL_{b,ep,i}$	= Base Case DCGL for basement embedded pipe (Bc*DCGLb,ep*) for ROC *i*
$Elev\ Conc_{b,ep,i}$	= Concentration of ROC *i* in elevated area
SA_{Elev}	= surface area of the elevated area
SA_{SU}	= surface area of FSS unit

Equation 5-11

$$BcSOF_{b,f} = \sum_{i=1}^{n} \frac{Mean\ Conc_{b,f,i}}{BcDCGL_{b,f,i}}$$

where:

$BcSOF_{b,f}$	= SOF for basement fill using Base Case DCGLs
$Mean\ Conc_{b,f,i}$	= Mean concentration for the systematic measurements taken during the FSS of the open land survey unit from which the fill was excavated for each ROC *i*. Any ROC that is not positively detected will be assigned a concentration of zero.
$Base\ Case\ DCGL_{b,f,i}$	= Base Case DCGL for basement fill (Bc$DCGL_{b,f}$) for ROC *i*

Equation 5-12

$$BcSOF_{bp} = \sum_{i=1}^{n} \frac{Mean\ Conc_{bp,,i}}{BcDCGL_{bp,i}} + \frac{\left(Elev\ Conc_{bp,i} - Mean\ Conc_{bp,i}\right)}{\left[BcDCGL_{bp,i} \times \left(\frac{SA_{SU}}{SA_{Elev}}\right)\right]}$$

where:

$BcSOF_{bp}$	= SOF for buried pipe survey unit using Base Case DCGLs
$Mean\ Conc_{bp,i}$	= Mean concentration for the systematic measurements taken during the FSS of buried pipe survey unit for ROC *i*
$Base\ Case\ DCGL_{bp,i}$	= Base Case DCGL for buried pipe (Bc$DCGL_{bp}$) for ROC *i*
$Elev\ Conc_{bp,i}$	= Concentration of ROC *i* in elevated area
SA_{Elev}	= surface area of the elevated area
SA_{SU}	= surface area of FSS unit

Equation 5-13

$$BcSOF_{agb} = \sum_{i=1}^{n} \frac{Mean\ Conc_{agb,i}}{BcDCGL_{agb,i}} + \frac{\left(Elev\ Conc_{agb,i} - Mean\ Conc_{agb,i}\right)}{\left[BcDCGL_{agb,i} \times \left(\frac{SA_{SU}}{SA_{Elev}}\right)\right]}$$

where:

$BcSOF_{agb}$	= SOF for above ground building survey unit using Base Case DCGLs
$Mean\ Conc_{agb,i}$	= Mean concentration for the systematic measurements taken during the FSS of above ground building survey unit for ROC *i*
$Base\ Case\ DCGL_{agb,i}$	= Base Case DCGL for above ground building (Bc$DCGL_{agb}$) for ROC *i*
$Elev\ Conc_{agb,i}$	= Concentration of ROC *i* in elevated area
SA_{Elev}	= surface area of the elevated area
SA_{SU}	= surface area of FSS unit

Equation 5-14

$$BcSOF_{egw} = \sum_{i=1}^{n} \frac{Max\ Conc_{egw,i}}{BcDCGL_{egw,i}}$$

where:

$BcSOF_{egw}$	= SOF for existing groundwater using Base Case DCGLs
$Max\ Conc_{egw,i}$	= Maximum positive groundwater concentration for radionuclide *i* measured during groundwater monitoring
$DCF_{egw,,i}$	= Existing groundwater dose conversion factor for radionuclide *i*

Volumetric samples of soil and concrete will be collected during FSS to confirm the HTD to surrogate radionuclide ratios used for the surrogate calculation. C-14 and Sr-90 will be analyzed in the FSS confirmatory concrete samples, and only C-14 will be analyzed for soil samples and sediment samples obtained within buried piping. Concrete samples will be collected from the Containment Building, Auxiliary Building, Turbine Building, and Intake Structure basements (including Circulating Water Tunnels) where concrete will remain. The number of samples collected and analyzed for ROC HTD will be 10 percent of the number of *In Situ* Object Counting System (ISOCS) measurements. The concrete sample locations will be selected from the floor and lower walls in the survey unit to alleviate safety concerns from working at heights and to focus on the areas expected to contain the majority of residual radioactivity. Concrete samples taken during FSS will be analyzed for ROC HTD radionuclides. For concrete samples with positive results for both a HTD ROC (C-14 and Sr-90) and the corresponding surrogate radionuclide (Cs-137), the HTD to surrogate ratio will be calculated. For soil, 100 percent of the FSS samples collected from open land survey units will also be analyzed for ROC HTD radionuclides. If material remains after cleanout of buried pipe and embedded pipe, at least one volumetric sample will be collected during FSS to confirm the HTD ratio. For soil and (sediment within) buried piping with positive results for both a HTD ROC (C-14) and the corresponding surrogate radionuclide (Cs-137), the HTD to surrogate ratio will be calculated. If the calculated surrogate ratio exceeds the applicable ratio from Table 5-16, then the calculated surrogate will be applied to the given survey unit. The survey unit specific ratios used and the survey data serving as the basis for the ratios will be documented in the release record for the survey unit.

5.3 Final Status Survey Design

The general approach prescribed by MARSSIM for FSS requires that at least a minimum number of measurements or samples be taken within a survey unit, so that the nonparametric statistical tests used for data assessment can be applied with adequate confidence. Decisions regarding whether a given survey unit meets the release criterion are made based on the results of these tests. Scanning measurements are used to confirm the design basis for the survey by evaluating if any small areas of elevated radioactivity exist that would require reclassification, tighter grid spacing for the systematic measurements, or both.

The level of survey effort required for a given survey unit is determined by the potential for contamination as indicated by its classification. Class 3 survey units receive judgmental (biased) scanning and randomly located measurements or samples. Class 2 survey units receive scanning over a portion of the survey unit based on the potential for contamination, combined with total surface contamination measurements or sampling performed on a systematic grid. Class 1 survey units receive scanning over 100 percent of the survey unit combined with total surface contamination measurements or sampling performed on a systematic grid. Depending on the sensitivity of the scanning method, the grid spacing may need to be adjusted to ensure that small areas of elevated radioactivity are detected.

5.3.1 Sample Size Determination

Section 5.5 of MARSSIM and Appendix A of NUREG-1757 describe the process for determining the number of sampling and measurement locations (sample size) necessary to ensure that the data set is sufficient for statistical analysis such that there is reasonable assurance that the survey unit will pass the requirements for release. The number of sampling and measurement locations is dependent upon the anticipated statistical variation of the final data set such as the standard deviation, the decision errors, and is a function of the gray region as well as the statistical tests to be applied.

5.3.1.1 Decision Errors

The probability of making decision errors is established as part of the DQO process in establishing performance goals for the data collection design and can be controlled by adopting a statistical approach through hypothesis testing. In this approach, the survey results will be used to select between the null hypothesis or the alternate condition (the alternative hypothesis) as defined and shown below.

- Null Hypothesis (H_0) – The survey unit does not meet the release criterion; and,
- Alternate Hypothesis (H_a) – The survey unit does meet the release criterion.

A Type I decision error would result in the release of a survey unit containing residual radioactivity above the release criterion, or false negative. This occurs when the null hypothesis is rejected when in fact it is true. The probability of making this error is designated as "α".

A Type II decision error would result in the failure to release a survey unit when the residual radioactivity is below the release criterion, or false positive. This occurs when the null hypothesis is accepted when it is in fact not true. The probability of making this error is designated as "β".

Appendix E of NUREG-1757 recommends using an α of 0.05 and states that any value for the β is acceptable. Following the guidance in NUREG-1757, the decision errors that will be used for the FSS at FCS are:

- The α value will always be set at 0.05 (5 percent) unless prior NRC approval is granted for using a less restrictive value.
- The β value will also be initially set at 0.05 (5 percent), but may be modified, as necessary, after weighing the resulting change in the number of required sampling and measurement locations against the risk of unnecessarily investigating and/or remediating survey units that are truly below the release criterion.

5.3.1.2 Unity Rule

The unity rule or SOF, will be used for the survey planning and data evaluations since multiple radionuclide-specific measurements will be performed. As a result, the evaluation criteria and data must be normalized using SOF in order to accurately compare and relate the various data measurements to the release criteria.

5.3.1.3 Gray Region

The gray region is defined in MARSSIM as the range of values for the specified parameter of interest for the survey unit in which the consequences of making a decision error is relatively minor. This can be explained as the range of values for which there is a potential of making a decision error; however, there is reasonable assurance that the parameters will meet the specified criterion for the rejection of the null hypothesis.

The gray region is established by setting an upper and lower boundary. Values for the specified parameter above and below these boundaries usually result in a "black and white" or "go no go" decision. Values between the upper and lower boundary are within the "gray region" where decision errors apply most.

5.3.1.4 Upper Bound of the Gray Region (UBGR)

For the purposes of the FSS, release parameters at or near the release guidelines will typically result in a decision that the survey unit will not meet the requirements for release, with the exception of evaluating elevated areas. As a result, the UBGR is typically set as the Operational DCGL.

5.3.1.5 Lower Bound of the Gray Region (LBGR)

The LBGR is the point at which the Type II error (β), or false positive, applies. In accordance with NUREG-1757, Appendix A, the LBGR should be set at the mean concentration of residual radioactivity that is estimated to be present in the survey unit. However, if no other information is available regarding the survey unit, the LBGR may be initially set equal to 0.5 times the applicable Operational DCGL and may be set as low as the MDC for the specific analytical technique. This will help in maximizing the relative shift and effectively reduce the number of required sampling and measurement locations based upon acceptable risks and decision errors.

5.3.1.6 Relative Shift

The relative shift (Δ/σ) for the survey unit data set is defined as the shift (Δ), which is the UBGR minus the LBGR, divided by sigma (σ), which is the standard deviation of the data set used for survey design. For survey design purposes, sigma values in a survey unit will initially be calculated from characterization survey, RA, RASS, and/or investigation data to assess the readiness of a survey area for FSS. For a survey unit where no significant concentrations of residual radioactivity are identified or anticipated, then survey design for FSS will use a coefficient of variation (a unitless measure that allows the comparison of dispersion across several data sets) of 30% as a reasonable value for σ in accordance with the guidance in MARSSIM, Section 5.5.2.2. Standard deviation values, as determined from the characterization data are generally not recommended for Class 1 survey units as this will typically contain values in excess of the guidelines and have excessive variability which will not be representative of the conditions at the time of the FSS. The standard deviation at the time of the FSS will be approximated as best as possible to ensure the FSS requirements are not too restrictive. This can be accomplished by taking additional measurements in a survey unit prior to performing FSS to establish the standard deviation at the time of FSS. The optimal value for the relative shift should range between (and including) 1 and 3.

5.3.2 Statistical Test

At FCS, the Sign test will be used for the statistical evaluation of the survey data. The Sign test will be implemented using the unity rule, surrogate methodologies, or combinations thereof as described in MARSSIM and Chapters 11 and 12 of NUREG-1505.

The Sign test is the most appropriate test for FSS at FCS, as background is expected to constitute a small fraction of the $DCGL_W$ based on the results of characterization surveys. Consequently, the Sign test will be applied when demonstrating compliance with the unrestricted release criterion without subtracting background (other than ambient background for buildings and piping as discussed in Section 5.2.4).

The number of sampling and measurement locations (N) that will be collected from the survey unit will be determined by establishing the acceptable decision errors, calculating the relative shift, and using Table 5-5 of MARSSIM. As stated in Section 5.3.1.6, optimal values for the relative shift are between (and including) 1 to 3. Smaller values for relative shift substantially increase the number of required sampling and measurement locations, while larger values do little to reduce the required number.

By reading the relative shift from the left side of the Table 5-5 of MARSSIM and cross referencing to the specified decision errors, the number of sampling and measurement locations can be determined. The specified number within the table includes the recommended 20 percent adjustment or increase to ensure an adequate set of data is collected for statistical purposes. MARSSIM Equation 5-2 may alternatively be used to calculate the number of sampling and measurement locations. The result will be rounded up by 20 percent. The sample size calculations will be performed using a specially designed software package such as COMPASS or, as necessary, using hand calculations and/or spreadsheets.

5.3.3 Small Areas of Elevated Activity

Section 2.5.1.1 of MARSSIM addresses the concern of small areas of elevated radioactivity in the survey unit that exceed the $DCGL_W$ (equivalent to the BcDCGL at FCS). Rather than using statistical methods, a simple comparison to an investigation level is used to assess the impact of potential elevated areas. This is referred to as the EMC. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_W$ modified by an AF to account for the reduced dose from a small area of the elevated radioactivity. Note that at FCS, the EMC will only be applied to Class 1 open land (soil) survey units as Class 2 and Class 3 survey units should not have contamination in excess of the $DCGL_W$. For all media other than soil, any residual radioactivity identified by a FSS measurement at concentrations in excess of the respective BcDCGL will be remediated.

The statistical tests that determine if the residual radioactivity exceeds the $DCGL_W$ are not adequate for providing assurance that small areas of elevated radioactivity are successfully detected, as discussed in Section 5.5.2.4 of MARSSIM. Systematic sampling and measurement locations in conjunction with surface scanning are used to obtain adequate assurance that small, elevated areas comply with the $DCGL_{EMC}$; however, the number of statistical systematic sampling and measurement locations must be compared to the scan sensitivity to determine the adequacy of the sampling density. The calculation of the $DCGL_{EMC}$ is detailed in Section 5.2.6.5.

The comparison begins by determining the area bounded by the statistical systematic sampling and measurement locations. This value is calculated by dividing the area of the survey unit (A_{SU}) by N for the Sign test.

Equation 5-15

$$A = \frac{A_{SU}}{n}$$

where:

A	=	Area bounded by samples
A_{SU}	=	Area of the survey unit
n	=	number of samples (N)

The AF is selected from Table 5-17 or Table 5-18, as appropriate for soils corresponding to the bounded area (*A*) calculated. If the calculated bounded area (*A*) falls between two area categories on Table 5-17 or Table 5-18, then the larger of the two areas will be selected along with the corresponding AF. $DCGL_{EMC}$ is then derived by multiplying the selected AF by the applicable $DCGL_W$.

The required scan MDC, which is equal to the $DCGL_{EMC}$, is then compared to the actual scan MDC. If the actual scan MDC is less than or equal to the required scan MDC, then the spacing of the statistical systematic sampling and measurement locations is adequate to detect small areas of elevated radioactivity. If the actual scan MDC is greater than the required scan MDC, then the spacing between locations needs to be reduced due to the lack of scanning sensitivity.

To reduce the spacing, a new number of sampling and measurement locations must be calculated. First, a new AF that corresponds to the actual scan MDC is calculated as follows:

Equation 5-16

$$Adjusted\ AF = \frac{Actual\ Scan\ MDC}{DCGL_W}$$

Next, the adjusted AF is used to look up a new adjusted area (A') from Table 5-17 or Table 5-18. Finally, using the adjusted area (A'), an adjusted number of statistical systematic sampling and measurement locations (n_{EMC}) is calculated as follows:

Equation 5-17

$$n_{EMC} = \frac{A_{SU}}{A'}$$

Therefore, the number of systematic sampling and measurement locations in the survey unit will be adjusted to equal to the value derived for n_{EMC}. When multiple measured radionuclides are present, this process is repeated for each measured radionuclide or the surrogate radionuclide, if a surrogate radionuclide is used. The greatest number of systematic sampling and measurement locations determined from the radionuclides will be used for the survey design.

5.3.4 Scan Coverage

The purpose of scan measurements is to confirm that the area was properly classified and that any small areas of elevated radioactivity are within acceptable levels (i.e., are less than the applicable $DCGL_{EMC}$). Depending on the sensitivity of the scanning method used, the number of total surface contamination measurement locations may need to be increased so the spacing between measurements is reduced.

The amount of area to be covered by scan measurements is presented in Table 5-20, which is reproduced from the portion of Table 5.9 from MARSSIM. As intended by the guidance, the emphasis will be placed on a higher frequency of scans in areas of higher risk. The scan coverage requirements that will be applied for scans performed in support of the FSS of open land and above-ground structure survey units are:

- For Class 1 survey units, 100 percent of the accessible area will be scanned.
- For Class 2 survey units, between 10 percent and 100 percent of the accessible area will be scanned, depending upon the potential of contamination. The amount of scan coverage for Class 2 survey units will be proportional to the potential for finding areas of elevated radioactivity or areas close to the release criterion. Accordingly, the site will use the results of individual measurements collected during characterization to correlate this radioactivity potential to scan coverage levels.

- For Class 3 survey units, judgmental (biased) surface scans will typically be performed on areas with the greatest potential of contamination. For open land areas, this will include surface drainage areas and collection points. In the absence of these features the locations of these judgmental scans will at the discretion of the survey designer.

Table 5-20 Recommended Survey Coverage for Open Land Areas and Structures

Survey Unit Classification	Surface Scans	Soil Samples/Static Measurements
Class 1	100%	Number of sample/measurement locations for statistical test, additional sample/measurements to investigate areas of elevated activity
Class 2	10% to 100%, Systematic and Judgmental	Number of sample/measurement locations for statistical test
Class 3	Judgmental	Number of sample/measurement locations for statistical test

5.3.5 Sample Size Determination for FSS of Basement Wall/Floor

The ISOCS was selected as the instrument of choice to perform FSS in basement surfaces. In summary, the ISOCS detector will be oriented perpendicular to the surface of interest. In most cases, the exposed face of the detector will be positioned at a distance of 3 meters above the surface. A plumb or stand-off guide attached to the detector will be used to establish a consistent source to detector distance and center the detector over the area of interest. With the 90-degree collimation shield installed, this orientation corresponds to a nominal field of view (FOV) of 28 m^2.

Based on the contamination potential of each basement wall/floor survey unit, along with the corresponding required areal coverage for scan surveys, the number of ISOCS measurements required in each FSS unit can be calculated as the quotient of the ISOCS FOV divided into the surface area required for areal coverage. For survey units where physical constraints prevent a FOV of 28 m^2, the detector to source distance can be reduced, thereby reducing the FOV, which will increase the number of measurements to ensure that the required FSS coverage is achieved. In most cases, the measurement will be acquired using the ISOCS with a geometry that evaluates residual activity over the activity depth.

To ensure that the number of ISOCS measurements required for the necessary areal coverage in a basement wall/floor survey unit is sufficient to satisfy a statistically based sample design, a calculation will be performed to determine sample size using the process described in section 5.3.2. This calculation will be applied to the Class 1, Class 2, and Class 3 basement wall/floor survey units. If the sample size based on the statistical design required more ISOCS measurements than the number of ISOCS measurement required by the areal coverage, then the number of ISOCS measurements will be adjusted to meet the larger sample size. For Class 1 basement wall/floor survey units where 100% areal coverage by ISOCS is required, the number of measurements are expected to exceed that required by the statistical test but the process to determine sample size is followed to confirm that this is the case. Additionally, for Class 1 basement wall/floor survey units, sufficient measurements will be taken to ensure that 100% of the surface area is surveyed (ISOCS FOV will be overlapped to ensure that there are no un-surveyed corners and gaps).

In the Class 2 basement wall/floor survey units (where less than 100% ISOCS coverage is required), measurement spacing will be determined in accordance with Section 5.3.6.2 of this Chapter. The number of measurements will also be increased in survey units that exceed 1,000 m^2 to correspond with the MARSSIM recommended survey size density for a Class 2 structure (measurements/1,000 m^2). If the grid spacing allows, the location of the center of each ISOCS measurement FOV will be determined at a distance equal to the radius of the ISOCS FOV from the boundaries of the FSS unit and the FOV radius of other measurement locations. If possible, the FOV for individual measurements should not overlap. If FOV overlap cannot be avoided, then adjustments shall be made, including taking additional measurements to ensure that the required areal coverage is achieved. If a selected location is found to be either inaccessible or unsuitable, then the location will be adjusted to the closest adjacent suitable location. In these cases, a notation will be made in the field log and the coordinates of the new location documented. In addition to the prescribed areal coverage, additional judgmental measurements will be collected at locations with higher potential for containing elevated concentrations of residual radioactivity based on professional judgment.

In the Class 3 basement survey units, each measurement location will be randomly selected using a random number generator. If a selected location is found to be either inaccessible or unsuitable, then the location will be adjusted to the closest adjacent suitable location. In these cases, a notation will be made in the field log and the coordinates of the new location documented. In addition to the prescribed areal coverage, additional judgmental measurements will be collected at locations with higher potential for containing elevated concentrations of residual radioactivity based on professional judgment.

5.3.6 Reference Grid, Sampling and Measurement Location

The survey sampling and measurement locations are a function of the sample size and the survey unit size. The guidance provided in Section 4.8.5 and Section 5.5.2.5 of MARSSIM has been incorporated in this section. For the FSS open land survey units, reference coordinates will be acquired using a Global Positioning System (GPS) coupled with the North American Datum (NAD) standard topographical grid coordinate system, typically based off the survey unit perimeter. For the FSS of above-ground buildings and below-ground basements, reference coordinates will be determined based on the (x, y) system using a local origin.

5.3.6.1 Reference Grid

A reference grid will be used for reference purposes and to locate the sampling and measurement locations. The reference grid will be physically marked during the survey to aid in the collection of samples and measurements. At a minimum, each survey unit will have a benchmark defined that will serve as an origin for documenting survey efforts and results. This benchmark (origin) will be provided on the map or plot included in the FSS sample plan.

5.3.6.2 Systematic Sampling and Measurement Locations

Systematic sampling and measurement locations for Class 1 and Class 2 survey units will be located in a systematic pattern or grid. The grid spacing (L), will be determined using a triangular or square grid. Where in most cases, a triangular grid will be preferred, a square grid can be used if the physical dimensions of a survey unit are conducive to the square grid approach. The equations used to determine the grid spacing for systematic measurement locations in Class 1 and Class 2 open land survey units are as follows:

Equation 5-18

$$L=\sqrt{\frac{A}{0.866N}} \text{ (for a triangular grid or),}$$

$$L=\sqrt{\frac{A}{N}} \text{ (for a square grid)}$$

where:

L	=	grid spacing (dimension is square root of the area)
A	=	the total area of the survey unit
N	=	the desired number of measurements

Once the grid spacing is established, a random starting point will be established for the survey pattern using a random number generator. Starting from this randomly-selected location, a row of points will then be established parallel to one of the survey unit axes at intervals of L. Additional rows will then be added parallel to the first row. For a triangular grid, additional rows will be added at a spacing of 0.866L from the first row, with points on alternate rows spaced mid-way between the points from the previous row. For a square grid, points and rows will be spaced at intervals of L.

The grid spacing can be rounded down for ease of locating sampling and measurement locations on the reference grid. The number of sample and measurement locations identified will be counted to ensure the appropriate number of locations has been identified. Depending upon the configuration and layout of the survey unit and the starting grid location, the minimum number of sampling and measurement locations could fall outside of the survey unit boundary. In this event, either a new random starting location will be specified or the grid spacing adjusted downward until the appropriate number of locations is reached.

Software tools such as Visual Sample Plan (VSP) that accomplish the necessary grid spacing, including random starting points and triangular or square shape, will be employed during FSS design. When available, this software will be used with suitable mapping programs to determine coordinates for a GPS. The use of these tools will provide a reliable process for determining, locating and mapping measurement locations in open land areas separated by large distances and will be helpful during independent verification.

For Class 3 survey units, each sampling and measurement location will be randomly selected using a random number generator.

The systematic sampling and measurement locations within each survey unit will be clearly identified and documented for the purposes of reproducibility. Actual measurement locations will be marked and identified by tags, labels, flags, stakes, paint marks, GPS location, photographic record, or equivalent.

5.3.7 FSS Survey Packages and Sample Plans

The product of the survey design process is a sample plan. Sample plans are prepared for each survey unit independently and address various elements of the survey, including, but not limited to:

- survey unit preparation requirements including isolation and control measures
- maps showing the survey unit boundaries, measurement/sample locations, and locations of elevated readings
- applicable DCGLs
- instrumentation to be used and methodologies for use
- types and quantities of measurements or samples to be made or collected
- measurement and sample nomenclature
- chain of custody requirements
- investigation criteria
- QA/QC requirements (e.g., replicate measurements or split samples)
- applicable health and safety precautions and mitigation measures

A survey package will be prepared and maintained for each survey unit independently. A survey package is a collection of sample plans, files, or other historical data and will contain all of the quality records and other documents relevant to the FSS of a survey unit, including any characterization sample plans and results, RA sample plans and results, RASS sample plans and results, and FSS sample plans and results including the survey unit release records. Survey packages and the documents within are QA records when complete and as such, the records are maintained and controlled in accordance with the site Record Management Program.

5.4 Final Status Survey Implementation

Trained and qualified personnel will perform survey measurements and collect samples. The types of measurements performed for FSS include surface scans, static measurements, gamma spectroscopy of volumetric materials, and in-situ gamma spectroscopy. The requirements and objectives outlined in this LTP have been incorporated into procedures. These procedures will govern the survey design process, survey performance and data assessment (decision making).

5.4.1 Survey Methods

The survey methods to be employed in the FSS will consist of combinations of scanning, static measurements, soil and other media sampling and in-situ gamma spectroscopy. Additional specialized methods may be used if such become available between the time this plan is approved and the completion of FSS activities. Any new technologies will meet the applicable DQOs of this plan, and the technical approach will be documented for subsequent NRC review.

5.4.1.1 Scanning

Scanning is performed in order to locate small, elevated areas of residual activity above the investigation level. It is the process by which a surveyor passes a portable radiation detector within close proximity of a surface with the intent of identifying residual radioactivity. Scan surveys that identify locations where the magnitude of the detector response exceeds an investigation level indicate that further investigation is warranted to determine the amount of residual radioactivity. The investigation levels will be based on the $MDCR_{SURVEYOR}$, Operational DCGL, a fraction of the Operational DCGL, or the $DCGL_{EMC}$ for Class 1 soils. Investigations may also be performed based on review of walkover surveys utilizing GPS and data post-processed using GIS software. Investigations may include the use of a Z-score map, in which areas above various standard deviations above the mean are displayed as candidates for further investigation.

One of the most important elements of a scan survey is defining the limit of detection in terms of the *a priori* scanning MDC in order to gauge the ability of the field measurement system to confirm that the unit is properly classified, and to identify any areas where residual radioactivity levels are elevated relative to the Operational DCGL. If the scanning indicates that the survey unit or a portion of the survey unit has been improperly classified, then the survey design process must be evaluated to either assess the effect of reclassification on the survey unit as a whole (if the whole unit requires reclassification) or a new design must be established for the new unit(s) (in the case of subdivision). A new survey design will require a re-evaluation of the survey strategy to decide if it can meet the requirements of the revised survey design. If not, the survey strategy must be revised based on the available instrumentation and methods.

Technicians will respond to indications of elevated areas while surveying. Upon detecting an increase in visual or audible response, the technician will reduce the scan speed or pause and attempt to isolate and verify the elevated area exceeds the investigation level. If the elevated activity is verified to exceed the established investigation level, the area will be bounded (e.g., marked and measured to obtain an estimated affected surface area).

If surface conditions prevent scanning at the specified distance, the detection sensitivity for an alternate distance will be determined and the scanning technique adjusted accordingly. Whenever possible, surveyors will monitor the visual and audible responses to identify locations of elevated activity that require further investigation and/or evaluation.

For the FSS of basement walls/floors, the surface area covered by a single ISOCS measurement is large (a nominal range of 10-30 m^2) which eliminates the need for traditional scan surveys.

5.4.1.1.1 Beta-Gamma Scanning

The surfaces of above-ground buildings and miscellaneous materials will be scanned for beta-gamma radiation with appropriate instruments such as those listed in Table 5-21. The measurements will typically be performed at a distance of one-half inch or less from the surface and at a scan speed of one detector width/second for hand-held instruments. Audible and visual signals will be monitored.

5.4.1.1.2 Gamma Scanning

Gamma scans will be performed over open land surfaces to identify locations of residual surface activity. NaI gamma scintillation detectors (typically 2" x 2") will be used for these scans. OPPD CPP FC-19-006, "Ludlum Model 44-10 Detector Sensitivity" [14] presents the response and scan MDC of the Ludlum Model 44-10 NaI detectors to Co-60 and Cs-137 radionuclides when used for scanning surface soils. Where appropriate, gamma emitters such as Cs-137 will be used as surrogates to infer the amount of any HTD radionuclides that may be present in the distribution in accordance with Section 5.2.6.1.11.

When using hand-held detectors, gamma scanning is generally performed by moving the detector in a serpentine pattern, usually within 15 cm (6 in) from the surface, while advancing at a rate of approximately 0.5 m (20 in) per second. Audible and visual signals will be monitored.

5.4.1.2 Fixed Measurements

Fixed measurements are taken by placing a detector at a defined distance above a surface, taking a discrete measurement for a pre-determined time interval, and recording the reading. Fixed measurements are collected at random or systematic locations in above-ground building survey units to show compliance with the release criterion. Fixed measurements are also collected within penetrations, and embedded or buried piping survey units at pre-determined increments. Fixed measurements are also collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response.

5.4.1.3 Volumetric Sampling

Volumetric sampling is the process of collecting a portion of a media as a representation of the locally remaining media. The collected portion of the medium is then analyzed to determine the radionuclide concentration. Examples of materials that will be sampled include soil, sediments, and groundwater for open land areas and concrete for structures. Bulk material samples will be analyzed via gamma spectroscopy, alpha spectroscopy and/or liquid scintillation counting.

Trained and qualified individuals will collect and control samples. All sampling activities will be performed under approved procedures. OPPD will utilize a proceduralized chain-of-custody (COC) process to ensure sample integrity.

Quality assurance requirements for FSS activities that apply to sample collection (e.g., split samples, duplicates, etc.) and on-site and off-site laboratories employed to analyze samples as a part of the FSS process will be controlled by approved procedures, in conformance with FCSD-RA-LT-100, "Quality Assurance Project Plan for the License Termination Plan Development, Site Characterization and Final Status Survey Projects at Fort Calhoun Station" [15] (QAPP) and is further described in Section 5.6.

5.4.1.3.1 Sampling of Surface Soils

In this context, surface soil refers to outdoor areas where the soil is, for purposes of dose modeling, considered to be uniformly contaminated from the surface down to a depth of 15 cm (6 inches). These areas will be surveyed through combinations of sampling, scanning, and *in situ* measurements.

Compliance with the release criterion in open land survey units is demonstrated through volumetric sampling of the soil. Samples of surface soil (including sediment or sludge) will be obtained from designated random or systematic locations and at areas of elevated activity identified by gamma scans. An appropriate volume of soil (typically 0.5-1 liter) will be collected at each sampling location using hand trowels, bucket augers, or other suitable sampling tools. A GPS reading will be obtained at each surface soil location and a pinned flag or similar marker will be placed in the ground to mark the location.

Sample preparation includes removing extraneous material and homogenizing and drying the soil for analysis. Separate containers are used for each sample and each container is tracked through the analysis process using a chain-of-custody process.

All surface soil samples taken during continuing characterization and FSS will be analyzed by gamma spectrometry using the on-site laboratory. For continuing characterization, 10 percent of all media samples collected in a survey unit during continuing characterization will be analyzed for the full suite of ROC from Table 5-2. The sample(s) selected for full suite analysis to meet the 10 percent requirement will be from the highest gamma activity of the sample population; however additional samples (above 10 percent) will be sent if they exhibit sufficient activity such that the HTD ROCs will likely be detectable by the laboratory using the nominal surrogate ratios and MDCs. In the absence of detectable gamma activity, locations will be selected based on the potential for the presence of activity using HSA information or other process knowledge data. For FSS, 100 percent of all media samples collected in a survey unit during FSS will be analyzed for C-14.

5.4.1.3.2 Sampling of Subsurface Soils

Subsurface soil refers to soil that is greater than 15 cm below the ground surface where ground surface in this context includes the exposed surface of an excavated area before backfill. Soil that will remain beneath structures such as basement floors/foundations or pavement at the time of license termination are considered subsurface soil.

In accordance with NUREG-1757, Appendix G, if the HSA indicates that there is no likelihood of substantial subsurface residual radioactivity, subsurface surveys are not necessary. The HSA as well as the results of the extensive characterization of subsurface soils in the impacted area surrounding the FCS facility have shown that there is no indication of residual radioactivity in subsurface soil. Consequently, minimal subsurface sampling will be performed during FSS.

In Class 1 open land survey units, a subsurface soil sample will be taken at 10 percent of the systematic surface soil sample locations in the survey unit with the location(s) selected at random. In addition, if during the performance of FSS, the analysis of a surface soil sample indicates the potential presence of residual radioactivity in excess of the subsurface Operational DCGLs (from Table 5-8), then additional biased subsurface soil sample(s) will be taken to the appropriate depth within the area of concern as part of the investigation.

In Class 2 and Class 3 open land survey units, no subsurface soil sample(s) will be taken as part of the survey design. However, as with the Class 1 open land survey units, if during the performance of FSS, the analysis of a surface soil sample indicates the potential presence of residual radioactivity at a concentration in excess of the subsurface Operational DCGLs, then biased subsurface soil sample(s) will be taken to the appropriate depth within the area of concern as part of the investigation.

GeoProbe®, split spoon sampling or other methods can be used to acquire subsurface soil samples. Subsurface soil samples will be obtained to a depth of at least 1 meter or refusal, whichever is reached first. In cases where refusal is met because of bedrock, the sample will be used “as is”. In cases where a non-bedrock refusal is met prior to the 1-meter depth, the available sample will be used to represent the 1-meter sample. If residual radioactivity is detected at concentrations in excess of the subsurface Operational DCGL in the 1-meter sample, an additional meter of depth will be sampled and analyzed.

Subsurface soil samples will be segmented and homogenized over each one-meter of depth. Extraneous material will be removed from each segment and the sample will be adequately dried. The material will then be placed into a clean sample container and properly labeled. All samples will be tracked from time of collection through the final analysis in accordance with procedure and survey package instructions.

All subsurface soil samples taken during continuing characterization and FSS will be analyzed by gamma spectrometry and all subsurface soil samples will be analyzed for C-14.

5.4.1.3.3 Sampling of Subsurface Soils below Structure Basement Foundations

The foundation walls and basement floors below 1,001 feet AMSL of the Containment Building, Auxiliary Building, Turbine Building, Circulating Water Discharge Tunnels, and Intake Structure will remain at the time of license termination. Based on the results of subsurface soil sampling performed during site characterization, it is not likely that the residual radioactivity concentrations in soil beneath these building foundations exceed the site-specific Base Case DCGLs as presented in Table 5-7. However, prior to license termination, it will be necessary to ascertain the radiological conditions of these sub-slab soils to demonstrate suitability for unrestricted release.

As stated in Chapter 2 of the LTP, Section 2.5, the soils under the basement concrete of the Containment Building, the Turbine Building, and the Spent Fuel Pool (SFP) have been designated as “continuing characterization” areas once commodity removal and building demolition have progressed to a point where access can be achieved. Continuing characterization will consist of soil borings or use of GeoProbe technology at the nearest locations along the foundation walls that can be feasibly accessed. The under-basement soil activity will be determined by interpreting results from borings collected at the nearest locations. Locations selected for sampling will be biased to locations having a high potential for the accumulation and migration of radioactive contamination to subsurface soil. Angled soil borings will also be performed to directly access the sub-foundation soils. The exact number and location of the soil borings will be determined using DQOs during the survey design. All samples taken from sub-foundation soils will be analyzed by gamma spectrometry. Ten percent of any sub-slab soil samples taken for continuing characterization will be analyzed for the initial suite of radionuclides from Table 5-2.

If possible, survey design will also consider the possibility of coring through the basement concrete floor slabs to facilitate the collection of soil samples. However, to date, this has been not possible due to the intrusion of groundwater into the basements through the bore hole. This is especially true for the Containment Building basement, as sampling through the foundation would require compromising the integrity of the internal steel liner. To address the issue of groundwater intrusion and still investigate the potential for migration of contamination from building interiors to the sub-foundation soils, any continuing characterization performed in the Containment Building basement, Turbine Building basement and the SFP will include cores into the concrete floor, but not fully through the foundation or liner. The cores will be biased to areas with higher potential of providing a pathway for migration of contamination to sub-foundation soil including stress cracks, floor and wall interfaces, and penetrations through walls and floors for piping. If the analysis of the deepest 0.5 inch "puck" from the core in the foundation does not contain detectable activity, then it will be assumed that the location was not a source of sub-foundation soil contamination. If activity is positively detected at the deepest point in the core, continuing the core to the soil under the foundation will be considered depending on the levels of activity identified and the potential for groundwater intrusion.

If residual radioactivity is detected in subsurface soils adjacent to or under a basement surface, then the investigation will also include an assessment of the potential contamination of the exterior of the structure. A sample plan for the investigation will be created as specified by procedure and the plan and investigation results will be provided to NRC for evaluation. Based on the results of the investigation, OPPD will assess the dose consequences of the subsurface soil contamination or will remediate, as necessary.

5.4.1.3.4 <u>Volumetric Sampling of Concrete</u>

Two methodologies were used for the sampling of concrete during the initial site characterization, and one or both of them will be utilized for continuing characterization and FSS. Standard coring may be performed using a heavy-duty drill that is bolted to a floor or wall surface. A 3-inch wide by 6-inch deep (usually) core is obtained and sliced into one-half inch pucks. Deeper cores will be obtained as necessary to bound the depth of contamination. As an alternative to core boring, a patented procedure that uses a hollow drill bit may be used to obtain exact volumes of concrete material at certain depths while utilizing a vacuum collection system. Material from each of the incremental depths at a location may be captured in a separate container for each depth increment via use of the vacuum system. Ten percent of concrete samples taken for continuing characterization will be analyzed for the initial suite of radionuclides from Table 5-2. The sample(s) selected for full suite analysis to meet the 10 percent requirement will be from the highest gamma activity of the sample population; however additional samples (above 10 percent) will be sent if they exhibit sufficient activity such that the HTD ROCs will likely be detectable by the laboratory using the nominal surrogate ratios and

MDCs. In the absence of detectable gamma activity, locations will be selected based on the potential for the presence of activity using HSA information or other process knowledge data.

5.4.1.3.5 *Volumetric Sampling of Sediments and Surface Water*

Sediments will be assessed by collecting samples within locations of surface water ingress or by collecting composite samples of bottom sediments. Such samples will be collected using approved procedures based on accepted methods for sampling of this nature.

Sediment samples will be evaluated against the site-specific soil Operational DCGLs for each of the potential ROC as presented in Table 5-8. The assessment of residual radioactivity levels in surface water drainage systems will be made through the sampling of sediments, total surface contamination measurements, or both, at traps and other access points where it is expected that radioactivity levels will be representative or bounding of the residual radioactivity on the interior surfaces.

5.4.1.4 Excavations

Any soil excavation created to expose or remove a potentially contaminated basement structure will be subjected to FSS prior to backfill. The FSS will be designed as an open land survey using the classification of the removed structure in accordance with Section 5.3 of the LTP using the Operational DCGLs for surface soils or subsurface soils (depending on the thickness of contamination) as the release criterion.

During decommissioning of FCS, any surface or subsurface soil contamination that is identified by continuing characterization or operational radiological surveys that is in excess of the Base Case DCGLs for each of the potential ROC as presented in Table 5-7 will be remediated. The remediation process will include performing a RASS of the open excavations in accordance with procedures. The RASS will include scan surveys and the collection of soil samples during excavation to gauge the effectiveness of remediation, and to identify locations requiring additional excavation. The scan surveys and the collection of and subsequent laboratory analysis of soil samples will be performed in a manner that is intended to meet the DQOs of FSS. The data obtained during the RASS is expected to provide a high degree of confidence that the excavation, or relevant portion of the excavation, meets the criterion for the unrestricted release of open land survey units. Soil samples will be collected to depths at which there is high confidence that deeper samples will not result in higher concentrations. Alternatively, an NaI detector or intrinsic germanium detector of sufficient sensitivity to detect residual radioactivity at the Operational DCGL can be used to scan the exposed soils in an open excavation to identify the presence or absence of soil contamination, and the extent of such contamination. If the detector identifies the presence of contamination at a significant fraction of the Operational DCGL, additional confirmatory investigation and analyses of soil samples of the suspect areas will be performed.

5.4.1.5 Reuse of Excavated Soils

OPPD will not stockpile and store excavated soil for reuse as backfill in basements, with the exception of the soil that was excavated as part of the rail spur expansion project. Approximately 132,000 cubic yards of spoils produced from the excavation of this area are planned to be used on-site as fill material in basement structures after FSS of the structure surfaces and embedded pipe. As such, the survey of this material was equivalent to an FSS, and a dose will be attributed to the material and included in the basement fill inventory. Chapter 6 of the LTP, Section 6.20, provides a detailed discussion on establishing the basement fill material DCGLs (adjusted for IC dose).

There will be overburden soils that are created to expose buried components (e.g., concrete pads, buried pipe, buried conduit, etc.) that will be removed and disposed of as waste or, to install a new buried system. In these cases, the overburden soil will be removed, the component will be removed or installed, and the overburden soil will be replaced back into the excavation. In these cases, an RA will be performed. The footprint of the excavation, and areas adjacent to the excavation where the soil will be temporarily staged, will be scanned prior to the excavation at a frequency used for FSS commensurate with the classification of the land area in which the excavation is located. In addition, periodic scans will be performed on the soil as it is excavated and the exposed surfaces of the excavated soil will be scanned after it is piled next to the excavation for reuse. In addition, a final scan will be performed when the excavation is complete.

Soil samples will be obtained throughout the excavation process at the location of any elevated areas identified during the scan and on a periodic basis at a frequency of approximately one sample for every 300 cubic feet of soil removed; however, a minimum of one sample will be obtained upon completion of the excavation activities and prior to backfill. Additionally, a soil sample will be acquired at any scan location that indicates activity in excess of the investigation levels in Table 5-23 using the surface soil Operational DCGLs. Any soil confirmed as containing residual radioactivity at concentrations exceeding the investigation levels in Table 5-23 using the surface soil Operational DCGLs will not be used to backfill the excavation and will be disposed of as waste. All soil samples taken during the RA of an excavation will be analyzed by gamma spectrometry using the on-site laboratory. Additionally, 10 percent of all soil samples collected during the RA of an excavation will be analyzed for the initial suite of radionuclides from Table 5-2.

An RA is performed prior to introducing off-site material to OPPD for use as backfill in a basement, or for any other use. The RA will be performed at the borrow pit, landfill, or other location from where the material originated and will consist of gamma scans and material sampling using the criteria specified in the preceding paragraph. Gamma scans are performed *in situ*, or by package (using a hand-held instrument or through the use of a truck monitor). Soil samples of overburden soils will be analyzed by gamma spectroscopy.

All RAs will be documented in the release record for the relevant survey units in which they are performed.

5.4.1.6 Pavement Covered Areas

Paved surfaces that remain at the site following decommissioning activities will require surveys for residual radioactivity. Paved areas will be incorporated into the larger open land survey units in which they reside. This is appropriate as the pavement is outdoors where the exposure scenario is most similar to direct radiation from surface soil. Pavement will be released as a surface soil and surveyed accordingly in accordance with the classification of the open land survey unit in which it resides. Samples of the pavement will be acquired at each systematic sample location. The sample media will be pulverized, analyzed by gamma spectrometry and compared with the Operational DCGL for surface soil for each of the potential ROC. If pavement exhibits residual radioactivity in excess of the Base Case DCGL for surface soil, then the pavement will be removed and disposed of as radioactive waste and the soil beneath will be investigated.

5.4.1.7 Buried Piping

Designated sections of buried piping will be remediated in place as necessary, and undergo FSS. If specific buried piping has not undergone characterization prior to FSS design, then prior to remediation, a minimum of one sediment sample will be obtained, if available, and analyzed by gamma spectrometry using the on-site laboratory. Additionally, all sediment samples will be analyzed at an off-site laboratory for C-14. The inventory of buried piping located below 1,001 feet AMSL that will remain and be subjected to FSS is provided in CPP FC-21-002. Compliance with the Operational DCGL values for buried piping, as presented in Table 5-10, will be demonstrated by measurements of total surface contamination.

The radiological survey of pipe system interiors involves the insertion of appropriately sized detectors into the pipe interior by a simple “push-pull” methodology, whereby the position of the detector in the piping system can be easily determined in a reproducible manner.

The detectors are configured in a fixed geometry relative to the surveyed surface, thus creating a situation where a defensible efficiency can be calculated. The detectors are then deployed into the actual pipe and timed measurements are acquired at intervals commensurate with the contamination potential of the pipe. A conservative “area of detection” is assumed for each pipe size. It is also conservatively assumed that any activity is uniformly distributed in the area of detection.

A static measurement is acquired at a pre-determined interval for the areal coverage to be achieved. The measurement output represents the gamma activity in gross cpm for each increment of piping traversed. This measurement value in cpm is then converted to dpm using the efficiency of the detector. The total activity in dpm is then adjusted for the assumed total effective surface area commensurate with the pipe diameter, resulting in measurement results in units of dpm/100 cm^2. A surrogate correction based upon the radionuclide distribution present in the pipe is then applied to the gamma emission to account for the presence of other non-gamma emitting radionuclides in the mixture. This measurement result represents a commensurate and conservative gross measurement that can be compared to the buried pipe Operational DCGLs.

Radiological evaluations for piping that cannot be accessed directly will be performed via measurements made at traps and other appropriate access points where the radioactivity levels are deemed to either bound or be representative of the interior surface radioactivity levels providing that the conditions within the balance of the piping can be reasonably inferred based on those data.

5.4.1.8 Embedded Piping and Penetrations

The end state will include embedded piping and penetrations. An embedded pipe is defined as a pipe that runs vertically through a concrete wall or horizontally through a concrete floor and is contained within a given building. A penetration is defined as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end. The list of penetrations and embedded piping to remain is provided in CPP FC-21-002.

Embedded pipe and penetrations have separate Operational DCGLs. The Operational DCGLs for embedded piping are listed in Table 5-12. For penetrations, the DCGLs used are the same as the structure in which the penetration resides. However, the survey methods are the same for embedded piping and penetrations.

The residual radioactivity remaining in each section of embedded piping/penetration applicable to each FSS unit will be assessed and quantified by direct survey. Shallow penetrations or short lengths of embedded pipe that are directly accessible will be surveyed using hand-held portable detectors, such as gas-flow proportional or scintillation detectors. Lengths of embedded pipe or penetrations that cannot be directly accessed by hand-held portable detectors will be surveyed using applicable sized NaI or Cesium iodide (CsI) detectors that will be inserted and transported through the pipe using flexible fiber-composite rods or attached to a flexible video camera/fiber-optic cable. The ISOCS will not be used to perform FSS in any embedded pipe or penetration.

The interior of embedded pipe or penetration sections that cannot be accessed directly will be inspected prior to survey using a miniature video camera designed to assess the physical condition of the pipe/sleeve interior surfaces. The miniature camera with supporting lighting components as well as the subsequent detectors that will be used to survey the pipe/sleeve interior surfaces will be maneuvered through the pipe/sleeve by the manipulation of fiber-composite rods which will be manually pushed or pulled to provide locomotion. The detectors will be deployed into the actual pipe/sleeve and a timed measurement acquired at a specified distance traversed into the pipe. This distance will be determined as a DQO based on the contamination potential in the pipe/sleeve. As an example, based upon a conservative "area of detection" for the detectors used, a measurement interval of one measurement for each foot of pipe will conservatively provide 100% areal coverage of all accessible pipe/sleeve interior surfaces.

The detector output will represent the gamma activity in gross cpm. This gamma measurement value in cpm will then be converted to dpm using an efficiency factor based on the calibration source. The total activity in dpm will adjusted for the assumed total effective surface area commensurate with the pipe/penetration diameter, resulting in measurement results in units of dpm/100 cm^2. This measurement result will then represent a commensurate and conservative gamma surface activity. The gamma surface activity for each FSS measurement is then converted to a gamma measurement result (in units of pCi/m^2) for each gamma ROC based on the mixture applicable to the pipe/sleeve surveyed. HTD ROC are inferred to the applicable gamma radionuclide concentration to derive a concentration for each ROC for each measurement taken. The measurement concentration for each ROC is then divided by the applicable Operational DCGL to produce a dose fraction for each ROC. The individual ROC dose fractions are then summed to produce a SOF for the measurement. There is no EMC applicable to embedded pipe or penetrations. Consequently, a measurement SOF that exceeds one would require investigation. For embedded pipe and penetrations, areas of elevated activity are defined as any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL. The SOF (based on the Operational DCGL) for a systematic measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign test (using the Operational DCGL) and, the mean SOF (using the Operational DCGL) for the survey unit does not exceed one. If the SOF for a sample/measurement(s) exceeds one when using Base Case DCGLs, then remediation is required.

5.4.1.9 Survey Considerations for Above-Ground Buildings and Miscellaneous Structures

The FSS of above-ground buildings will consist of scan surveys and static measurements using a beta/gamma detector coupled to a data-logger instrument. The impacted above ground structures that will remain are:

- Training Building
- FLEX Building
- OCA Access Building
- Switchyard 3451 Old Building
- Switchyard 3451 New Building
- 1251 Control and Switchgear Building

Because there are no dose model implications, if the number and identification of above-ground buildings to remain and be subject to FSS changes, the NRC will be notified but NRC approval of the change will not be requested prior to implementation.

The above ground structure listed above will be subjected to FSS using the screening values for building surface contamination from Table H.1 of NUREG-1757, Appendix H.

The FSS of minor solid structures, such as but not limited to the Switchyard, telephone poles, fencing, culverts, duct banks and electrical conduit will be included in the FSS of the open land survey unit in which they reside. These items will be subjected to biased scanning and sampling using the DQO process.

5.4.1.10 Groundwater

Assessments of any residual radioactivity in groundwater at the site will be via groundwater monitoring wells installed at FCS. Ongoing monitoring of surface water and groundwater at FCS include REMP, Radiological Groundwater Protection Program, and National Pollutant Discharge Elimination System (NPDES) monitoring.

5.4.1.11 Survey Considerations for Suspected Discreet Radioactive Particle Areas

Although extensive controls will be utilized to prevent the spread of contamination during decommissioning, previous decommissioning projects have had issues with discrete radioactive particles (DRP) migrating to soil areas. This section provides guidance should DRPs be discovered during the course of decommissioning. The approach outlined below takes precedent from past decommissioning projects, as there is limited regulatory guidance concerning DRPs.

DRP are defined as specks of radioactive material identified usually as either activated corrosion product such as cobalt-60, or an irradiated fuel fragment exhibiting greater than 10,000 corrected counts per minute (100,000 dpm). In open land (soil) survey units where the presence of DRPs is suspected, special survey techniques and actions must be considered. DQOs will be developed that will address instrument sensitivity requirements, increased sample densities, and DRP-focused survey methodologies.

Scan speeds will be decreased from one-half meter/second to a maximum of one-quarter meter/second. If an alarm is verified by re-scan, a soil sample will be obtained. The void space created by the sample will be surveyed to ensure the source of activity was removed by the process of taking the soil sample. If elevated readings are still encountered, then an investigation of the remaining soils will be performed to isolate and capture the potential DRP. If radioactivity levels remain constant during the investigation, then sample analyses or the use of a portable gamma spectroscopy system may be used to identify the source of radioactivity. If the investigations do not identify plant-related derived radionuclides, the increased radioactivity may be deemed to originate from natural radioisotopes and the investigation may cease.

In a laboratory setting, the characteristics of the DRP should be ascertained and documented (e.g., dimensions, weight, material composition, appearance, dose rates). All DRPs will be analyzed using gamma spectroscopy at the on-site laboratory, and then sent to an accredited off-site laboratory and analyzed for the full suite of radionuclides from Table 5-2, if necessary (e.g., as part of an investigation).

A condition report will be generated for each DRP encountered. The LT/FSS Manager will maintain a list of all DRPs, their characteristics, and the locations where the DRPs were found.

5.4.2 Final Status Survey Instrumentation

Radiation detection and measurement instrumentation for performing FSS is selected to provide both reliable operation and adequate sensitivity to detect the ROC identified at the site at levels sufficiently below the Operational DCGL. Detector selection is based on detection sensitivity, operating characteristics and expected performance in the field.

The DQO process includes the selection of instrumentation appropriate for the type of measurement to be performed (i.e., scan, static measurement) that are:

- calibrated to respond to a radiation field under controlled circumstances,
- evaluated periodically for adequate performance to established quality standards, and
- sensitive enough to detect the ROC with a sufficient degree of confidence.

Instrumentation selection will be made to identify the ROC at levels sufficiently below the Operational DCGL. The instrumentation will, to the extent practicable, use data logging to automatically record measurements to minimize transcription errors. Commercially available portable and laboratory instruments and detectors are typically used to perform the three basic survey measurements: 1) surface scanning, 2) static measurements, and 3) radionuclide specific analysis of media samples such as soil and other bulk materials.

Specific implementing procedures will control the issuance, use, and calibration of instrumentation used for FSS. The specific DQOs for instruments are established early in the planning phase for FSS activities, implemented by standard operating procedures (SOP) and executed in the survey plan. Further discussion of the DQOs for instruments is provided below.

5.4.2.1 Instrument Selection

The selection and proper use of appropriate instruments for both total surface contamination measurements and laboratory analyses is one of the most important factors in assuring that a survey accurately determines the radiological status of a survey unit and meets the survey objectives. The survey plan design must establish acceptable measurement techniques for scanning and direct measurements. The DQO process must include consideration as to the type of radiation, energy spectrum and spatial distribution of radioactivity as well as the characteristics of the medium to be surveyed.

Radiation detection and measurement instrumentation will be selected based on the type and quantity of radiation to be measured. For direct measurements and sample analyses, MDCs less than 10% of the Operational DCGL are preferable while MDCs up to 50% of the Operational DCGL are acceptable. Instruments used for scan measurements in Class 1 areas are required to be capable of detecting radioactive material at the Base Case DCGL. Measurement results with associated MDC that exceed these values may be accepted as valid data after evaluation by FSS supervision. The evaluation will consider the actual MDC, the reported value for the measurement result, the reported uncertainty and the fraction of the Operational DCGL identified in the sample.

Other measurement instruments or techniques may be utilized. The acceptability of additional or alternate instruments or technologies for use in the FSS will be justified in a technical basis evaluation document prior to use. Technical basis evaluations for alternate FSS instruments or techniques will be provided for NRC review 30 days prior to use. This evaluation will include the following:

- description of the conditions under which the method would be used,
- description of the measurement method, instrumentation and criteria,
- justification that the technique would provide the required sensitivity for the given survey unit classification, and
- demonstration that the instrument provides sufficient sensitivity for measurement.

Instrumentation currently proposed for use in the FSS is listed in Table 5-21. Instrument MDCs are discussed in Section 5.4.2.4 and nominal MDC values for the proposed instrumentation are presented in Table 5-22.

Table 5-21 Typical FSS Survey Instrumentation

Measurement Type	Detector Type	Effective Detector Area & Window Density	Instrument Model	Detector Model
Beta Static/Scan Measurement	Scintillation	1.2 mg/cm^2 0.01" Plastic Scintillation 125 cm^2	Ludlum 3001	Ludlum 44-116
Gamma Scan Measurement	Scintillation	2" diameter x 2" length NaI	Ludlum 3001	Ludlum 44-10
Gamma Scan Measurement	Scintillation	3" diameter x 3" length NaI	Ludlum 3001	Ludlum 44-20
Gamma Static/Scan Measurement	High-purity Germanium	N/A	Canberra *In Situ* Object Counting System (ISOCS)	
Gamma Pipe Static Measurement	CsI NaI NaI	0.75" x 0.75" 2" x 2" 3" x 3"	Ludlum 3001	Ludlum 44-159 Ludlum 44-157 Ludlum 44-162
Surface and Volumetric Material (soil, etc.)	High-purity Germanium/CZT	N/A	Canberra Lab *In Situ* Detector or Kromek	N/A

Table 5-22 Typical FSS Instrument Detection Sensitivities

Instruments and Detectors [a]	Radiation	Background Count Time (minutes)	Typical Background (cpm)	Typical Instrument Efficiency [b] (ε_t)	Count Time (minutes)	Static MDC (dpm/100 cm^2)	Scan MDC
Model 44-116	Beta	1.0	200	0.20	1.0	510	1591[c]
Model 44-10	Gamma	1.0	8,000	N/A	1.0	N/A	5.2 pCi/g [d]
Model 44-20	Gamma	1.0	23,000	N/A	1.0	N/A	3.5 pCi/g [d]
ISOCS	Gamma	Up to 60	N/A	60% relative	5-60	10% of the Operational DCGL (pCi/m^2)	N/A
Model 44-159 [e]	Gamma	10.0	450	0.002	1	38,320	N/A
Model 44-157 [e]	Gamma	10.0	6,300	0.025	1	11,750	N/A
Model 44-162 [e]	Gamma	10.0	16,000	0.050	1	8,789	N/A

[a] Detector models listed are used with the Ludlum 3001 Multi-Detector Data Logger

[b]Typical calibration source used is Cs-137. The efficiency is determined by counting the source with the detector in a fixed position from the source (reproducible geometry). The ε_t value is based on ISO-7503-1 and conditions noted for each detector.

[c]Scan MDC, in dpm/100 cm^2, for the 44-116, the detector width is 7.62 cm and results in a count time of 1.00 seconds at 7.62 cm/second scan speed.

[d]Scan MDC in pCi/g is calculated using the approach described in Section 6.7.2.1 of MARSSIM for a Cs-137 nuclide fraction of 0.95 and a Co-60 fraction of 0.05 with a determined detector sensitivity of 940 and 430 cpm per uR/hr for each radionuclide, respectively.

[e]The efficiency varies for the pipe detectors depending on the pipe diameter used. The efficiency used for the table is the averaged efficiency value for the pipe diameters. The detectors and diameters are: model 44-159: 2-4 in. dia., model 44-157: 4-8 in. dia., model 44-162: 8-12 in. dia.

5.4.2.2 Calibration and Maintenance

Instruments and detectors will be calibrated for the radiation types and energies of interest or to a conservative energy source. Instrument calibrations will be documented with calibration certificates and/or forms and maintained with the instrumentation and project records. Calibration labels will also be attached to all portable survey instruments. Prior to using any survey instrument, the current calibration will be verified and all operational checks will be performed.

Instrumentation used for FSS will be calibrated and maintained in accordance with approved OPPD site calibration procedures. Radioactive sources used for calibration will be traceable to the National Institute of Standards and Technology (NIST) and have been obtained in standard geometries to match the type of samples being counted. When a characterized high-purity germanium (HPGe) detector is used, suitable NIST-traceable sources will be used for calibration, and the software set up appropriately for the desired geometry. If vendor services are used, these will be obtained in accordance with purchasing requirements for quality related services, to ensure the same level of quality.

5.4.2.3 Response Checks

Prior to use on-site, all project instrument calibrations will be verified and initial response data collected. These initial measurements will be used to establish performance standards (response ranges) in which the instruments will be tested against on a daily basis when in use. An acceptable response for field instrumentation is an instrument reading within ±20% of the established check source value. Laboratory instrumentation standards will be within ±3 sigma as documented on a control chart.

Instrumentation will be response checked in accordance with OPPD procedures for instrumentation use. Response checks will be performed daily before instrument use and again at the end of use. The check sources used for response checks will emit the same type of radiation as that being measured in the field and will be held in fixed geometry jigs for reproducibility. If the instrument response does not fall within the established range, the instrument will be removed from use until the reason for the deviation can be resolved and acceptable response again demonstrated. If the instrument fails a post-survey source check, all data collected during that time period with the instrument will be carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure. In the event that data is discarded, replacement data will be collected at the original locations.

5.4.2.4 Measurement Sensitivity

The measurement sensitivity or MDC will be determined *a priori* for the instruments and techniques that will be used for FSS. The MDC is defined as the *a priori* activity level that a specific instrument and technique can be expected to detect 95 percent of the time. When stating the detection capability of an instrument, this value should be used. The MDC is the detection limit, (L_D), multiplied by an appropriate conversion factor to give units of activity. The critical level, (L_C), is the lower bound on the 95 percent detection interval defined for L_D and is the level at which there is a 5 percent chance of calling a background value “greater than background.”

This is the value used when actually counting samples or making direct radiation measurements. Any response above this level should be considered as above background (i.e., a net positive result). This will ensure 95 percent detection capability for L_D. The MDC is dependent upon the counting time, geometry, sample size, detector efficiency and background count rate.

5.4.2.4.1 Total Efficiency

Instrument efficiencies (ε_i) for surface measurements are derived from the surface emission rate of the radioactive source(s) used during the instrument calibration. Total efficiency (ε_t) is calculated by multiplying the instrument efficiency (ε_i) by the surface efficiency (ε_s) commensurate with the radionuclide's alpha or beta energy using the guidance provided in ISO 7503-1, "Evaluation of surface contamination - Part 1: Beta-emitters (maximum beta energy greater than 0.15 MeV) and alpha-emitters" [16].

5.4.2.4.2 Static Minimum Detectable Concentration

For static (direct) surface measurements with conventional detectors, the MDC is calculated using the following equation:

Equation 5-19

$$MDC_{static} = \frac{\frac{3.0}{t_s} + 3.29\sqrt{\frac{R_b}{t_s} + \frac{R_b}{t_b}}}{\varepsilon_t \left(\frac{A}{100cm^2}\right)}$$

where:

MDC_{static}	=	Minimum Detectable Concentration in dpm/100cm^2
t_s	=	sample count time
t_b	=	background count time
R_b	=	background count rate (cpm)
ε_t	=	total efficiency
A	=	detector window area (cm^2)

5.4.2.4.3 Beta-Gamma Scan Measurement Minimum Detectable Concentration

Following the guidance of Sections 6.7 and 6.8 of NUREG-1507, MDCs for surface scans of surfaces for beta and gamma emitters will be computed in accordance with the following equation. For determining scan MDCs, a rate of 95% of correct detections is required and a rate of 60% of false positives is determined to be acceptable. Consequently, a sensitivity index value of 1.38 was selected from Table 6.1 of NUREG-1507. The formula used to determine the scanning MDC at the 95% confidence level is:

Equation 5-20

$$MDC_{scan} = \frac{d'\left(\sqrt{b_i} \times \frac{60}{i}\right)}{\varepsilon_t \sqrt{p}\left(\frac{A}{100}\right)}$$

where:

MDC_{scan}	=	Minimum Detectable Concentration in dpm/100cm^2
d'	=	index of sensitivity (1.38)
i	=	observation interval (seconds)
b_i	=	background counts per observation interval
ε_t	=	total efficiency
p	=	surveyor efficiency (0.5-0.75)
A	=	detector window area (cm^2)

The numerator in the beta-gamma scan MDC equation represents the minimum detectable count rate (MDCR) that the observer would "observe" at the performance level represented by the sensitivity index. The surveyor efficiency (*p*) variable is set at 0.5, as recommended by Section 6.7.1 of NUREG-1507. The surveyor efficiency (*p*) variable may be set at 0.75 if the survey is performed using gamma detection instrumentation coupled to a GPS unit and the data is subsequently post-processed using GIS software. The factor of 100 corrects for probe areas that are not 100 cm^2. The observation interval (*i*) is considered to be the amount of time required for the detector field of view to pass over the area of concern. This time depends upon the scan speed, the size of the source, and the fraction of the detector’s sensitive area that passes over the source. The scan speed is based on approximately one detector window width per second. For the Ludlum Model 43-68 gas flow proportional detector, the window width is 8.8 cm resulting in a scan speed of approximately 3.5 inches per second. The floor monitor detector is the Ludlum Model 43-37 with a window width of 13.35 cm which results in a scan speed of 5.25 inches per second. The source efficiency term (ε_s) will be selected to account for effects such as self-absorption, using the values found in Tables 2 and 3 in ISO 7503-1.

5.4.2.4.4 Gamma Scan Minimum Detectable Concentration

In addition to the MDCR and detector characteristics, the scan MDC (in pCi/g) for land areas is based on the areal extent of the hot spot, depth of the hot spot, and the radionuclide (i.e., energy and yield of gamma emissions). If one assumes constant parameters for each of the above variables, with the exception of the specific radionuclide in question, the scan MDC may be reduced to a function of the radionuclide alone.

The evaluation of open land areas requires a detection methodology of sufficient sensitivity for the identification of small areas of potentially elevated activity. Scanning measurements are performed by passing a hand-held detector, typically 2" x 2" NaI gamma scintillation detector, in gross count rate mode across the land surface under investigation. The centerline of the detector is maintained at a source-to-detector distance within 15 cm (6 in) and moved from side to side in a one-meter-wide serpentine pattern at a rate of 0.5 m/sec. This serpentine scan pattern is designed to cross each survey cell (one square meter) five times in approximately ten seconds. The audible and visuals signals are monitored for detectable increases in count rate. An observed count rate increase results in further investigation to verify findings and define the level and extent of residual radioactivity.

An *a priori* determination of scanning sensitivity is performed to ensure that the measurement system is able to detect concentrations of radioactivity at levels below the regulatory release limit. Expressed in terms of scan MDC, this sensitivity is the lowest concentration of radioactivity for a given background that the measurement system is able to detect at a specified performance level and surveyor efficiency.

This method represents the surface scanning process for land areas defined in NUREG-1507 and is the basis for calculation of the scanning detection sensitivity (scan MDC). The gamma scan MDC is discussed in detail in CPP FC-19-006, which examines the gamma sensitivity for 5.08 by 5.08 cm (2" x 2") NaI detectors to several radionuclide mixtures of Co-60 and Cs-137 using sand (SiO_2) as the soil base. CPP FC-19-006 derives the MDC for the radionuclide mixtures at various detector distances and scan speeds. The model in CPP FC-19-006 uses essentially the same geometry configuration as the model used in MARSSIM. CPP FC-19-006 provides MDC values for the expected soil mixture based on detector background condition, scan speed, soil depth (15 cm), soil density (1.6 g/cm^3) and detector distance to the suspect surface.

5.4.2.4.5 HPGe Spectrometer Analysis

The onsite FCS laboratory maintains gamma isotopic spectrometers that are calibrated to various sample geometries, including a 500 mL flat-bottomed beaker geometry for soil analysis. The geometries are created using the Canberra LABSOCS software. These systems are calibrated using a NIST-traceable mixed gamma source. On-site laboratory counting systems are set to meet a maximum MDC of 0.15 pCi/g for Cs-137 in soil which is calculated in accordance with the following equation:

Equation 5-21

$$MDC_{(pCi/g)} = \frac{3 + 4.65\sqrt{B}}{K \times V \times t}$$

where:

B = number of background counts during the count interval *t*
K = proportionality constant that relates the detector response to the activity level in a sample for a given set of measurement conditions
V = mass of sample (g)
t = count time (minutes)

5.4.2.4.6 Pipe Survey Instrumentation

Pipe survey instruments proposed for use with pipe having diameters between 0.75 and 18 inches have been shown to have efficiencies ranging from approximately 0.02 to 0.5. This equates to detection sensitivities of approximately 350 dpm/100 cm^2 to 5,200 dpm/100 cm^2. This level of sensitivity is adequate to detect residual radioactivity below the Operational DCGLs derived for the unrestricted release of buried pipe as presented in Table 5-10 and embedded pipe as presented in Table 5-12.

5.5 Final Status Survey Data Assessment

The DQA approach being implemented at FCS is an evaluation method used during the assessment phase of FSS to ensure the validity of FSS results and demonstrate achievement of the survey plan objectives. The level of effort expended during the DQA process will typically be consistent with the graded approach used during the DQO process. The DQA process will include a review of the DQOs and survey plan design, will include a review of preliminary data, will use appropriate statistical testing, will verify the assumptions of the statistical tests, and will draw conclusions from the data. The DQA includes:

- verification that the measurements were obtained using approved methods,
- verification that the quality requirements were met,
- verification that the appropriate corrections were made to any gross measurements and that the data is expressed in the correct reporting units,
- verification that the MDC requirements have been met,
- verification that the measurements required by the survey design, and any measurements required to support investigation(s) have been included,
- verification that the classification and associated survey unit design remain appropriate based on a preliminary review of the data,
- subjecting the measurement results to the appropriate statistical tests, and
- determining if the residual radioactivity levels in the survey unit meet the applicable release criterion, and if any areas of elevated radioactivity exist.

Once the FSS data are collected, the data for each survey unit will be assessed and evaluated to ensure that it is adequate to support the release of the survey unit. Simple assessment methods such as comparing the survey data mean result to the appropriate Operational DCGL will be performed first. The SOF will be calculated to ensure a value less than unity. The specific nonparametric statistical evaluations will then be applied to the final data set as necessary including the EMC (if applicable) and the verification of the initial data set assumptions. Once the assessment and evaluation are complete, conclusions will be made as to whether the survey unit actually meets the site release criterion or whether additional actions will be required.

In some cases, data evaluation will show that all of the measurements made in a given survey unit were below the applicable Operational DCGL. If so, demonstrating compliance with the release criterion is simple and requires little in the way of analysis. In other cases, residual radioactivity may exist where measurement results both above and below the Operational DCGL are observed. In these cases, statistical tests must be performed to determine whether the survey unit meets the release criterion. The statistical tests must also be used in the survey design to ensure that a sufficient number of measurements are collected.

For FCS, the Sign test is the most appropriate test for FSS. Characterization surveys indicate that Cs-137 found in background due to global fallout constitutes a small fraction of the DCGL. Consequently, the Sign test will be applied to open land, basement surfaces (to include penetrations and steel liner), above ground buildings, embedded pipe and buried piping when demonstrating compliance with the unrestricted release criteria. Section 5.2.4 provides additional discussion on background reference areas, media specific background and area ambient background.

Survey results will be converted to appropriate units of measure (e.g., dpm/100 cm^2, pCi/g, pCi/m^2) and compared to investigation levels to determine if the action levels for investigation have been exceeded. Measurements exceeding investigation action levels will be investigated. If confirmed within a Class 1 soil survey unit, the location of elevated concentration may be evaluated using the EMC, or the location may be remediated and re-surveyed. If direct measurements exceeding investigation action levels are confirmed within a Class 2 (Operational DCGL) or 3 survey unit (0.5 Operational DCGL), in most cases, the entire survey unit will be reclassified and a re-survey performed consistent with the change in classification.

5.5.1 Review of DQOs and Sample Plan Design

Prior to evaluating the data collected from a survey unit against the release criterion, the data are first confirmed to have been acquired in accordance with all applicable procedures and QA/QC requirements.

The DQO outputs will be reviewed to ensure that they are still applicable. The data collection documentation will be reviewed for consistency with the DQOs, such as ensuring the appropriate number of measurements or samples were obtained at the correct locations and that they were analyzed with measurement systems with appropriate sensitivity. A checklist will be incorporated into the approved procedure for FSS data assessment and this checklist will be used in the review. Any discrepancies between the data quality or the data collection process and the applicable requirements will be resolved and documented prior to proceeding with data analysis. Data assessment will be performed by qualified and trained personnel using the approved procedure.

5.5.2 Preliminary Data Review

The first step in the data review process is to convert all of the survey results to the appropriate units. Basic statistical quantities are then calculated for the sample data set (e.g., mean, standard deviation, and median). An initial assessment of the sample and measurement results will be used to quickly determine whether the survey unit passes or fails the release criterion or whether one of the specified nonparametric statistical analyses must be performed.

Individual measurements and sample concentrations will be compared to the Operational DCGL for evidence of small areas of elevated radioactivity or results that are statistical outliers relative to the rest of the measurements. For most FSS, interpreting the results from a survey is most straightforward when all measurements are higher or lower than the Operational DCGL. In such cases, the decision that a survey unit meets or exceeds the release criterion requires little in terms of data analysis. However, formal statistical tests provide a valuable tool when a survey unit’s measurements are neither clearly above nor entirely below the Operational DCGL.

5.5.2.1 Data Validation

The initial step in the preliminary review of the FSS data is a validation of the data to ensure that the data is complete, fully documented and technically acceptable. At a minimum, data validation should include the following actions:

- Ensure that the instrumentation MDC for direct measurements and sample analyses was less than 10 percent of the Operational DCGL, which is preferable. MDCs up to 50 percent of the Operational DCGL are acceptable.
- Ensure that the instrument calibration was current and traceable to NIST standards.
- Ensure that the field instruments used for FSS were source checked with satisfactory results before and after use each day that data were collected.
- Ensure that the MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey.
- Ensure that the survey methods used to collect data were proper for the types of radiation involved and for the media being surveyed.
- Ensure that the sample was controlled from the point of sample collection to the point of obtaining results.
- Ensure that the data set is comprised of qualified measurement results collected in accordance with the survey design which accurately reflect the radiological status of the facility.
- Ensure that the data have been properly recorded.

If the data review criteria are not met, the discrepancies will be evaluated and the decision to accept or reject the data will be documented in accordance with approved procedures. A condition report will be generated to document and resolve discrepancies as applicable.

5.5.2.2 Graphical Data Review

Graphical analyses of survey data that depict the spatial correlation of the measurements are especially useful for such assessments and will be used to the extent practical. At a minimum, a graphical review will consist of a posting plot and a frequency plot or histogram. Additional data review methodologies can be used and are detailed in Section 8.2.2 of MARSSIM.

5.5.2.2.1 Posting Plot

Posting plots can be used to identify spatial patterns in the data. The posting plot consists of the survey unit map with the numerical data shown at the location from which it was obtained. Posting plots can reveal patches of elevated radioactivity or local areas in which the Operational DCGL is exceeded. Posting plots can be generated for background reference areas to point out spatial trends that might adversely affect the use of the data. Incongruities in the background data may be the result of residual, undetected activity, or they may just reflect background variability.

5.5.2.2.2 Frequency Plot

Frequency plots can be used to examine the general shape of the data distribution. Frequency plots are basically bar charts showing data points within a given range of values. Frequency plots reveal such things as skewness and bimodality (having two peaks). Skewness may be the result of a few areas of elevated activity. Multiple peaks in the data may indicate the presence of isolated areas of residual radioactivity or background variability due to soil types or differing materials of construction. Variability may also indicate the need to more carefully match background reference areas to survey units or to subdivide the survey unit by material or soil type.

5.5.3 Applying the Statistical Test

The statistical evaluations that will be performed will test the null hypothesis (H_0) that the residual radioactivity within the survey unit exceeds the Operational DCGL. There must be sufficient survey data at or below the Operational DCGL to statistically reject the null hypothesis and conclude the survey unit meets the site release criteria. These statistical analyses can be performed using a specially designed software package such as COMPASS or, as necessary, using hand calculations and/or electronic spreadsheets and/or databases.

5.5.3.1 Sum of Fractions

The SOF will be applied to FSS data in accordance with the guidance provided in Section 2.7 of NUREG-1757. This will be accomplished by calculating a fraction of the Operational DCGL for each sample or measurement by dividing the reported concentration by the Operational DCGL. If a sample has multiple ROC, then the fraction of the Operational DCGL for each ROC will be summed to provide a SOF for the sample.

If a surrogate Operational DCGL was calculated as part of the survey design for the FSS, then the surrogate Operational DCGL calculated will be used for the selected surrogate radionuclide. Unity rule equivalents will be calculated for each measurement result using the surrogate adjusted Operational DCGL (typically using Cs-137) as shown in the following equation:

Equation 5-22

$$SOF \leq 1 = \frac{Conc_{Cs\text{-}137}}{DCGL_{Cs\text{-}137_s}} + \frac{Conc_{Co\text{-}60}}{DCGL_{Co\text{-}60}} + \ldots \frac{Conc_n}{DCGL_n}$$

where:

$Conc_{Cs\text{-}137}$ = measured mean concentration for Cs-137
$DCGL_{Cs\text{-}137s}$ = Surrogate Operational DCGL for Cs-137
$Conc_{Co\text{-}60}$ = measured mean concentration for Co-60
$DCGL_{Co\text{-}60}$ = Operational DCGL for Co-60
$Conc_n$ = measured mean concentration for radionuclide n
$DCGL_n$ = Operational DCGL for radionuclide n

The unity rule equivalent results will be used to perform the Sign test.

5.5.3.2 Sign Test

The Sign test is a nonparametric statistical evaluation typically used in situations when evaluating sample analyses where the ROC are not present in background, or they are present at acceptably low fractions as compared to the Operational DCGL. The Sign test will be applied using the guidance in Section 8.3 of MARSSIM.

In the event that the Sign test fails, the survey unit will be re-evaluated to determine whether additional remediation will be required or the FSS re-designed to collect more data (i.e., a higher frequency of measurements and samples).

5.5.3.3 Elevated Measurement Comparisons

During FSS, areas of identified elevated activity (hot spots) must be evaluated both individually and in total to ensure compliance with the release criteria. The EMC is only applicable to Class 1 open land (soil) survey units when an elevated area is identified by surface scans and/or biased and systematic samples or measurements. At FCS, the application of the $DCGL_{EMC}$ does not apply to basement surfaces, above-ground building surfaces, embedded pipe, or buried pipe.

The soil investigation level for the EMC is the $DCGL_{EMC}$, which is the Base Case DCGL modified by an AF. Locations identified by surface scans or sample analyses which exceed the Base Case DCGL are subject to additional surveys to determine compliance with the elevated measurement criteria. Based upon the size of the elevated measurement area, the corresponding AF will be determined from Table 5-17 or Table 5-18, as appropriate.

Any identified elevated areas are each compared to the specific $DCGL_{EMC}$ value calculated for the size of the affected area. If the individual elevated areas pass, then they are combined and evaluated under the unity rule. This will be performed by determining the fraction of dose contributed by the average radioactivity across the survey unit and by adding the additional dose contribution from each individual elevated area following the guidance as provided in Section 8.5.1 and Section 8.5.2 of MARSSIM.

The average activity of each identified elevated areas is determined as well as the average activity value for the survey unit. The survey unit average activity value is divided by the Base Case DCGL, the survey unit average value is then subtracted from the average activity value for the elevated area and the result is divided by the appropriate $DCGL_{EMC}$. The net average activity for each identified elevated area is evaluated against its applicable $DCGL_{EMC}$. The fractions are summed and the result must be less than unity for the survey unit to pass. This is summarized in the equation as follows:

Equation 5-23

$$\frac{\delta}{DCGL_W}+\frac{\tau_1-\delta}{DCGL_{EMC_1}}+\frac{\tau_2-\delta}{DCGL_{EMC_2}}+\cdots\frac{\tau_n-\delta}{DCGL_{EMC_n}}<1$$

where:

δ	=	the survey unit average activity
$DCGL_W$	=	the survey unit Base Case DCGL concentration
τ_n	=	the average activity value of hot spot *n*
$DCGL_{EMCn}$	=	the $DCGL_{EMC}$ concentration of hot spot *n*

5.5.4 Data Assessment for FSS of Basement Walls/Floors

After a sufficient number of ISOCS measurements are taken in a basement survey unit, the data will be summarized, including any judgmental or investigation measurements. The measured activity for each gamma-emitting ROC (and any other gamma emitting radionuclide that is positively detected by ISOCS) will be recorded (in units of pCi/m^2). Background will not be subtracted from any measurement. Using the radionuclide mixture fractions applicable to the survey unit, an inferred activity will be derived for HTD ROC using the surrogate approach specified in Section 5.2.6.1.11. The surrogate ratios that will be used are presented in Table 5-15. An SOF calculation will be performed for each measurement by dividing the reported concentration of each ROC by the Operational $DCGL_B$ for each ROC to calculate an individual ROC fraction. The individual ROC fractions will then be summed to provide a total SOF value for the measurement.

The Sign test will be used to evaluate the remaining residual radioactivity against the dose criterion. The SOF for each measurement will be used as the sum value for the Sign test. If the Sign test demonstrates that the mean activity for each ROC is less than the Operational $DCGL_B$ at a Type I decision error of 0.05, then the mean of all the total SOFs for each measurement in a given survey unit is calculated. If the Sign test fails, or if the mean of the total SOFs in a basement exceeds one (using Operational DCGLs), then the survey unit will fail FSS. If a survey unit fails FSS, then an investigation will be implemented in accordance with Section 5.5.5.

For basement walls/floors, areas of elevated activity are defined as any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL. Any area that exceeds the Base Case DCGL will be remediated. The SOF (based on the Operational DCGL) for a systematic or a judgmental measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign test and, the mean SOF (based on the Operational DCGL) for the survey unit does not exceed one. Once the survey data set passes the Sign test (using Operational DCGLs), the mean radionuclide activity (pCi/m^2) for each ROC from systematic measurements along with any identified elevated areas from systematic and judgmental samples will be used with the Base Case DCGLs to perform a SOF calculation for each surface FSS unit in a basement in accordance with the following equation. The dose from residual radioactivity assigned to the FSS unit is calculated using Equation 5-8.

5.5.5 Data Investigations

During the FSS, any areas of concern will be identified and investigated. This will include any areas as identified by the surveyor in real-time during the scanning, any areas identified during post-processing and reviewing of scan survey data, and any results of soil or bulk material analyses that exceed the Operational DCGL. Based on this review, the suspect areas will be addressed by further biased surveys and sampling, as necessary.

5.5.5.1 Investigation Levels

An important aspect of the FSS is the selection and implementation of investigation levels. Investigation levels are levels of radioactivity used to indicate when additional investigations may be necessary. Investigation levels also serve as a quality control check to determine when a measurement process begins to deviate from expected norms. For example, a measurement that exceeds an investigation level may indicate a failing instrument or an improper measurement. However, in general, investigation levels are used to confirm that survey units have been properly classified.

When an investigation level is exceeded, the first step is to confirm that the initial measurement/sample actually exceeds the particular investigation level. This may involve re-taking the initial measurement/sample. Depending on the results of the investigation actions, the survey unit may subsequently require reclassification, remediation, and/or resurvey. Investigation levels are established for each classification of survey unit. The applicable investigation levels (criteria) are provided in Table 5-23.

Table 5-23 Investigation Levels

Classification	Scan Investigation Levels	Direct Investigation Levels
Class 1	>Operational DCGL or >MDC_{scan} if MDC_{scan} is greater than Operational DCGL	>Operational $DCGL_W$
Class 2	>Operational DCGL or >MDC_{scan} if MDC_{scan} is greater than Operational DCGL	>Operational $DCGL_W$
Class 3	>Operational DCGL or >MDC_{scan} if MDC_{scan} is greater than Operational DCGL	>0.5 Operational $DCGL_W$

5.5.5.2 Remediation, Reclassification and Resurvey

In Class 1 open land survey units, any areas of elevated residual radioactivity above the $DCGL_{EMC}$ will be remediated to reduce the residual radioactivity to acceptable levels. In Class 1 survey units for media other than soil (structural surfaces, embedded pipe, and buried pipe), any areas of elevated residual radioactivity above the Base Case DCGL will be remediated. If an area is remediated, then a RASS will be performed to ensure that the remediation was sufficient.

If an individual FSS measurement (ISOCS for basements, sample for soil, or instrument reading for pipe and above-ground structures) in a Class 2 survey unit exceeds the Operational DCGL, then the survey unit, or portion of the survey unit will be investigated. If small areas of elevated activity exceeding the Operational DCGL are confirmed by this investigation or, if the investigation suggests that there is a reasonable potential that contamination is present in excess of the Operational DCGL, then all or part of the survey unit will be reclassified as Class 1 and the survey strategy for that survey unit will be redesigned.

If an individual survey measurement in a Class 3 survey unit exceeds 50 percent of the Operational DCGL, then the survey unit, or a portion of a survey unit, will be investigated. If the investigation confirms residual radioactivity in excess of 50 percent of the Operational DCGL, then the survey unit, or the impacted portion of the survey unit will be reclassified to a Class 1 or a Class 2 survey unit and the survey will be re-designed and re-performed as a Class 1 or Class 2 survey unit.

The DQO process will be used to evaluate the remediation, reclassification and/or resurvey actions to be taken if an investigation level is exceeded. Based upon the failure of the statistical test or the results of an investigation, Table 5-24 presents actions that will be required.

Reclassification of a survey unit from a less restrictive classification to a more restrictive classification may be done without prior NRC approval. However, reclassification to a less restrictive classification requires prior NRC approval.

Table 5-24 Remediation, Reclassification and Resurvey Actions

<table>
<tr><td colspan="4">REMEDIATION</td></tr>
<tr><td colspan="3">Remediation Criteria</td><td>Proposed Remediation</td></tr>
<tr><td rowspan="3">Class 1
Survey Unit</td><td colspan="2">1) Passes Sign test and the mean SOF for survey unit is less than or equal to unity (1) (SOF EMC for open land survey units or Equation 5-5 or 5-6 for structural survey units)</td><td>None</td></tr>
<tr><td colspan="2">2) Passes Sign test and the mean SOF for survey unit is less than or equal to unity (1) with several elevated areas present that require remediation (>$DCGL_{EMC}$ for soils or Base Case DCGL for other media)</td><td>Spot Remediation & Resurvey under Existing Survey Design</td></tr>
<tr><td colspan="2">3) Does not pass Sign test, or the mean SOF is greater than unity</td><td>General Remediation and Restart FSS under new Survey Design</td></tr>
<tr><td rowspan="2">Class 1
Basement
Survey Unit</td><td colspan="2">1) The mean inventory fraction (total mean dose for the survey unit divided by the dose criterion of 25 mrem/year) is greater than or equal to one.</td><td rowspan="2">General Remediation and Restart FSS under new Survey Design</td></tr>
<tr><td colspan="2">2) The sum of the mean inventory fractions for each FSS unit contained within a building basement is greater than or equal to one.</td></tr>
<tr><td colspan="4">RECLASSIFICATION</td></tr>
<tr><td colspan="3">Reclassification Criteria</td><td>Proposed Action</td></tr>
<tr><td rowspan="3">Class 2
Survey Unit</td><td rowspan="3">One or several survey measurements (scan, sample or direct measurement) exceed the Operational DCGL or a portion of the survey unit is remediated.</td><td>The extent of the elevated area relative to the total area of the survey unit is minimal and the source of the residual radioactivity is known</td><td>Reclassify only the bounded discrete area of elevated activity to Class 1.</td></tr>
<tr><td>The extent of the elevated area relative to the total area of the survey unit is minimal and the source of the residual radioactivity is unknown</td><td>Reclassify 2,000 m^2 for soils or 100 m^2 for structures around the area of elevated activity as Class 1.</td></tr>
<tr><td>The extent of the elevated area relative to the total area of the survey unit is significant.</td><td>Reclassify the entire survey unit as Class 1.</td></tr>
<tr><td rowspan="3">Class 3
Survey Unit</td><td rowspan="3">One or several survey measurements (scan, sample or direct measurement) exceed 50% of the Operational DCGL or a portion of the survey unit is remediated.</td><td>The extent of the elevated area relative to the total area of the survey unit is minimal</td><td>Reclassify the area of elevated activity to Class 2.</td></tr>
<tr><td>The extent of the elevated area relative to the total area of the survey unit is significant with measurements or concentrations above the OpDCGL.</td><td>Reclassify the area of elevated activity to Class 1 and create a Class 2 buffer zone of appropriate size around the area.</td></tr>
<tr><td>The extent of the elevated area relative to the total area of the survey unit is significant with no measurement or concentration above the OpDCGL.</td><td>For soils, reclassify 10,000 m^2 around the area of elevated activity to Class 2. For structures, reclassify 1,000 m^2 around the area of elevated activity to Class 2.</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
</table>

Table 5-24 (continued) Remediation, Reclassification and Resurvey Actions

RESURVEY			
Resurvey Criteria			Proposed Action
Class 1 Survey Unit	The survey unit has been remediated.	Survey unit passed Sign test and the mean SOF for survey unit was less than unity with several elevated areas present that required remediation. The power of the original survey is unchanged.	Re-scan remediated area; collect samples/measurements within the remediated area to demonstrate that remediation was successful.
		Survey unit did not pass Sign test, or mean SOF exceeded unity	Resurvey entire survey unit using a new survey design.
	Survey unit has been reclassified from a Class 2 survey unit.	No remediation was performed.	Increase scan or areal coverage to 100%. Additional statistical samples are not required.
Class 2 Survey Unit	Survey unit has been divided to accommodate a new Class 1 survey unit.	The area of the new Class 1 survey unit relative to the area of the initial Class 2 survey unit is minimal and no statistical samples were affected.	No resurvey required in the Class 2 survey unit. Perform survey design and FSS in the Class 1 portion.
		Statistical sample population was affected by the reclassification.	Resurvey entire survey unit using a new survey design.
Class 3 Survey Unit	Survey unit has been divided to accommodate a new Class 2 survey unit.	The area of the new Class 2 survey unit relative to the area of the initial Class 3 survey unit is minimal and the power of the original Class 3 survey is unchanged.	No resurvey required in the Class 3 survey unit. Perform survey design and FSS in the Class 2 portion.
		Statistical sample population was affected by the reclassification.	Resurvey entire survey unit using a new survey design.

5.5.6 Data Conclusions

The results of the statistical testing, including the application of the EMC, allow for one of two conclusions to be made. The first conclusion is that the survey unit meets the site release criterion through the rejection of the null hypothesis. The data provide statistically significant evidence that the level of residual radioactivity within the survey unit does not exceed the release criteria. The decision to release the survey unit will then be made with sufficient confidence and without any further analyses.

The second conclusion that can be made is that the survey unit fails to meet the release criteria. The data may not be conclusive in showing that the residual radioactivity is less than the release criteria. As a result, the data will be analyzed further to determine the reason for failure. Potential reasons may include:

- The average residual radioactivity exceeds the Operational DCGL.
- The average residual radioactivity in soils is less than the Base Case DCGL; however, the survey unit fails the EMC test.
- The survey design or implementation was insufficient to demonstrate compliance for unrestricted release, (i.e., an adequate number of measurements was not performed).
- The test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

"Power" in this context refers to the probability that the null hypothesis is rejected when it is indeed false. The power of the statistical test is a function of the number of measurements made and the standard deviation of the measurement data. Quantitatively, the power is $1 - \beta$ where β is the Type II error rate (the probability of accepting the null hypothesis when it is actually false). A retrospective power analysis can be used in the event that a survey unit is found not to meet the release criterion to determine if this is indeed due to excess residual radioactivity or if it is due to an inadequate sample size. In the case of such a failure, a retrospective power analysis will be performed using the methods as described in Section I.9 and Section I.10 of MARSSIM.

If the retrospective power analysis indicates insufficient power, then an assessment will be performed to determine whether the observed median concentration and/or observed standard deviation are significantly different from the estimated values used during the DQO process. The assessment will identify and propose alternative actions to meet the objectives of the DQOs. These alternative actions can include failing the unit and starting the DQO process over, remediating some or all of the survey unit and starting the DQO process over and adjusting the LBGR to increase sample size. For example, the assessment determines that the median residual concentration in the survey unit exceeds the Operational DCGL or is higher than was estimated and planned for during the DQO process. A likely course of action might be to fail the unit or remediate and resurvey using a new sample design.

There may be cases where the decision was made during the DQO process by the planning team to accept lower power. For instance, during the DQO process the calculated relative shift was found to be less than one. The planning team adjusts the LBGR, evaluates the impact on power and accepts the lower power. In this case, the DQA process would require the planning team to compare the prospective power analysis with the retrospective power analysis and determine whether the lower power is still justified and the DQOs satisfied.

5.6 Quality Assurance

All FSS activities are conducted in accordance with the "Quality Assurance Project Plan for the License Termination Plan Development, Site Characterization and Final Status Survey Projects at Fort Calhoun Station."

As the licensee, OPPD is responsible for all licensing activities, safety, radiation protection, environmental safety and health, engineering and design, quality assurance, construction management, environmental management, waste management and financial management. OPPD interfaces directly with the NRC and other stakeholders on all issues pertaining to decommissioning project activities at FCS.

OPPD has developed and is implementing a comprehensive QA Program to assure conformance with established regulatory requirements. The governing QA program for the FCS site is the OPPD NO-FC-10, "Quality Assurance Topical Report" (QATR) [17]. The quality requirements and quality concepts presented in the QATR adequately encompass all risk-significant decommissioning activities. The participants in the OPPD QA Program assure that the design, procurement, construction, testing, operation, maintenance, repair, modification, dismantlement and remediation of nuclear reactor components are performed in a safe and effective manner.

The OPPD QA Program complies with the requirements set forth in Appendix B of 10 CFR 50, Appendix H of 10 CFR 71, and Appendix G of 10 CFR 72. References to specific industry standards for QA and QC measures governing FSS activities are reflected in the QAPP as well as all applicable supporting procedures, plans, and instructions. Effective implementation of QA and QC measures will be verified through audit activities, with corrective actions being prescribed, implemented and verified in the event any deficiencies are identified. These measures will also apply to the any FSS related services provided by off-site vendors, in addition to on-site sub-contractors.

The QAPP has been prepared to ensure the adequacy of data being developed and used during FSS. It supplements the quality requirements and quality concepts presented in the QATR. Compliance with the QAPP will serve to ensure that FSS are performed by trained individuals using approved written procedures and properly calibrated instruments that are sensitive to the suspected ROC. In addition, QC measures will be taken to obtain quantitative information to demonstrate that measurement results have the required precision and are sufficiently free of errors to accurately represent the area being investigated. QC checks will be performed as prescribed by the QAPP for both field measurements and laboratory analysis. Effective implementation of FSS operations will be verified through periodic audit and surveillance activities, including field walk-downs by management and program self-assessments. Corrective actions will be prescribed, implemented, and verified in the event any deficiencies are identified.

5.6.1 Project Management and Organization

OPPD has established the LT/FSS Group (through partnership with Energy*Solutions*) with sufficient management and technical resources to fulfill project objectives and goals. The LT/FSS Group is responsible for:

- site characterization,
- LTP development and implementation, and
- the performance of FSS.

Characterization and FSS encompasses all survey and sampling activities related to the LTP. This includes site characterization surveys, RASS, RA, and FSS. The duties and responsibilities of key LT/FSS managers as well as the various key positions within the LT/FSS Group are provided in Section 3.3 of the QAPP. Responsibilities for each of the positions described may be assigned to a designee. An organizational chart is provided as Figure 5-1.

5.6.2 Quality Objectives and Measurement Criteria

The QA objectives for FSS is to ensure the survey data collected are of the type and quality needed to demonstrate, with sufficient confidence, that the site is suitable for unrestricted release. The objective is met through use of the DQO process for FSS design, analysis and evaluation. Compliance with the QAPP ensures that the following items are accomplished:

- The elements of the FSS plan are implemented in accordance with the approved procedures.
- Surveys are conducted by trained personnel using calibrated instrumentation.
- The quality of the data collected is adequate.
- All phases of package design and survey are properly reviewed, with QC and management oversight provided.
- Corrective actions, when identified, are implemented in a timely manner and are determined to be effective.

The following describe the basic elements of the QAPP.

5.6.2.1 Written Procedures

Sampling and survey tasks will be performed properly and consistently in order to assure the quality of FSS results. The measurements will be performed in accordance with approved, written procedures. Approved procedures describe the methods and techniques used for FSS measurements.

5.6.2.2 Training and Qualifications

Personnel performing FSS measurements will be trained and qualified. Training will include the following topics:

- procedures governing the conduct of the FSS

- operation of field and laboratory instrumentation used in the FSS
- collection of FSS measurements and samples

Qualification is obtained upon satisfactory demonstration of proficiency in implementation of procedural requirements. The extent of training and qualification will be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity required to be performed by that individual. Records of training and qualification are recorded in the OPPD Personnel Qualification Database and will be maintained in accordance with approved procedures.

5.6.2.3 Measurement and Data Acquisitions

The FSS records will be designated as quality documents and will be governed by site quality programs and procedures. Generation, handling and storage of the original FSS design and data packages will be controlled by site procedures. Each FSS measurement will be identified by individual, date, instrument, location, type of measurement, and mode of operation.

5.6.2.4 Instrument Selection, Calibration and Operation

Proper selection and use of instrumentation will ensure that sensitivities are sufficient to detect radionuclides at the required *a priori* MDC as well as assure the validity of the survey data. Instrument calibration will be performed with NIST traceable sources using approved procedures. Issuance, control and operation of the survey instruments will be conducted in accordance with the approved procedures.

5.6.2.5 Chain of Custody

Responsibility for custody of samples from the point of collection through the determination of the FSS results is established by procedure. When custody is transferred outside of the organization, a CoC form will accompany the sample for tracking purposes. Secure storage will be provided for archived samples.

5.6.2.6 Control of Consumables

In order to ensure the quality of data obtained from FSS surveys and samples, new sample containers will be used for each sample taken. Tools used to collect samples will be cleaned to remove potential contamination prior to taking additional samples. Tools will be decontaminated after each sample collection and surveyed for contamination, as applicable.

5.6.2.7 Control of Vendor-Supplied Services

Vendor-supplied services, such as instrument calibration and laboratory sample analysis, will be procured from appropriate vendors in accordance with approved quality and procurement procedures.

5.6.2.8 Database Control

Software used for data reduction, storage or evaluation will be fully documented. The software will be tested and validated prior to use by an appropriate test data set. Commercially available software (e.g., Excel) is exempt from this requirement.

5.6.2.9 Data Management

Survey data control from the time of collection through evaluation will be specified by procedure and sample plan instructions. Manual data entries will be verified by a second individual.

5.6.3 Measurement/Data Acquisition

QC surveys and samples will be performed primarily as verification that the original FSS results are valid. QC surveys may include replicate surveys, field blanks and spiked samples, split samples, third-party analysis and sample recounts.

Replicate measurements apply to scan and static direct measurements performed in any type of survey unit. Replicate surveys refer to the complete repeat FSS of a specified non-land area survey unit. Performance of replicate measurements is the preferred method for ensuring quality during measurement/data acquisition because they are performed in all survey units over the course of decommissioning. Therefore, replicate measurements are required whereas replicate surveys are only performed as directed by the LT/FSS Manager.

Field blanks and sample recounts apply to loose-surface and material sampling surveys. Spiked samples and split samples apply to volumetric material sampling surveys. Third-party analysis applies to volumetric material samples counted by a different laboratory than normally used. QC survey results will be evaluated and compared to the original FSS results in accordance with the appropriate acceptance criteria.

5.6.3.1 QC Replicate Measurements

A replicate measurement is an independent scan or static direct measurement performed by a qualified technician, other than the one who obtained the original FSS measurement, using a separate but similar instrument. In cases where the instrumentation used is highly specialized or of limited quantity (e.g., ISOCS), the same instrument may be used with the approval of the LT/FSS Manager. The original technicians may perform the replicate measurement with the approval of the responsible LT/FSS Manager in cases where specialized training is required in measurement acquisition or in operation of the instrument.

Replicate measurements will be performed on 5 percent of the static and scan locations in each applicable survey unit at locations chosen at random. The results of the replicate measurements are directly compared to the result of the original measurement. The acceptance criteria for replicate scan surveys are that the same conclusion is reached for each measurement location (e.g., alarm or no alarm), and that no additional locations greater than the scan investigation level for the area classification are found. If the same conclusion is not reached, or any exceptions are reported that were not reported in the original survey, further evaluations will be performed. The acceptance criteria outlined in Section C.4.2.2 of MARLAP are used to determine the acceptability of replicate static measurements.

- If gross activity is below MDC in the standard and replicate measurements, then no assessment will be conducted.
- If gross activity is below MDC in one measurement and above MDC in the other, then no assessment will be conducted. An evaluation of the measurements will be performed.
- If gross activity is above MDC in the standard and replicate measurements, then the assessment will be performed in accordance with MARLAP. If the assessment results in a failure, an investigation will be performed.

5.6.3.1.1 <u>QC Replicate Surveys</u>

The QC replicate surveys only apply to the FSS of below-ground basements, above-ground buildings, embedded piping, and buried piping. Generally, QC replicate surveys will be performed on randomly selected survey units from the known population of survey units. Some circumstances, such as when the survey of a unit may require burdensome support activities or use of specialty equipment, may dictate selecting an alternate survey unit for the QC replicate survey. QC replicate surveys are designed and modeled in the same manner as the original FSS. The replicate sample plan will be an addendum to the original sample plan.

The acceptance criteria for QC replicate surveys is that both data sets either pass or fail the Sign test for that survey unit. Agreement is ultimately determined that the same conclusion is reached for each data set. If the same conclusion is not reached or any exceptions are reported that were not reported in the original survey, further evaluations will be performed.

5.6.3.2 Duplicate and Split Samples

The collection of duplicate samples or split samples will be the primary means of assessing survey precision and accuracy when collecting volumetric and/or material samples for FSS. A duplicate sample is a second complete sample taken at the same location and approximately the same time as the original. A split sample is when the original sample aliquot is separated into two aliquots and analyzed as separate samples. For the FSS of surface and subsurface soils, asphalt, and sediment, a split sample analysis will be performed on 5 percent of the soil samples taken in a survey unit with the locations selected at random. Duplicate samples will be acquired in accordance with the direction in the specific survey package or sample plan. In addition, approximately 5% of the total number of split samples taken will be sent for analysis by a qualified off-site laboratory or separate sample analysis by the on-site laboratory using a separate detector.

Section C.4.2.2 of MARLAP will be used to determine the acceptability of split and duplicate sample analyses. The sample results will be compared to determine accuracy and precision as follows:

- If Cs-137 is not identified in the standard and comparison samples, then no assessment will be conducted.
- If Cs-137 is identified in one and not the other, then no assessment will be conducted, provided the Cs-137 activity is less than the 95% UCL for background Cs-137, which at

FCS is equivalent to 0.599 pCi/g. If the Cs-137 activity is greater than 95% UCL for background, an evaluation of the samples will be performed.

- If Cs-137 is mutually identified at levels below the 95% UCL, then MARLAP Section C.4.2.2 will be used for the assessment. This will be a trending assessment that will not result in failure.
- If Cs-137 is mutually identified at levels above the 95% UCL, then MARLAP Section C.4.2.2 will be used for the assessment. If the assessment results in a failure, an investigation will be performed.

5.6.3.3 Field Blanks and Spiked Samples

Field blanks will not be performed on a routine basis. The purpose of obtaining field blanks is to detect and identify any contaminant from the sampling site. Field blanks may be prescribed if cross-contamination during the sampling process is suspected.

The acceptance criteria for field blank samples are that no plant derived radionuclides above background are detected. If the analysis of the field blank shows the presence of plant derived radionuclides, then further evaluations will be performed.

Spiked samples check the matrix of the sample itself to see if its composition is impacting the analysis method. In lieu of performing spiked samples as part of the QC program, the on-site laboratory participates in an analytics cross check program, with testing occurring quarterly. An independent laboratory creates spiked samples in typical power plant matrixes for blind testing. The site’s effluent and environmental programs vendors also participate in these programs and their results are reviewed by the site.

5.6.3.4 QC Investigations

If QC replicate measurements or sample analyses fall outside of their acceptance criteria, a documented investigation will be performed in accordance with approved procedures and if necessary, shall warrant a condition report. The investigation will include verification that the proper data sets were compared, the relevant instruments were operating properly and the survey/sample points were properly identified and located. Relevant personnel will be interviewed to determine if proper instructions and procedures were followed and proper measurement and handling techniques were used including CoC, where applicable. If the investigation reveals that the data is suspect and may not represent the actual conditions, additional measurements will be taken. Following the investigation, a documented determination is made regarding the usability of the survey data and if the impact of the discrepancy adversely affects the decision on the radiological status of the survey unit.

5.6.4 Assessment and Oversight

5.6.4.1 Assessments

Management assessments of FSS activities will be performed in accordance with applicable procedural guidance. The findings will be tracked and trended.

5.6.4.2 Independent Review of Survey Results

Randomly selected survey packages (approximately 5 percent) from survey units will be independently reviewed to ensure that the survey samples/measurements have been taken and documented in accordance with approved procedures.

5.6.4.3 Corrective Action Process

The corrective action process, already established as part of the site QA Program, will be applied to FSS for the documentation, evaluation, and implementation of corrective actions. Reports of audits and trend data will be reported to management in accordance with the QAPP and approved procedures.

5.6.5 NRC Confirmatory Measurements

The NRC may take confirmatory measurements to assist in deciding, in accordance with 10 CFR 50.82(a)(11), that the FSS and associated documentation demonstrate the site is suitable for release in accordance with the criteria for decommissioning in 10 CFR 20.1402. Confirmatory measurements may include collecting radiological measurements for the purpose of confirming and verifying the adequacy of the FCS FSS measurements. Timely and frequent communications with the NRC will ensure it is afforded sufficient opportunity for these confirmatory measurements prior to implementing any irreversible decommissioning actions.

5.7 Final Radiation Survey Reporting

Documentation of the FSS will be contained in two types of reports and will be consistent with section 8.6 of MARSSIM. A Survey Unit Release Record will be prepared to provide a complete record of the as-left radiological status of an individual survey unit, relative to the specified release criteria. Survey Unit Release Records will be made available to the NRC for review as appendices to the appropriate FSS Final Report. An FSS Final Report, which is a written report that is provided to the NRC for its review, will be prepared to provide a summary of the survey results and the overall conclusions which demonstrate that the site, or portions of the site, meets the radiological criteria for unrestricted use, including ALARA.

It is anticipated that the FSS Final Report will be provided to the NRC in phases as remediation and FSS are completed with related portions of the site. The phased approach for submittal is intended to provide NRC with detailed insight regarding the remediation and FSS early in the process, to provide opportunities for improvement based on feedback, and to support a logical and efficient approach for technical review and independent verification.

5.7.1 FSS Unit Release Records

A Survey Unit Release Record will be prepared upon completion of the FSS for a specific survey unit. Sufficient data and information will be provided in the release record to enable an independent re-creation and evaluation at some future time. The Survey Unit Release Record will contain the following information:

- survey unit description, including unit size, descriptive maps, plots or photographs and reference coordinates

- classification basis, including significant HSA and characterization data used to establish the final classification
- DQOs stating the primary objective of the survey
- survey design describing the design process, including methods used to determine the number of samples or measurements required based on statistical design, the number of biased or judgmental samples or measurements selected and the basis, method of sample or measurement locating, and a table providing a synopsis of the survey design
- survey implementation describing survey methods and instrumentation used, accessibility restrictions to sample or measurement location, number of actual samples or measurements taken, documentation activities, QC requirements and scan coverage
- results of FSS including types of analyses performed, types of statistical tests performed, surrogate ratios, statement of pass or failure of the statistical test(s)
- survey results for RA and RASS, if performed
- QC results to include discussion of split samples and/or QC replicate measurements
- results of any investigations
- any remediation activities, both historic and resulting from the performance of the FSS
- any changes from the FSS survey design including field changes
- DQA conclusions
- any anomalies encountered during performance of the survey or in the sample results
- conclusion as to whether or not the survey unit satisfied the release criteria and whether or not sufficient power was achieved

5.7.2 FSS Final Reports

The ultimate product of FSS is an FSS Final Report which will be, to the extent practical, a stand-alone document. To facilitate the data management process, as well as overall project management, FSS Final Reports will usually incorporate multiple Survey Unit Release Records. To minimize the incorporation of redundant historical assessment and other FSS program information, and to facilitate potential partial site releases from the current license, FSS Final Reports will be prepared and submitted in a phased approach. FSS Final Reports will contain the following information:

- a brief overview discussion of the FSS Program including descriptions regarding survey planning, survey design, survey implementation, survey data assessment, and QA and QC measures
- a description of the site, the applicable survey area(s) and survey unit(s), a summary of the applicable HSA information, conditions at the time of survey, identification of potential contaminants, and radiological release criteria

- a discussion regarding the DQOs, survey unit designation and classification, background determination, FSS plans, survey design input values and method for determining sample size, instrumentation (detector efficiencies, detector sensitivities, instrument maintenance and control and instrument calibration), ISOCS Efficiency Calibration geometry, survey methodology, QC surveys, and a discussion of any deviations during the performance of the FSS from what was described in this LTP
- a description of the survey findings including a description of surface conditions, data conversion, survey data verification and validation, evaluation of number of sample/measurement locations, a map or drawing showing the reference system and random start systematic sample locations, and comparison of findings with the appropriate Operational DCGL or Action Level including statistical evaluations
- description of any judgmental and miscellaneous sample data collected in addition to those required for performing the statistical evaluation
- description of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of the Operational DCGL
- if survey unit fails the statistical test, a description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity, the investigation conducted to ascertain the reason for the failure and the impact that the failure has on the conclusion that the facility is ready for final radiological surveys, and a discussion of the impact of the failure on survey design and result for other survey units
- description of how good housekeeping and ALARA practices were employed to achieve final activity levels
- within the last submitted FSS Final Report, the final dose compliance equation will be provided that lists the maximum dose to each of the dose components as described in Section 5.2.6.8 and Equation 5-5. This information may be submitted as an attachment to a letter requesting partial site release

As appendices to the Final Report, the applicable Survey Unit Release Record(s), all applicable implementing procedures and all applicable CPPs or TSDs will be attached. If during a phased submittal, procedures and TSDs are submitted with the initial report, all subsequent submittals will only contain any revisions or additions to the applicable implementing procedures and/or TSDs.

5.8 References

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

[2] U.S. Nuclear Regulatory Commission, "NUREG-1505, Revision 1, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," 1998.

[3] U.S. Nuclear Regulatory Commission, "NUREG-1507, Revision 1, Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," 2020.

[4] U.S. Nuclear Regulatory Commission, "NUREG-1700, Revision 2, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," 2018.

[5] U.S. Nuclear Regulatory Commission, "NUREG-1757, Volume 2, Revision 2, Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria - Draft Report for Comment," 2020.

[6] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.179, Revision 2, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," 2019.

[7] U.S. Nuclear Regulatory Commission, "NUREG-1575, Supplement 1, Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME)," 2009.

[8] U.S. Nuclear Regulatory Commission, "NUREG-1576, Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP)," 2004.

[9] Omaha Public Power District, "FC-20-006, Fort Calhoun Station Building End State Concrete Surface Areas and Volumes".

[10] Omaha Public Power District, "FC-21-002, Description of Embedded Piping, Penetrations, and Buried Pipe to Remain in Fort Calhoun End State".

[11] Radiation Safety and Control Services, "TSD 20-001, Historical Site Assessment for Fort Calhoun Station," 2020.

[12] Radiation Safety and Control Services, "TSD 21-043, Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan," 2021.

[13] U.S. Nuclear Regulatory Commission, "NUREG/CR-5512, Volume 3, Residual Radioactive Contamination From Decommissioning, Parametrer Analysis," 1999.

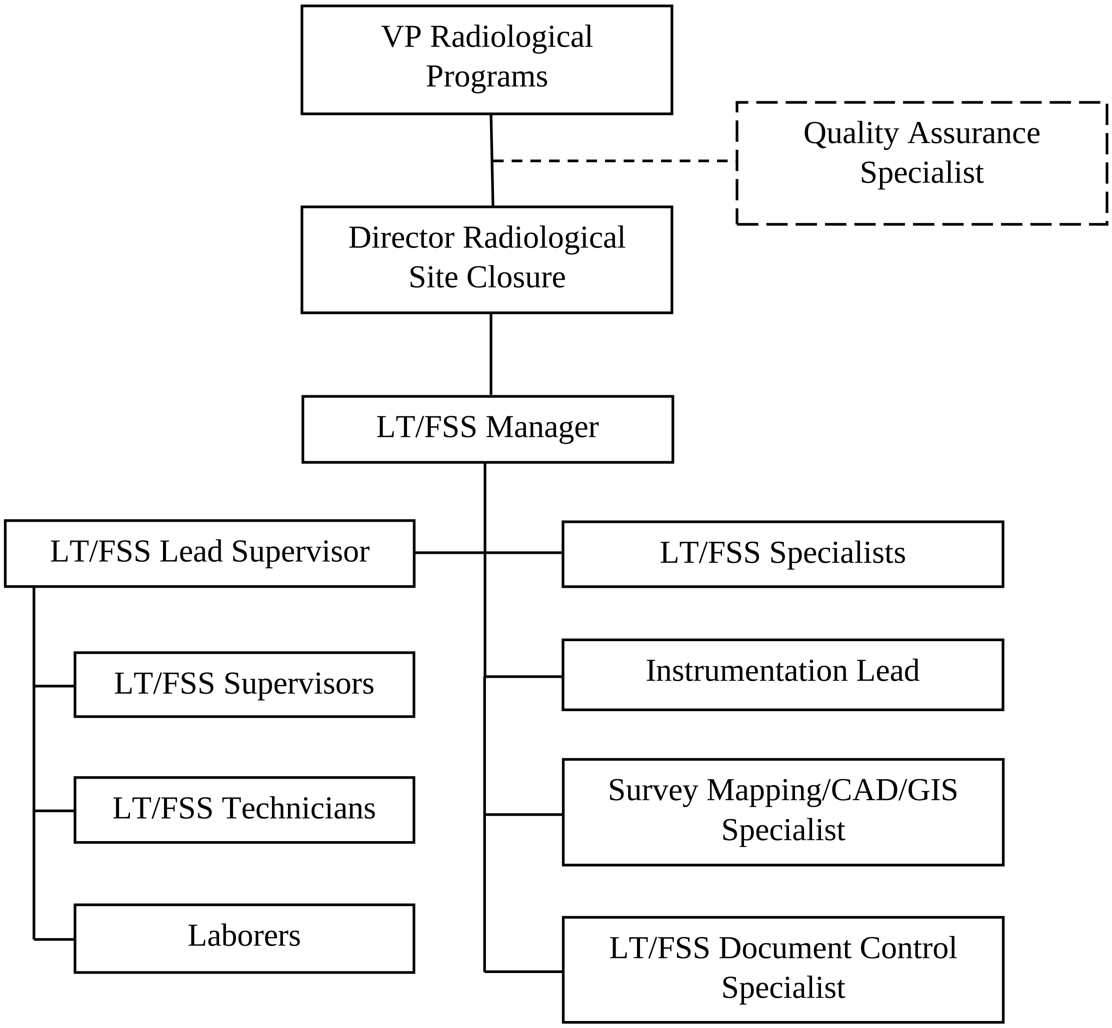
[14] Omaha Public Power District, "FC-19-006, Ludlum Model 44-10 Detector Sensitivity".

[15] Omaha Public Power District, "FCSD-RA-LT-100, Quality Assurance Project Plan for the License Termination Plan Development, Site Characterization and Final Status Survey Projects at Fort Calhoun Station".

[16] International Organization for Standardization, "ISO 7503-1, Part 1, Evaluation of Surface Contamination, Beta-Emitters (maximum beta energy greater than 0.15 MeV) and Alpha-Emitters," 2016.

[17] Omaha Public Power District, "NO-FC-10, Quality Assurance Topical Report".

Figure 5-1 LT/FSS Organization Chart



FORT CALHOUN STATION DECOMMISSIONING PROJECT
LICENSE TERMINATION PLAN

CHAPTER 6
COMPLIANCE WITH RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

REVISION 0

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ABBREVIATIONS

ALARA	as low as is reasonably achievable
AMCG	average member of the critical group
AMSL	above mean sea level
bgs	below ground surface
DA	deconstruction area
DCF	dose conversion factor
DCGL	derived concentration guideline level
DSAR	Defueled Safety Analysis Report
DSR	dose to source ratio
FCS	Fort Calhoun Station
FSS	final status survey
HTD	hard-to-detect
GW	groundwater
HSA	Historical Site Assessment
IC	insignificant contributor
K_d	distribution coefficient
LLBP	less likely but plausible
LTP	license termination plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
OCA	Owner Controlled Area
OpDCGL	Operational DCGL
OPPD	Omaha Public Power District
PDF	Probability Density Function
PRCC	Partial Rank Correlation Coefficient
RESRAD	RESidual RADioactive

6 COMPLIANCE WITH RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

6.1 License Termination Criteria

The Fort Calhoun Station (FCS) decommissioning project will apply the radiological criteria for unrestricted use in Title 10, Section 20.1402, of the Code of Federal Regulations (10 CFR 20.1402).

Dose Criterion: residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water.

ALARA Criterion: residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA).

Chapter 4 describes the methods and results for demonstrating compliance with the ALARA criterion. This Chapter describes the methods for demonstrating compliance with the dose criterion.

6.2 General Site Description

The FCS site is located on the west bank of the Missouri River at river mile 646.0 on approximately 540 acres, approximately 19.4 miles north of Omaha, Nebraska. Figures 6-1 to 6-4 show the FCS location, owner-controlled area, site layout, and site topography. A detailed Conceptual Site Model of hydrogeological conditions at the site is provided in Haley & Aldrich, "Hydrogeological Conceptual Site Model, Rev. 2, Fort Calhoun Station, Blair, NE," 2021 [1]. A summary is provided below.

6.2.1 Geology

The site elevation is 1004 feet above mean sea level (AMSL). The soils below the FCS include thick beds of limestone, dolomite, shale, and sandstone with some thin layers of coal beds. The deeper formations were deposited in marine depositions with the shallow soils from the lateral migration of the paleo river channel. At the beginning of the Pleistocene period, the Missouri River Valley and its main tributaries were established in their approximate present positions. Subsequently under successive glacial movements, the valleys were filled and reopened several times. During this period, the Peorian loess was deposited on the terraces and adjacent uplands. It is probable that only the upper part of the alluvium in the Missouri River Valley is actually of recent age and that deeper deposits are mostly of Pleistocene age.

Unconsolidated sediments at the plant site generally range from 65 to 75 feet in thickness. The soils are typically interstratified and cross-bedded. These soils may be grouped generally into two units:

1) An upper fine-grained sand and silt with some clay, ranging from 20 to 50 feet in thickness.
2) An underlying fine to coarse sand with some gravel, extending down to the relatively

flat-lying carbonate bedrock surface at a depth of approximately 65 to 75 feet below ground surface (bgs).

The upper units represent former river deposits and are not likely continuous, but rather have preferential channels formed by paleo-oxbow deposits. Pennsylvanian-aged limestone and shale (bedrock) of the Kansas City Formation are encountered below the overburden soils. The bedrock below the site consists of various types of limestone formations.

6.2.2 Hydrogeology

From the FCS Historical Site Assessment (HSA) [2], groundwater at the site is in hydraulic communication with the Missouri River, with the water table ranging from 2 to 20 feet bgs depending on the river stage. Water levels taken in a series of borings drilled during July and August 1966, reveal that the ground water levels at the site varied from elevations 993.7 to 992.4 feet AMSL, while the river levels recorded during this same period ranged from elevations 993.2 to 992.4 feet [3]. Based on the gauging stations both upriver and downriver, located approximately 25 miles from the plant in each direction, the historical river stage varies widely and is prone to flooding [1]. Water levels collected in June 2020 show that the groundwater elevations ranged from 993.5 to 995.5 feet [1], i.e., 10.5 to 8.5 feet bgs. Based on water level data collected in June 2020 and comparisons to the gage height of the Missouri River, the onsite water table is estimated to vary between approximately 10 to 15 feet bgs [1] with periods of higher elevations corresponding to river stage.

Groundwater flow directions have been reported to be both toward the Missouri River (northeasterly) and away from the Missouri River (south-southwesterly) depending on the river stage. Shallow flow directions toward the river represent times when the river levels are at normal or low stages. Shallow flow directions away from the river are likely to occur during times of high river stage (i.e., at flood or near flood stage), causing bank storage effects. Based on the groundwater depth to water data, and corresponding groundwater elevation data, groundwater contours are consistent with historical data and are flowing towards the river.

Horizontal hydraulic gradients at the site are nearly flat with relatively slow groundwater velocity, with only a gentle slope toward the Missouri River. Shallow and deep aquifer groundwater contours from June 2020 are presented in Figure 6-5 and Figure 6-6. From elevation data collected in June 2020, the approximate average gradient in both the shallow and deeper system was 0.0008 ft/ft [1].

Based on the classification of fine to medium sands and silts for the shallow soils, expected hydraulic conductivities for the shallow aquifer range from 10^{-5} centimeters per second (cm/sec) (or 10^{-1} feet per day) to 10^{-1} cm/sec (or 100 feet/day). Using the data collected from modified pumping tests and/or slug tests completed in June 2020, calculated hydraulic conductivities ranged from 9.4×10^{-3} to 1.8×10^{-2} cm/sec [1]. The measures values are within the range for these types of soils. The calculated average hydraulic conductivity for the shallow aquifer is 4,352 m/yr [1]. In-situ falling head tests were completed at seven open borehole locations to determine the field coefficient of permeability (or hydraulic conductivity) of unsaturated soils (i.e., vadose zone). The average hydraulic conductivity of the unsaturated zone is 34.4 m/yr [1].

Using groundwater contours, horizontal hydraulic gradients, and hydraulic conductivity information, groundwater velocity data for the shallow and deeper aquifers were calculated [1]. Groundwater velocity calculations are estimated from the June 2020 field measurements and contour plans, where the flow direction is towards the Missouri River. The resulting velocity calculations are therefore representative of this flow direction. Based upon the analysis, the groundwater velocities are very similar, with the shallow system ranging from 5.73 to 34.39 feet per year (ft/yr) and ranging from 5.86 to 35.16 ft/yr in the deeper system [1].

6.2.3 Area Land Use

Area Land use information is summarized from [1] and [3].

The plant site is located on the alluvial plain of the Missouri River in a predominantly agricultural region roughly ten miles north of the Omaha metropolitan area. There are no residences within one half mile of the location. An industrial park is located north of the plant property. Industries include a large corn processing facility, agricultural fertilizer storage facilities, and various other light industrial plants.

The DeSoto National Wildlife Refuge occupies approximately 7,821 acres east of the plant site. This area is open to the public for day use year-round. Visitors to the refuge generally use areas from two to five miles from the plant. The State of Nebraska operates the Fort Atkinson State Historic Park five and one-half miles southeast of the plant site. This day use facility is mostly seasonal. The State of Iowa maintains Wilson Island State Park south of the DeSoto National Wildlife Refuge and four miles southeast of the plant site.

Two private facilities lie to the north of the plant along the Missouri River. The Cottonwood Cove Marina & RV Resort is located approximately four and one-half miles from the plant. River View Park Resort & Marina is a private campground lying directly to the south of Cottonwood Marina and ranging from four to four and one-half miles from the plant.

The nearest municipality is the city of Blair, Nebraska, about three miles northwest, with a population of 7,990 per the 2010 census. Fort Calhoun, Nebraska, is about five miles southeast of FCS. The 2010 census reported a population of 908 in Fort Calhoun and 167 in Kennard Village, about seven miles from the plant site.

The Omaha metropolitan area includes the cities of Omaha, Nebraska, and Council Bluffs, Iowa, and the adjoining areas of Douglas, Washington, and Sarpy Counties, Nebraska, and Pottawattamie County, Iowa. The area lies 10 to 25 miles southeast of the site, with the main concentration of population beyond 15 miles.

The U.S. Census data shows an increase in population in the Omaha metropolitan area and in most of the nearby cities but a decrease in the rural and farm population. While it is probable that the area around the plant site outside of the Omaha metropolitan area will remain largely agricultural and that the population will increase slowly, a general decline of the rural population will continue, reflecting the movement of people into towns and cities. The expansion of the Omaha metropolitan area has been generally south and westward, coinciding with the interstate highway. It is expected that future growth of the metropolitan area will continue south and west and also northwestward. Thus, it is probable that the area surrounding the plant site will continue to remain largely agricultural.

6.2.4 Area Groundwater Use

A search on the Nebraska Department of Natural Resources Permits & Registrations Division, Groundwater Section database and web viewer, and the Iowa Geological Survey web viewer, yielded 31 active domestic/private wells and 13 irrigation wells [1]. The well search was concentrated in the flood plain approximately 2.5-miles upstream and 4.5-miles downstream from the FCS site. There were several additional wells that are used for public water supply (for recreation areas), environmental monitoring, or commercial/industrial operations but these uses are not applicable to the compliance scenario conceptual model of residential farming (see Section 6.4.2).

Fifteen of the 31 domestic wells encountered either shale (six wells between 130 to 255 feet bgs with an average static water level of approximately 137.7 feet bgs) or limestone (nine between 34 to 279 feet bgs with an average static water level of approximately 75.2 feet bgs). Each of the 31 screens were installed in sand or straddled a sand/clay interval. Of the 13 irrigation wells that were mainly located in Iowa, six encountered bedrock. The deepest irrigation well encountered shale at approximately 180 feet bgs and was screened in sandy clay from 165 to 175 feet bgs. The static water level for this well was approximately 148 feet bgs. Five other irrigation wells encountered limestone ranging from approximately 68 to 110 feet bgs. These wells had an average static water level of approximately 7.4 feet bgs. The bottom of the well screens ranged from 81 to 110 feet bgs.

6.3 End State at License Termination

The “End State” is defined as the configuration of the remaining below-ground basements, above-ground buildings, piping, and open land areas at the time of license termination. Chapter 3 of this LTP provides a more detailed discussion of the FCS end state.

Six above-ground buildings will remain at the time of license termination: the Training Building, FLEX Building, Owner Controlled Area (OCA) Entrance Building, Switchyard 3451 Old Building, Switchyard 3451 New Building, and the 1251 Control and Switchgear Building. The Switchyard will also remain untouched by decommissioning. All other structures will be removed to a minimum of three feet below grade (approximately 1,001 feet AMSL). The basements of the Turbine Building, Containment Building, Auxiliary Building, Intake Structure, and Circulating Water Tunnels will remain with all interior walls removed, with the exception of the Turbine Building where the turbine pedestal will remain up to 3’ below grade. All other structures will be removed in their entirety with the exception of the six above-ground buildings listed above. Once D&D activities are completed, as well as subsequent FSS, each basement will be backfilled to grade level (approximately 1,004 feet AMSL). The general basement configurations, basement elevations, and surface areas are shown in Figures 6-7 to 6-17.

The end state will also include a range of buried pipe and embedded pipe. The inventory of buried pipe and embedded pipe to remain is provided in Omaha Public Power District (OPPD) Calculations and Position Paper (CPP) FC-21-002, “Description of Embedded Piping, Penetrations, and Buried Pipe to Remain in Fort Calhoun End State” [4].

OPPD added three rail lines to the rail spur to allow for the direct loading of waste into rail cars. The added rail lines will remain in place after license termination with the exception of the portion of the line that resides within the waste processing structure.

6.4 Exposure Scenario, Critical Group, and Pathways

6.4.1 Reasonably Foreseeable Exposure Scenarios, Critical Groups and Pathways

From the review of area land use in Section 6.2.3 six reasonably foreseeable scenarios were identified. The land use scenarios, environmental pathways, exposure pathways and associated average member of the critical group (AMCG) are summarized in Table 6-1. The aquatic pathway from an onsite pond is not credible due to engineering and cost issues of construction and proximity to the Missouri River which negates any foreseeable need. Therefore, ingestion of aquatic foods is not included as an exposure pathway in any of the land use scenarios.

6.4.2 Compliance Exposure Scenario

A qualitative analysis of the reasonably foreseeable scenarios in Table 6-1 concludes that the residential farming scenario will result in the highest dose. All other scenarios either have environmental pathways removed or lower occupancy times and food ingestion rates. The one possible change to the impact of an environmental pathway in the various scenarios is the well water concentration when the well pumping rate is lower or higher. For example, the residential scenario would have a lower well pumping rate than the residential farming scenario due to decreased irrigation. The recreational and industrial scenarios could have higher pumping rates. The pumping rates were evaluated through sensitivity analysis and the well water concentrations found to be insensitive to pumping rate (see Section 6.11.3).

The bounding residential farming exposure scenario is applied to demonstrate compliance with the dose criterion for soil, backfilled basements (including embedded pipe and penetrations), backfill soil, buried pipe, and existing groundwater. The bounding building occupancy exposure scenario is applied to demonstrate compliance for above ground buildings. The doses attributable to each media - soil, backfilled basements (including embedded pipe and penetrations), backfill soil, buried pipe, above ground buildings, and existing groundwater - are summed to ensure that the compliance dose is prudently conservative (see Section 0).

6.4.3 Compliance Scenario Critical Group and Exposure Pathways

The AMCG is a residential farmer that resides on the site and derives a large fraction of annual food intake from onsite agriculture and livestock. The environmental pathways and exposure pathways provide links between the radiological source, the transport of contaminants within environmental media, the critical group receptor location, and behaviors of the receptor that lead to exposure to residual radioactivity through direct exposure, inhalation, and ingestion of water, soil, plants, crops, meat, and milk. The media (sources of residual radioactivity), environmental pathways, and exposure pathways for the residential farmer compliance scenario are listed in Table 6-2.

Table 6-1 Reasonably Foreseeable Land Use Scenarios, Critical Groups. and Pathways

Environmental Pathway	Exposure Pathway	Land Use Scenario: Residential Farming	Industrial (Light Industry)	Industrial (Commercial Agriculture)	Residential	Urban Residential	Recreational
Direct Exposure	External Radiation	✓	✓	✓	✓	✓	✓
Airborne (Particulate, H-3)	Inhalation	✓	✓	✓	✓	✓	✓
Plant Foods	Ingestion	✓			✓		
Livestock - Meat	Ingestion	✓					
Livestock - Milk	Ingestion	✓					
Onsite Groundwater	Ingestion	✓	✓	✓	✓		✓
Direct Soil Contact	Ingestion	✓	✓	✓	✓	✓	✓
Critical Group		Adult that resides onsite and derives a large fraction of annual food intake from onsite agriculture and livestock.	Adult that works onsite full time, predominantly indoors, performing light industrial activities.	Adult worker that periodically occupies the onsite area which is part of a larger commercial farming operation	Adult resident that derives a small fraction of annual food intake from an onsite garden. No livestock raised onsite.	Adult resident that does not maintain a garden or livestock. Drinking water is supplied by an offsite municipal source	Adult that periodically accesses the site (after being converted to parkland or recreation area) for hiking, camping, etc. No food is harvested from the site but an onsite well is drinking water source.

Table 6-2 Compliance Scenario Environmental Pathways and Exposure Pathways

No.	Source	Environmental Pathway(s)	Exposure Pathway
1	Soil	resuspension of soil into air (airborne particulate)	Inhalation
2	Soil	onsite direct exposure	External radiation
3	Soil	soil to edible plant	Ingestion of plants
4	Soil	soil to forage plant → beef cow	Ingestion of meat
5	Soil	soil to forage plant → dairy cow	Ingestion of milk
6	Soil	soil to groundwater → well → plant by irrigation	Ingestion of plants
7	Soil	soil to groundwater → well → forage by irrigation → beef cow	Ingestion of meat
8	Soil	soil to groundwater → well → forage by irrigation → dairy cow	Ingestion of milk
9	Soil	soil to groundwater → well → drinking water	Ingestion of drinking water
10	Soil	soil to hand → mouth	Ingestion of soil
11	Buried Pipe	activity from internal surface of buried pipe to subsurface soil → environmental pathways 2-9	Direct exposure, ingestion of plants, meat, milk, and drinking water
12	Buried Pipe	excavation of buried pipe → activity from internal surface of buried pipe to surface soil → environmental Pathways 1-10	Exposure Pathways 1-10

No.	Source	Environmental Pathway(s)	Exposure Pathway
13	Backfilled Basement Surfaces (walls/floors)	release from concrete to fill material → equilibrium desorption from fill to pore water → pore water to well → environmental pathways 2-9 (with fill replacing soil)	Direct Exposure, ingestion of plants, meat, milk, and drinking water
14	Backfilled Basement Surfaces (walls/floors)	basement concrete incorporated with drilling spoils during well installation → drilling spoils to ground surface and treated as soil → environmental pathways 1-10	Exposure Pathways 1-10
15	Backfilled Basement Concrete	excavation of basement concrete → concrete debris spread on ground surface → concrete debris treated as soil → environmental pathways 1-10	Exposure Pathways 1-10
16	Basement Embedded Pipe	activity from internal surface of embedded pipe released to fill → equilibrium desorption from fill to pore water → pore water to well → environmental pathways 6-9 (with fill replacing soil)	Ingestion of plants, meat, milk, and drinking water
17	Existing Groundwater	groundwater to well → well water pathways 6-9	Ingestion of plants, beef, milk, and drinking water
18	Basement Fill	equilibrium desorption from fill to pore water → pore water to well → environmental pathways 2-9 (with fill replacing soil)	Direct Exposure, ingestion of plants, meat, milk, and drinking water

6.4.4 Less likely But Plausible Exposure Scenarios

There are two exposure scenarios that are categorized as less likely but plausible (LLBP). The first is exposure to contaminated drilling spoils brought to the ground surface during installation of a water well through the Auxiliary Building or Turbine Building basement that contacts an embedded pipe. The receptor is the resident farmer AMCG. The dose to a worker will be less due to decreased occupancy time and elimination of food pathways. Note that exposure to drilling spoils where the source term is contaminated concrete is included as a compliance scenario (see Section 6.11.2).

The second LLBP exposure scenario is offsite processing/recycling/disposal of excavated basement concrete and Containment Building steel liner. The offsite processing/recycle/disposal dose is calculated using the dose factors for concrete and steel in NUREG-1640, "Radiological Assessments for Clearance of Materials from Nuclear Facilities" Table 2.1 [5]. There are multiple exposure scenarios (and receptors) evaluated in [5] such as dismantling and handling while decommissioning, exposure to waste piles, transport to the recycle or disposal site, disposal at landfill, and commercial reuse of the materials.

The LLBP scenarios are evaluated in Section 6.21.

6.5 Radionuclides Potentially Present (Initial Suite) and Mixture Fractions

The selection of the radionuclides potentially present at the FCS site (initial suite) included identifying the radionuclides as well as eliminating those radionuclides which would not be present due to decay or may be present but in insignificant concentrations. A systematic approach was taken including reviewing applicable nuclear industry guidance documents, relevant FCS specific historical information and representative available sample radionuclide data.

A preliminary evaluation was performed in Omaha Public Power District, "FC-18-002, Potential Radionuclides of Concern During the Decommissioning of Fort Calhoun Nuclear Station," 2018 [6] to document an initial suite of radionuclides for the decommissioning of FCS. From November 2019 through March 2020, FCS implemented a comprehensive concrete characterization program. From this sample population, 37 samples were then sent to an offsite laboratory for analysis.

FC-20-007 "Fort Calhoun Station Potential Radionuclides of Concern" [7] used the offsite laboratory results to identify radionuclides present in detectable concentrations. The results were compared to the initial listing from FC-18-002 and an initial suite of radionuclides selected for Derived Concentration Guideline Level (DCGL) development.

Technical Support Document FC-21-043, Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan" [8] used the laboratory results for the initial 37 samples, plus the results for 20 additional samples to calculate the mixture fractions. The mixture fraction of each radionuclide in the initial suite at the time of license termination (October 5, 2026) was determined using the results of the 57 concrete characterization sample analyses.

Two sample populations were analyzed. The first population consisted of the Containment Building samples. The second population consisted of the Auxiliary Building, Turbine Building and Radioactive Waste Processing Building (AB/TB/RWPB) samples. The Containment Building and AB/TB/RWPB sample populations were analyzed separately to recognize the potential that these may contain different radionuclide mixtures.

For each of these two populations, the radionuclide fractions were determined from the decayed analytical data using three separate approaches. The three methods were chosen to represent common and conservative methods in determining activity fractions to ensure that the final selection of the radionuclide mixtures account for variability and represent conservative estimates. Each of the analysis methods use the actual reported laboratory values, whether detected or less than the reported MDC values. The results were then decay-corrected from the date of sample analysis to the anticipated date of license termination. The Am-241 decay corrected value was also adjusted to account for in-growth from Pu-241.

The first approach was to calculate the radionuclide activity fraction for each sample, each radionuclide, within each population from the reported radionuclide activity concentrations and then calculating the average activity fraction for each radionuclide, and population of samples.

The second approach was to calculate the 75th percentile of the population of samples. Once the 75th percentile fraction were calculated for each radionuclide, the data set was re-normalized to determine the percentile-based activity fractions.

The third approach was to calculate the individual radionuclide ratios to Cs-137 for each sample, calculate the 75th percentile for the sample group, then renormalize to determine the activity fractions.

The analyses described above removes the activity weighting and gives equal statistical weight to each of the sample results. Based on the evaluation using the three statistical methods, the two methods involving the use of the 75th percentiles result in very similar results.

Using the 75th percentile provides sufficient overall conservatism in the development of the radionuclide mixtures. In addition, there are precedents at other decommissioning sites in using the 75th percentile, particularly in the parameter selection process for dose modeling to support DCGL calculations.

The '75th percentile of the Cs-137 fractions' is selected to represent the overall nuclide mix for the Containment Building sample population and the AB/TB/RWPB sample population. See Table 6-3 for the initial suite radionuclides and mixture fractions. The Containment mixture will be applied to Containment. The AB/TB/RWPB mixture will be applied to all basements other than the Containment Building as well as embedded pipe, soil, buried pipe, and above ground buildings.

Table 6-3 Initial Suite Radionuclide Mixture Using the '75th Percentile of the Cs-137 Fractions' Approach

Nuclide	Containment Basement Sample Population Mixture Fraction	AB/TB/RWPB Sample Population Mixture Fraction
H-3	8.70E-02	2.52E-02
C-14	8.27E-01	5.14E-02
Fe-55	1.43E-05	2.33E-03
Co-58	1.47E-14	7.00E-14
Ni-59	1.53E-04	2.48E-03
Ni-63	1.76E-02	3.43E-01
Co-60	1.67E-03	1.15E-02
Tc-99	3.41E-04	5.23E-04
Sr-90	2.59E-04	2.51E-03
Sb-125	1.89E-05	1.65E-04
Cs-134	2.08E-05	1.67E-04
Cs-137	6.01E-02	5.55E-01
Ce-144	1.52E-06	3.36E-06
Eu-152	2.95E-03	7.15E-04
Eu-154	1.44E-04	1.13E-04
Eu-155	4.09E-05	2.72E-04
Pu-238	5.66E-06	3.01E-05
Pu-239/240	1.06E-05	2.47E-05
Pu-241	2.91E-03	4.77E-03
Am-241	4.78E-05	1.32E-04
Cm-243/244	1.27E-05	1.51E-05
Np-237	1.24E-06	0.00E+00

6.6 Radionuclide Chemical Form

The characterization process did not include analysis of source term chemical forms [9]. In the absence of such information, the dose modeling and DCGL calculations use the bounding chemical form(s), e.g., the chemical form(s) that give the individual the highest dose per unit intake as provided in Federal Guidance Report Number 11.

6.7 Conceptual Site Model – Major Assumptions and Critical Group

This section describes the characteristics of the conceptual model that apply to all of the contaminated media on the site with the exception of above ground buildings. The media are basement concrete (and remaining Containment liner), embedded pipe, buried pipe, soil, and existing groundwater. Screening values are applied to above ground buildings (see Section 6.17). The source term abstraction and release, as well as certain aspects of the conceptual model, are media specific as described in Sections 6.9 to 6.14.

The AMCG is a resident farmer that resides in a house constructed on the site and spends the majority of the year onsite conducting subsistence farming activities. The contamination is transported from soil to plants and from plants to animals. The contaminated plants, meat and milk are subsequently ingested by the AMCG. Contamination is also transported and ingested through hand to mouth behavior by the AMCG. Groundwater that becomes contaminated through the infiltration of precipitation and irrigation is captured by an onsite well and used for drinking water and irrigation (see environmental and exposure pathways in Table 6-2).

Fort Calhoun Station is situated on low flat land bordering the Missouri River. Because of its location on the flood plain of the river, the topography in the area of the power plant is relatively flat. The resulting hydraulic gradient within the unconsolidated sediments in the flood plain is also relatively flat. This low hydraulic gradient, combined with moderate hydraulic conductivity of the generally fine-grained alluvial aquifer material, results in relatively slow ground water flow velocity beneath the site [2]. The site-specific hydraulic gradient in $8.4x10^{-4}$ [1]. The unsaturated zone soil type is silt loam [1]. The saturated zone soil type is sand [1]. The site-specific hydraulic conductivities are 34.4 m/yr and 4350 m/yr, for unsaturated and saturated zones, respectively [1].

Soil analyses were conducted to determine site-specific values for density, total porosity, effective porosity, and field capacity [1]. The values for the soil parameters are provided in Attachment 6.1 which also contains values for the remaining physical parameters not mentioned here, such as irrigation and precipitation, and the metabolic and behavioral parameters of the AMCG. With the specific exceptions listed for each media in Sections 6.11, 6.13, and 6.14, the values listed in Attachment 6.1 apply to all media. The size of the conceptual site is assumed to be the total area of the FCS Class 1 open land areas, i.e., 79,600 m^2.

Onsite water levels collected from 17 wells in June 2020 ranged from 10.5 to 8.5 feet bgs (3.1 to 4.6 m). The HSA states that the water table ranges from 2 to 20 ft bgs [2]. However, the 2-foot bgs value is expected to correspond to a high stage of the Missouri River and therefore to occur infrequently. The CSM [1] states that a water table elevation range of 10 to 15 ft bgs is reasonable based on the review of site-specific measurements and Missouri River stage data. The conceptual site model conservatively assumes that the water table elevation is 1.1 m bgs as discussed below.

The vadose zone thickness at the site is assumed to be 1.1 m which was selected initially to accommodate a conceptual soil source term thickness of 1 m with a nominal unsaturated zone thickness of 0.1 m (see Section 6.9.2). Based on the groundwater elevations at the site as discussed above, it is not unreasonable to assume that the groundwater table will periodically be as high as 1.1 m bgs although the average levels are expected to be in the range of 10-15 ft bgs. The 1.1 m vadose zone thickness is also applied in the conceptual models for the other contaminated media (basement concrete (and Containment liner), embedded pipe, and buried pipe). Assuming a water table elevation at the high end of the site range (1.1 m bgs) is conservative because it reduces transport time and minimizes the effect of dispersion and decay in the calculation of groundwater concentrations. In addition, a water table at 1.1 m bgs results in the source terms for the *insitu* buried pipe scenario, *insitu* basement concrete scenario, and embedded pipe scenario being fully submerged in groundwater which increases the groundwater concentration compared to a source term that is contained in the vadose zone.

6.8 General Mathematical Model (RESRAD ONSITE 7.2)

The RESRAD ONSITE Version 7.2 (RESRAD) code contains conceptual models and corresponding mathematical formulations that are compatible with the FCS conceptual model and includes dose calculations for all of the exposure pathways listed in Table 6-2. The environmental pathways listed in Table 6-2 are also addressed directly in RESRAD with the exception of the initial source term release mechanisms for basement concrete (and steel liner in Containment), embedded pipe, and buried pipe. The contamination in these media require an initial release and reconfiguration of the source term to be compatible with RESRAD.

For example, the source term in buried pipe is contamination on the internal surface of the pipe. This is not compatible with RESRAD which requires a uniform, volumetric contaminated zone (source term) with soil-like properties. In this case, Excel is used to calculate the concentration in a unit volume of soil after the instant release of a unit concentration (1 pCi/cm^2) from the pipe and mixing with the soil (see Section 6.14). The volume and geometry of the soil after release and mixing are then used in RESRAD with a unitized concentration (1 pCi/g) to calculate dose to source ratios (DSR) (mrem/yr per pCi/g). The final step is to calculate DCGLs using the unitized soil concentration (pCi/cm^2 per pCi/g in the case of buried pipe) and the DSRs (mrem/yr per pCi/g).

The calculation of initial source term release to soil and reconfiguration for entry to RESRAD are required for the following media and environmental pathways:

- Basement wall/floor - release from concrete (and steel liner in Containment) to fill material (see Section 6.11 for details)
- Basement wall/floor – release of concrete contamination to surface soil via drilling spoils (see Section 6.11 for details)
- Basement walls/floor – excavation of basement concrete and deposition on the ground surface (see Section 6.11 for details)
- Basement walls/floor – excavation of steel liner in Containment and release to surface soil (see Section 6.11 for details)
- Basement embedded pipe - release of internal surface contamination to fill material (see Section 6.13 for details)
- Buried pipe - release of internal surface contamination to subsurface soil (see Section 6.14 for details)
- Buried pipe – release of internal surface contamination to surface soil (see Section 6.14 for details)

In summary, RESRAD ONSITE Version 7.2 code is used to calculate dose to the AMCG and the corresponding DSRs or DCGLs. The list of RESRAD reports generated for DCGL development is provided in Attachment 6-3. The reports are provided in a digital format. Excel spreadsheet calculations are used to produce source term abstractions for media other than soil that are suitable for use in RESRAD. The details of the media-specific conceptual models and mathematical models are described in Sections 6.9 to 6.18.

6.9 Soil DCGL

6.9.1 Soil Source Term Configuration and Spatial Variability

The findings of the HSA [2] and the results of the soil characterization campaign provide the basis for projecting the soil source and source term characteristics anticipated to exist at the time of FSS and site release for unrestricted use.

Based on a review of documented spills, outage and maintenance activities, and routine operational activities, the HSA classified the majority of the land in the protected area yard as MARSSIM Class 3 indicating that there is a low potential for residual radioactivity to exceed a small fraction of the DCGL. The yard area surrounds the entire Containment Building, Auxiliary building and support buildings including the intake structure and new warehouse. Spills early in plant operations introduced radioactivity to the railroad siding outside of the Auxiliary Building and to the area that is now covered by the radioactive waste building. Records indicate the immediate remediation and disposal of contaminated rail siding soil and material. The HSA categorizes the area as being "partly remediated" [2]. The HSA designates the areas affected by the spills as MARSSIM Class 2 based on a presumption that if radioactivity is present, it will not exceed the DCGLs [2].

Open land areas were investigated during the FCS characterization campaign [9]. The survey unit classifications for characterization sample size and scan coverage determination generally followed the HSA recommendations except the open land area within the protected area was designated as Class 1 as opposed to Class 2 or Class 3. No plant derived radionuclides were identified in any of the surface or subsurface soil samples collected from the former protected area or deconstruction Area (DA) with the exception of Cs-137 which was present at very low concentrations and indicative of background levels. No plant derived radionuclides were identified in any of the Class 2 or 3 open land areas outside of the DA with the exception of very low Cs-137 concentrations as in the DA.

In conclusion, it appears unlikely that significant residual radioactivity is present in the soil of the FCS site open land areas or will be present at license termination. Although there are areas that remain to be characterized such as the soil under the Containment Building, Turbine Building, and Spent Fuel Pool, the extent of soil contamination, if any, will be localized.

6.9.2 Soil Conceptual Model, Source Term Abstraction and Release

There are no additional physical conditions, pathways, or AMCG behavior associated with the soil dose assessment that are not addressed in Section 6.6 or Table 6-2. The soil contamination is assumed to be uniformly distributed at depths of 1 m and 0.15 m. The 1 m thickness is included along with the standard 0.15 m to address the possibility that soil contamination is identified during FSS that is between 0.15 and 1 m thick. In both cases, the unsaturated zone depth is 0.1 m. The contaminated area size is 79,600 m^2 which includes the entire FCS DA and the waste haul path and loading area (the FCS Class 1 areas). The waste haul path area is assumed to be contiguous with the DA for the purpose of selecting the contaminated area size.

As discussed in Section 6.9.1, no soil contamination was identified during characterization. If contamination is identified, the affected area is likely to be much smaller than the 20,000 m^2 area

RESRAD requires for the full meat and milk pathway and also likely to be smaller than the 1,000 m^2 area required for the full plant pathway. Assuming a 79,600 m^2 contaminated area is a conservative conceptual source term geometry.

There are two source term release pathways: resuspension to air and leaching to groundwater. Release of radionuclides by water is assumed to be a function of a constant infiltration rate, constant moisture content, and equilibrium desorption. The release of radioactivity to the unsaturated zone and groundwater is controlled by infiltration, driven by precipitation and irrigation, through the unsaturated soil from the ground surface down to the water table and by groundwater flow at depths below the water table. The partitioning of contaminants between the solid and aqueous phase is assumed to be a linear process controlled by the distribution coefficient. The airborne release is accounted for by assuming a constant mass-loading of respirable particles in air with resuspension and deposition in equilibrium.

6.9.3 Soil Mathematical Model

The RESRAD code contains conceptual models and corresponding mathematical formulations that are compatible with the soil conceptual model. RESRAD directly calculates the dose from environmental and exposure pathways 1 to 10 in Table 6-2.

6.9.4 Soil RESRAD Input Parameters for Uncertainty Analysis

Uncertainty analysis is performed to ensure that conservative values are selected for parameters that have a relatively high correlation to dose. Table 6-4 provides the input parameters used to perform the uncertainty analysis for the soil dose assessment. The deterministic parameter selection process is discussed below.

NUREG/CR-6697 "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes" (NUREG/CR-6697) [3], Appendix B, evaluates the relative sensitivity of the approximately 200 parameters in the RESRAD model. The results of the evaluation are provided in NUREG/CR-6697, Appendix B, Table 4.2 which categorizes parameters as Priority 1, 2 or 3 with 1 being most sensitive and 3 being least sensitive. Consistent with NUREG-1757 guidance, deterministic parameters are selected for behavioral, metabolic and Priority 3 physical parameters. A number of the Priority 1 and 2 physical parameters were assigned the probability density functions (PDF) from NUREG/CR-6697 (RESRAD default PDF). Several Priority 1 and 2 parameters were assigned site-specific deterministic values. A few parameters were assigned site-specific PDFs. The flow chart in Figure 6-17 describes the parameter selection process.

For distribution coefficients (K_d), the soil-specific PDFs listed in Reference [10], Tables 2.13.1, 2.13.2, and 2.13.10 are used for the uncertainty analysis based on the site-specific soil types. The contaminated zone and unsaturated zone are both silt loam soil type [1]. The soil type for the saturated zone is sand [1]. The median of the K_d distribution is applied as the deterministic value for the uncertainty analysis but the final K_ds applied in the DCGL calculation are the deterministic values selected through uncertainty analysis.

For simplicity and to ensure that the calculated dose is conservative for all radionuclides, all plant/meat/milk transfer factors were assigned the 75th percentile of the NUREG-6697 (RESRAD default) distribution. The RESRAD DCF file created to include the 75th percentile transfer factors is designated as “FCS FGR11”.

The Partial Rank Correlation Coefficient (PRCC) value reported in the RESRAD Regression and Correlation Report is used to evaluate parameter sensitivity of the Priority 1 and 2 physical parameters. The number of observations for the LHS analysis is at least 500 with three repetitions but may be increased to 1000 if results appear inconsistent. Parameters with PRCC values greater than or equal to the absolute value of 0.25 (|0.25|) are considered sensitive with respect to dose. This PRCC threshold is consistent with the methods used for parameter sensitivity analysis in NUREG/CR-6676, “Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Codes” (NUREG/CR-6676) [4]. If any one of the three repetitions is equal to or greater than |0.25| the parameter is designated as sensitive. Positively correlated parameters are assigned the 75th percentile of the PDF as a deterministic value. Negatively correlated parameters are assigned the 25th percentile of the PDF. Parameters that are not sensitive to dose are assigned the 50th percentile (median).

The PRCC cutoff value for categorizing parameters is not absolute. Each run is judged by the analyst. There are cases where the PRCC results are all below, but near, the |0.25| PRCC threshold, with consistent results for all three of the repetitions. In these cases, at the discretion of the analyst, a single parameter sensitivity analysis is used to augment the PRCC results. Using these additional methods, some parameters that do not exceed the |0.25| threshold are also designated as sensitive. Parameters that are designated as sensitive when the PRCC is below |0.25| are identified in Table 6-4, Table 6-5, and Table 6-6.

The uncertainty analysis is performed for each radionuclide individually. This conservatively disregards the reduced influence of low abundance radionuclides on the total dose and eliminates the potential impact of uncertainty in radionuclide mixture fractions. In addition, the uncertainty analysis is performed for a 1 m thickness of primary contamination. The results are also applicable to the 0.15 m contamination thickness.

The behavioral and metabolic parameters assigned are the D&D default deterministic values provided in NUREG/CR-5512, Volume 3, “Residual Radioactive Contamination from Decommissioning Parameter Analysis” (NUREG/CR-5512), Table 6.87 [11]. The D&D defaults are provided in the last column of Table 6.87 labeled “Solution”. Attachment 6.1 lists the deterministic values and PDFs selected for the soil uncertainty analysis and the reference or justification for the values. In some cases, more detailed explanation or calculations are necessary to document the parameter selections. This information is also provided in Attachment 6.1 (below the table). The deterministic values for the stochastic parameters (i.e., those with distributions listed in Attachment 6.1) are set to the 50th percentile of the distribution for the uncertainty analysis. The 25th, 50th, and 75th percentile values for the stochastic parameters are provided in Attachment 6.1, Section A.6.1.10.

6.9.5 Soil Uncertainty Analysis Results and Deterministic Parameter Selection

The results of the soil uncertainty analysis and the selected deterministic parameters are provided in Table 6-4, Table 6-5, and Table 6-6. When a parameter in Table 6-4 is identified as sensitive for one or more radionuclide, the selected deterministic value is assigned to all radionuclides, including those that are not sensitive to the given parameter. The majority of sensitive parameters have a consistent correlation with several radionuclides. The most obvious example is root depth which is negatively correlated due to the thickness of the contaminated zone. Another example is parameters related to inhalation that are consistently sensitive for certain actinides. The contaminated zone and unsaturated zone soil types, and corresponding K_ds, are the same. If either is sensitive, then both are changed for consistency. There were no conflicting results in contaminated zone and unsaturated K_d correlations, i.e., one positive and one negative.

Table 6-4 Soil RESRAD Parameter Uncertainty Analysis Results for Non-Nuclide Specific Parameters and Selected Deterministic Values

Parameter	Correlation to Dose	Radionuclide	Basis of Deterministic Parameter Selection	Selected Deterministic Value
Contaminated zone erosion rate (m/yr)	negative	Pu-241	25th percentile	7.59E-04
Contaminated zone b parameter (unitless)	negative	H-3	25th percentile	2.87
Evapotranspiration coefficient (unitless)	positive	C-14, Eu-152, Sb-125, Tc-99	75th percentile	0.87
Wind Speed (m/s)	negative	Am-241[1], C-14, Cm-243	25th percentile	3.27
Runoff coefficient (unitless)	positive	H-3, Tc-99	75th percentile	0.63
Depth of roots (m)	negative	Am-241, C-14[1], Ce-144, Cm-243, Co-58, Co-60, Cs-134, Cs-137, Eu-152, Eu-154, Eu-155, Fe-55, H-3, Ni-59, Ni-63, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Sb-125, Sr-90, Tc-99	25th percentile	1.23
Well pump intake depth (m)	negative	H-3, Np-237, Pu-238, Tc-99	25th percentile	21.4
b Parameter of Unsaturated zone (unitless)	NS[2]	NA	50th percentile	3.6
Mass loading for inhalation (g/m^3)	positive	Am-241, Pu-238, Pu-240[1], Cm-243	75th percentile	2.87E-05
Indoor dust filtration factor (unitless)	positive	Am-241, Pu-238, Cm-243	75th percentile	0.75
Depth of Soil Mixing Layer (m)	NS	NA	50th percentile	0.23
Wet foliar interception fraction of leafy vegetables (unitless)	NS	NA	50th percentile	0.58
Weathering removal constant all vegetation (unitless)	NS	NA	50th percentile	33
Wet weight crop yield of fruit, grain, and non-leafy vegetable (kg/m^2)	NS	NA	50th percentile	1.75
Humidity in air (H-3 only)	NS	NA	50th percentile	7.24

1) PRCC is less than |0.25| but correlation is indicated. Confirmed to have slight correlation to dose by single parameter sensitivity analysis.
2) not sensitive (NS)

Table 6-5 Soil RESRAD Parameter Uncertainty Analysis Results for K_d of Contaminated Zone and Unsaturated Zone and Selected Deterministic Values

Nuclide	Correlation to Dose	Basis of Deterministic Parameter Selection	Selected Deterministic Value (cm^3/g)
Am-241	NS[1]	50th percentile	1.25E+03
C-14	positive	75th percentile	96.7
Ce-144	NS	50th percentile	3.01E+03
Cm-243	NS	50th percentile	1.9E+04
Cm-244	NS	50th percentile	1.9E+04
Co-58	positive	75th percentile	5.05E+03
Co-60	positive	75th percentile	5.05E+03
Cs-134	NS	50th percentile	3.5E+03
Cs-137	NS	50th percentile	3.5E+03
Eu-152	positive	75th percentile	7.27E+03
Eu-154	positive	75th percentile	7.27E+03
Eu-155	positive	75th percentile	7.27E+03
Fe-55	NS	50th percentile	889
H-3	negative	25th percentile	4.3E-02
Ni-59	NS	50th percentile	179
Ni-63	positive	75th percentile	532
Np-237	negative	25th percentile	9.05
Pu-238	NS	50th percentile	953
Pu-239	NS	50th percentile	953
Pu-240	NS	50th percentile	953
Pu-241	NS	50th percentile	953
Sb-125	positive	75th percentile	128
Sr-90	positive	75th percentile	168
Tc-99	positive	75th percentile	1.47E-01

1) not sensitive (NS)

Table 6-6 Soil RESRAD Parameter Uncertainty Analysis Results for K_d of Saturated Zone and Selected Deterministic Values

Nuclide	Correlation to Dose	Basis of Deterministic Parameter Selection	Selected Deterministic Value (cm^3/g)
Am-241	NS[1]	50th percentile	1000
C-14	NS	50th percentile	11
Ce-144	NS	50th percentile	399
Cm-243	NS	50th percentile	3,390
Cm-244	NS	50th percentile	3.390
Co-58	NS	50th percentile	260
Co-60	NS	50th percentile	260
Cs-134	NS	50th percentile	528
Cs-137	NS	50th percentile	528
Eu-152	NS	50th percentile	829
Eu-154	NS	50th percentile	829
Eu-155	NS	50th percentile	829

Nuclide	Correlation to Dose	Basis of Deterministic Parameter Selection	Selected Deterministic Value (cm^3/g)
Fe-55	NS	50th percentile	321
H-3	NS	50th percentile	6.02E-02
Ni-59	NS	50th percentile	130
Ni-63	NS	50th percentile	130
Np-237	negative	25th percentile	5.49
Pu-238	NS	50th percentile	399
Pu-239	NS	50th percentile	399
Pu-240	NS	50th percentile	399
Pu-241	NS	50th percentile	399
Sb-125	NS	50th percentile	16.9
Sr-90	NS	50th percentile	22
Tc-99	NS	50th percentile	0.40

1) not sensitive (NS)

6.9.6 Soil Initial Suite DCGL

The deterministic parameter values in Table 6-4, Table 6-5, and Table 6-6 are used as the input values for all parameters designated as "stochastic" in Attachment 6-1. The deterministic parameter values in Attachment 6-1 are applied directly. RESRAD is then run with the deterministic parameters to calculate the soil DCGLs.

Table 6-7 provides DCGLs with no adjustment for the dose from insignificant contributor (IC) radionuclides. The calculation of the IC dose, and the selection of the radionuclides of concern (ROC) that remain after the elimination of the IC radionuclides, is discussed in Section 6.15. The final soil DCGLs for the ROC, after correction for IC dose, are provided in Section 6.16.

Table 6-7 Soil Initial Suite DCGLs (No IC Dose Adjustment)

Radionuclide	Soil DCGL 0.15 m (pCi/g)	Soil DCGL 1.0 m (pCi/g)
Am-241	1.402E+02	3.053E+01
C-14	5.996E+01	1.019E+01
Ce-144	2.746E+02	2.319E+02
Cm-243	6.747E+01	3.060E+01
Cm-244	2.944E+02	5.766E+01
Co-58	3.631E+01	3.128E+01
Co-60	3.970E+00	3.086E+00
Cs-134	6.424E+00	4.237E+00
Cs-137	1.374E+01	7.656E+00
Eu-152	8.857E+00	7.748E+00
Eu-154	8.220E+00	7.168E+00
Eu-155	3.081E+02	3.027E+02
Fe-55	3.660E+04	2.122E+04
H-3	1.195E+04	8.655E+02
Ni-59	1.128E+04	2.307E+03
Ni-63	4.120E+03	8.424E+02

Radionuclide	Soil DCGL 0.15 m (pCi/g)	Soil DCGL 1.0 m (pCi/g)
Np-237	4.723E+00	7.619E-01
Pu-238	1.752E+02	3.536E+01
Pu-239	1.578E+02	3.184E+01
Pu-240	1.578E+02	3.185E+01
Pu-241	5.666E+03	1.040E+03
Sb-125	2.662E+01	2.348E+01
Sr-90	1.111E+01	1.731E+00
Tc-99	1.356E+02	1.542E+01

6.10 Backfilled Basement Model Overview

The model for calculating dose and DCGLs from residual radioactivity in the backfilled basements is named the Basement Fill Model (BFM). The basements to remain are Containment Building, Auxiliary Building, Turbine Building, Intake Structure, and Circulating Water Tunnels. There are two media in backfilled basements: walls/floors and embedded pipe. A separate DCGL is calculated for each media. Detailed descriptions of the source term abstractions, conceptual models, and DCGL calculations for each media are provided in Sections 6.11, 6.11.11, and 6.13. A brief overview is provided here.

All basements include contaminated walls and floors (walls/floors) as a media or source term. The walls/floors are comprised of concrete and the remaining steel liner in Containment. Penetrations are included with the wall/floor media and have the same DCGLs. A single DCGL is calculated that applies to the Auxiliary Building, Turbine Building, Intake Structure, and Circulating Water Tunnel basements. A separate DCGL is required for Containment Building due to the presence of the liner.

The BFM wall/floor model includes three source release pathways:

- instant release from concrete (or from the surface of the Containment liner) to the water in the pore space of the fill material,
- capture of concrete (or steel liner in Containment) with drilling spoils generated during the installation of an onsite well, and
- excavation of concrete walls (or steel liner in Containment).

The doses from the three wall/floor release pathways are summed to calculate the final wall/floor DCGL. Details of the source release mechanisms and conceptual models for walls/floors are provided in Section 6.11.

The Auxiliary and Turbine Building basements also contain embedded pipe. The walls/floors and the embedded pipe have significant differences in source terms, physical configurations, conceptual models, and source term abstraction and release. Therefore, the DCGL calculation methods and results for walls/floors and embedded pipe are described separately in Section 6.11 and Section 6.13, respectively. The dose from walls/floors and embedded pipe are treated as separate media when calculating the final dose for FCS compliance with the 25 mrem/yr dose criterion (see Section 0).

6.11 BFM Wall/Floor Scenario DCGL

There are three scenarios that result in dose to the AMCG from residual radioactivity in the walls/floors: *insitu* scenario, excavation scenario, and drilling spoils scenario. The doses from the three scenarios are summed to determine the final wall/floor DCGL (see Section 6.11.11). This section describes the source term configuration, conceptual model, source term abstraction, and mathematical model for each the three BFM wall/floor scenarios. The dose assessment methods and the DCGL calculations for each scenario are described below.

6.11.1 BFM Wall/Floor Source Term Configuration and Spatial Variability

This section discusses the source term in basement walls/floors (including the steel liner to remain in Containment). The initial suite of radionuclides potentially present in the basements and their chemical form are discussed in Sections 6.5 and 6.6.

The major basement source term at license termination is expected to be in the Auxiliary Building basement concrete. Minimal contamination was introduced into the Turbine Building and Circulating Water Tunnels during operation or is expected to be introduced during decommissioning. The operations of the Intake Structure did not involve contact with plant-derived radionuclides. All concrete that is inside of the steel liner will be removed from the Containment Building leaving only the steel liner and the concrete outside of the liner. The vast majority of the contamination in the Containment Building will be removed with the concrete. The source term configuration and spatial variability in each basement is discussed below in a level of detail commensurate with the contamination potential at the time of license termination.

<u>Auxiliary Building Basement</u>

The Auxiliary Building contains all support systems for reactor operations that are not located in the Containment Building. In addition to reactor support systems typical of a pressurized water reactor (PWR) Auxiliary Building, the structure also contains the spent fuel pool, spent fuel support systems, emergency diesel generators, emergency core cooling and shutdown cooling systems. The Auxiliary Building is a large and complex structure with a wide range of radiological conditions within the structure. The contamination levels and spatial variability can be generally described by the Auxiliary Building gamma survey results from the HSA [2] which are provided in Table 6-8. The gamma results should be interpreted cautiously. Shine from contaminated equipment and structures present at the time of the survey, which will be removed during decommissioning, likely influence the readings.

Concrete wall/floor core samples were collected in the Auxiliary Building basement during characterization. The locations of characterization samples were biased to suspect areas. See Figures 2-10 and 2-11 in LTP Chapter 2 for sample location maps. Consideration was given to locations that exhibited measurable radioactivity (identified during the scan survey), depressions, discolored areas, cracks, low point gravity drain points, and actual or potential spill locations [9].

The core sample results are provided in the Characterization Report [9] and LTP Chapter 2. All Auxiliary Building basement interior walls will be removed leaving only the 971 and 989 foot elevation floors and outer walls at license termination. Therefore, only the samples from the 971 and 989 foot elevation floors and outer walls of the Auxiliary Building basement are germane to this discussion of source term present at the time of license termination. See Table 6-8.

Table 6-8 Auxiliary Building 971' and 989' Elevation Gamma Survey and Concrete Sampling Summary

Elevation	Sample No.	Room	Area or Component	General Area (mR/hr)	Max (mR/hr)	Cs-137 Concentration (0.0 – 0.5 in) (pCi/g)	Significant Cs-137 Contamination below 0.5 inches?
989'	1	24	SFP Transfer Canal Pumps	0.5 – 3	700/33[1]	2.23E+01	N
989'	2	24	SFP Transfer Canal Pumps	0.5 – 3	700/33	**4.40E+01**	N
989'	3	24	SFP Transfer Canal Pumps	0.5 – 3	700/33	2.35E+01	N
989'	4	24	SFP Transfer Canal Pumps	0.5 – 3	700/33	**2.68E+02**[2]	N
989'	5	6	Charging Pumps	0.5 – 2	2	5.11E+00	N
989'	6	6	Charging Pumps	0.5 – 2	2	**7.54E+02**	Y
989'	7	7	Charging Pump Valve Area	2 – 10	1240/80	9.30E+00	N
989'	8	7	Charging Pump Valve Area	2 – 10	1240/80	**1.33E+03**	Y
989'	9	7	Charging Pump Valve Area	2 – 10	1240/80	**5.09E+01**	N
989'	10	7	Charging Pump Valve Area	2 – 10	1240/80	**4.51E+02**	Y
989'	11	7	Charging Pump Valve Area	2 – 10	1240/80	**7.23E+01**	N
989'	19	5	Spent Fuel Pool Heat Exchanger	2 – 8	140/15	4.83E+00	N
989'	20	9b	Neutralization Tank	1.5 – 2	2	**1.25E+04**	Y
989'	21	9a	Waste Concentrator Pump	2 – 4	5	**4.73E+02**	N
989'	22	9	Waste Concentrator Tank	1.5 – 20	20	**1.81E+04**	Y
989'	23	9	Waste Concentrator Tank	1.5 – 20	20	**5.28E+01**	N
989'	24	12	Letdown Hx	1 – 18	70	1.33E+00	N
989'	25	12	Letdown Hx	1 – 18	70	**4.09E+01**	N
989'	26	10	Radwaste Monitor Tank	1 – 2	2	**2.18E+04**	Y
989'	27	10	Radwaste Monitor Tank	1 – 2	2	6.18E+00	N
989'	28	10	Radwaste Monitor Tank	1 – 2	2	2.81E+00	N
989'	29	10	Radwaste Monitor Tank	1 – 2	2	**3.00E+02**	Y
989'	30	11	Purification Filters	0.5 – 8	8	**3.10E+02**	N
989'	31	13	Blowdown Tank	0.5 – 4	300/23	**4.14E+02**	Y
989'	32	13	Blowdown Tank	0.5 – 4	300/23	3.36E+00	N

Elevation	Sample No.	Room	Area or Component	General Area (mR/hr)	Max (mR/hr)	Cs-137 Concentration (0.0 – 0.5 in) (pCi/g)	Significant Cs-137 Contamination below 0.5 inches?
989'	33	13	Blowdown Tank	0.5 – 4	300/23	2.63E-01	N
989'	34	14	Shutdown Cooling Hx A	0.5 – 3	4	8.21E-01	N
989'	35	14	Shutdown Cooling Hx A	0.5 – 3	4	**8.51E+02**	Y
989'	36	15	Shutdown Cooling Hx B	0.5 – 4	4	**1.08E+02**	N
989'	37	15a	Shutdown Cooling Hx Valves	3 – 18	18	4.29E+00	N
989'	38	15a	Shutdown Cooling Hx Valves	3 – 18	18	**1.45E+02**	N
989'	39	15a	Shutdown Cooling Hx Valves	3 – 18	18	**1.49E+02**	N
989'	40	20	Electrical Penetration Area	< 0.1	-	1.90E+01	N
971'	1	21	SI Pump Room West	1 – 3	1320/25	**5.30E+02**	N
971'	2	21	SI Pump Room West	1 – 3	1320/25	8.21E-01	N
971'	3	21	SI Pump Room West	1 – 3	1320/25	**3.20E+02**	Y
971'	4	21	SI Pump Room West	1 – 3	1320/25	4.18E+00	N
971'	5	21	SI Pump Room West	1 – 3	1320/25	**7.93E+02**	Y
971'	6	22	SI Pump Room East	1 – 4	685/12	**1.33E+02**	N
971'	7	22	SI Pump Room East	1 – 4	685/12	**5.81E+01**	N
971'	8	22	SI Pump Room East	1 – 4	685/12	1.80E-01	N
971'	9	22	SI Pump Room East	1 – 4	685/12	3.31E+01	N
971'	10	23	Spent Regenerant Tank and Pump Room	-	-	**8.26E+01**	N
971'	11	23	Spent Regenerant Tank and Pump Room	-	-	**5.73E+01**	N
971'	12	23	Spent Regenerant Tank and Pump Room	-	-	9.86E-01	N
971'	13	23	Spent Regenerant Tank and Pump Room	-	-	1.53E+00	N
971'	14	23	Spent Regenerant Tank and Pump Room	-	-	**1.30E+02**	Y
1) Contact/30 cm Dose Rates							
2) **Bold** values exceed the basement Operation DCGL (15% of 6.90E+06 pCi/m^2 IC adjusted BcDCGL from Table 6-1831) converted to 37 pCi/g Cs-137 in 0.5-inch concrete core sample. See LTP Chapter 5 for discussion of operational DCGLs.							

The results in the first 0.5-inch layer of concrete that exceed an operational DCGL (OpDCGL) criterion of 37 pCi/g Cs-137 are shown in bold in Table 6-8. See LTP Chapter 5 for the discussion and calculation of OpDCGLs. The OpDCGL is based on Cs-137 concentration only, assumes a wall/floor Base Case DCGL (BcDCGL) of 6.90E+06 pCi/m^2 for Cs-137 (see Table 6-31) which corresponds to 245 pCi/g uniformly distributed in a 0.5-inch layer of concrete. The OpDCGL *a priori* fraction is assumed to be 0.15 resulting in 37 pCi/g Cs-137. The last column of Table 6-8 identifies the locations where the core results for the 0.5 in -1.5 in depths indicate that concrete remediation deeper than 0.5 inch is, or may be, necessary.

From a review of Table 6-8, two observations are made regarding the Auxiliary basement source term configuration and spatial variability. The general area dose rates are much lower than the maximum dose rates identified in a given room. This indicates that the elevated contamination may be more localized than widespread. There are a number of rooms where significant contamination is found deeper than 0.5 inch in the concrete. However, sampling was limited to a depth of 1.5 inch at most locations. An investigation will be performed to bound the area and depth in the elevated areas.

The predominate radionuclide identified is Cs-137. Co-60 and Cs-134 are also positively identified but at much lower concentrations than Cs-137. The average Co-60 concentration in the 971 foot elevation was 0.3% of the Cs-137 concentration. For the 989 foot elevation, the average Co-60 concentration is 1.4% of the Cs-137 value. The Cs-134 fractions are lower than Co-60.

Eleven samples from the Auxiliary Building basement with elevated Cs-137 concentrations were sent to GEL laboratories for analysis of non-gamma emitting radionuclides (commonly referred to as hard to detect (HTD) radionuclides). Positively identified HTD radionuclides include H-3, C-14, Ni-59, Ni-63, Tc-99 and Sr-90. The HTD radionuclide with the highest abundance relative to Cs-137 is Ni-63.

Containment Building Basement

Significant concentrations of activated concrete were identified during the characterization of the under-vessel area (977 foot elevation). Lower concentrations were identified in the concrete at the 994 foot elevation with very limited indication of concrete activation [9]. See Chapter 2, Tables 2-16 to 2-20. However, all of the concrete in the Containment Building will be removed down to the steel liner. The source term at the time of FSS and license termination will likely be limited to surface contamination on the liner at low levels due to cross-contamination from dust generated during concrete removal. The spatial variability of the surface contamination is not known at this time but could be relatively widespread due to the scale of concrete removal activities. Nine concrete core samples were collected under the liner with no results above the minimum detectable activity (MDA).

Turbine Building Basement, Circulating Water Tunnels, and Intake Structure

From operational history, minimal contamination is expected in the Turbine Building basement, Circulating Water Tunnels, and Intake Structure. The liquid radwaste discharge point is within the Circulating Water Tunnels but the low concentrations in the liquid combined with high flow rate through the tunnels are expected to minimize the extent of concrete contamination. This will be verified during FSS planning. Eighteen concrete core samples were collected from the

Turbine Building basement during characterization and six cores were collected from the Intake Structure. No cores were collected from the Circulating Water Tunnels due to access issues. There were no plant-derived radionuclides detected in any of the concrete core samples [9]. Although there is always the potential for cross contamination during demolition and decommissioning activities, minimal residual radioactivity is expected to be present at the time of license termination.

6.11.2 BFM Wall/Floor Conceptual Model, Source Term Abstraction and Release

The basements of the Turbine Building, Containment Building, Auxiliary Building, Circulating Water Tunnels, and Intake Structure will remain with all interior walls removed. One exception is the Turbine Building where the pedestals will remain within the interior of the basement. The basements will be demolished to at least 3 feet (0.92 m) below grade which corresponds to an elevation of 1001 foot AMSL.

After remediation and FSS are completed, contingent upon the completion of confirmatory surveys and regulatory approval, the Turbine Building, Containment Building, and Auxiliary Building basements will be backfilled to the original grade of 1004 feet AMSL. The Circulating Water Tunnels and Intake Structure will be backfilled with flowable fill (grout). All basement walls/floors will therefore have a minimum clean cover thickness of 0.92 m.

The BFM wall/floor source term includes surface and volumetric contamination on and within the basement concrete and surface contamination on the remaining steel liner in Containment. The concrete and remaining sleeves in the penetrations are assumed to be a part of the wall/floor source term and have the same DCGLs as walls/floors. There are three conceptual model scenarios that have different source term release mechanisms. The doses from each of the three scenarios are summed to determine the BFM wall/floor DCGL. The scenarios and release mechanisms are:

- *insitu* scenario: release from concrete (or from the surface of the Containment liner) to the water in the pore space of the fill material,
- drilling spoils scenario: incorporation of concrete into drilling spoils generated during the installation of an onsite well, and
- excavation scenario: excavation of basement concrete walls and steel liner of Containment.

The environmental pathways and exposure pathways that apply to each of the three scenarios are listed in Table 6-2, source numbers 13, 14, and 15.

The *insitu* scenario source term release pathway from concrete and the steel liner leads to the contamination of fill material and water in the pore space of the fill. As discussed in 6.11.1, the majority of residual radioactivity is expected to be in the concrete as opposed to the liner. The ratio of the concentration in fill to that in the water is a function of the radionuclide specific K_ds.

The source term release pathway for the drilling spoils scenario is the removal of contaminated concrete during the installation of a water well. The contaminated concrete is brought to the ground surface with the overlying fill material and uniformly mixed within the drilling spoils.

The third release pathway is the excavation of basement concrete walls and the steel liner from

Containment. The contaminated and uncontaminated portions of concrete are inadvertently mixed during the process of excavation, rebar removal, sizing, and spreading the concrete debris on the ground with a thickness of 1 m. The excavated concrete is treated as soil in the dose calculation. The activity on the steel liner is released to the surface soil after excavation. The excavation of the liner would be a major construction project with significant disturbance of the surrounding soil. The released activity could be mixed over a range of depths in soil. Therefore, both a 0.15 m and 1.0 m mixing depth are evaluated and the maximum dose is used for the DCGL calculation.

The BFM conceptual model and source term configuration includes several abstractions and simplifications. The source is assumed to be a volumetric or surface layer at uniform concentrations (pCi/m^2) in or on the walls and floors of all basements. The assumption of uniform distribution over all surfaces overestimates the source term because, as discussed in Section 6.11.1, all basements except the Auxiliary Building basement are expected to contain minimal residual radioactivity at license termination. The thickness of the source in concrete is immaterial because all contamination at all depths is assumed to instantly release to the adjacent fill immediately after license termination. Instant release is a conservative assumption because the residual radioactivity would actually release more slowly by diffusion from concrete allowing for radioactive decay as well as continual source depletion as the water containing radioactivity is removed by the well. For the drilling spoils and excavation scenarios, the contamination is assumed to remain in the concrete which maximizes the source term.

The water supply well in the BFM *insitu* scenario is assumed to be located 1 m from the wall. To reach the well, the residual radioactivity in the concrete is assumed to instantly release and uniformly mix in the fill between the wall and the well. A 1 m mix distance corresponds to a surface area to volume ratio (SA/V) of 1, i.e., the residual radioactivity contained in 1 m^2 of wall surface is mixed into 1 m^3 of fill. The concentration in the fill is an inverse linear function of the distance from the wall to the well and the corresponding mixing distance. A reasonable alternate assumption would be to locate the well in the center of a given basement. From "OPPD Calculations and Position Paper FC-20-006, Fort Calhoun Station End State Building Concrete Surface Area and Volumes," 2021 [12], the total surface SA/V ratios for the Auxiliary Building basement, Containment Building basement, Turbine Building basement, Intake Structure, and Circulating Water Tunnels are 0.37, 0.65, 0.50, 0.41, and 1.05, respectively. Under the full mix assumption, the concentration in the fill would decrease as a function of the total SA/V. The 1 m well distance and corresponding SA/V of 1 is conservative compared to the full mix assumption for all basements except the Circulating Water Tunnels which has a SA/V of 1.05. The 5% exceedance for the Circulating Water Tunnels is considered trivial given the conservatism in the BFM and the expectation of minimal contamination in the Circulating Water Tunnels. Therefore the 5% full mix exceedance for the Circulating Water Tunnels is ignored.

As discussed above, a SA/V ratio of 1 is used to calculate the concentration in fill after release from walls/floors. However, there are penetrations through the walls of the Auxiliary Building, Containment Building and Turbine Building basements and the Intake Structure with SA/Vs of 7.9, 7.0 and 9.9, and 2.3, respectively [4] [13]. The impact of the penetrations on the total wall/floor source term is limited because the total internal surface areas of the penetrations are small relative to the surface area of the wall/floor. The penetration surface areas are 0.1%, 2%, 0.3% and 3% of the wall/floor surface areas for the Auxiliary Building, Containment Building, Turbine Building basements, and Intake Structure, respectively [4] [12] [13]. The additional source term contribution from penetrations is considered insignificant, particularly when compared to the approximately 30% overestimate of groundwater concentration by RESRAD as compared to the more realistic MT3D modeling (see Section 6.11.3). The wall/floor BFM *insitu* DCGLs are therefore applied directly to penetrations in order to simplify the conceptual model.

The justification for selecting a well location that is 1 m from the wall is based on two considerations. First, Section 6.11.11 provides a check calculation to confirm that the concentrations in the fill material would be less than the soil DCGLs when the concentration on the walls is equal to the BFM wall/floor DCGLs. The check calculation assumes that the fill within a 1 m distance from the wall is excavated based on the use of standard excavation equipment and practices. The activity from the wall is assumed to instantly release and mix within the adjacent 1 m^3 of fill which corresponds to a SA/V of 1. For consistency, the well location and corresponding fill mixing distance for the BFM *insitu* scenario is also set to 1 m. Second, the probability of installing a well at a distance 1 m from a wall, or closer, is low. The resident farmer compliance scenario assumes that a well is installed on the site. The well can be installed anywhere on the 79,600 m^2 site area assumed in the conceptual model (the actual licensed area is larger). Using Equation 6-1, the probability of installing the well within 1 m of a basement wall is 2.3%. This corresponds to a 97.7% confidence level that the well would be installed at a distance greater than 1 m from the walls.

Equation 6-1

$$p_{1m} = \frac{L_w d_m}{A_{cs}}$$

where:

p_{1m} = probability that a well would be located 1 m from a basement wall
L_w = total length of all basement walls (m) = 1800 m [12]
D_w = distance from wall to well (mix distance) = 1 m
A_{cs} = total area of conceptual site = 79,600 m^2

It is important to note that the dose contribution from the *insitu* scenario is less than the 25 mrem/yr dose attributable to the BFM wall/floor DCGLs which are based on the summation of dose from the excavation scenario, drilling spoils scenario, and *insitu* scenario (see Section 6.11.11). Using Cs-137 as an example, the percentage of the BFM wall/floor dose attributable to the *insitu* scenario is 29% of the 25 mrem/yr total (7.2 mrem/yr) (see Section 6.11.11). Further, the *a priori* dose fraction assigned for the calculation of the operational DCGL (OpDCGL) for the backfilled basement walls/floors is 0.3 (LTP Chapter 5). The Cs-137 dose attributable to the *insitu* scenario would therefore be no more than 2.2 mrem/yr (0.3 x 7.2 mrem/yr). The 2.3% probability that the well could be closer than 1 m, and that the *insitu* scenario dose is higher than that calculated assuming 1 m, relates to a nominal Cs-137 dose of 2.2 mrem/yr, not 25 mrem/yr, which further reduces the potential dose impact of uncertainty in the conceptual model.

The structures are assumed to provide no obstruction to groundwater flow, i.e., are not present. The characteristics of the site-wide hydrogeological conceptual model described in Section 6.6 apply to the backfilled basements given the assumption of walls not being present. The fill is assumed to have the characteristics of the soil in the saturated zone, i.e., sand. The actual fill material slated to be used is a silt loam soil type [1]. The K_ds of silt loam are higher than sand K_ds [10] and would therefore result in lower well water concentrations in the BFM *insitu* scenario and correspondingly lower dose. But the characteristics of the fill is not the only consideration. Under the assumption that the walls are not present and that the immediately adjacent surrounding soil is saturated zone sand, the water flowing from the fill to the surrounding sand would equilibrate in accordance with the sand K_d. If the well is located outside of the basement at the downstream edge, the well water concentrations may not differ greatly between sand or silt loam fill material.

The location of the well, and the corresponding mixing distance in fill, and the assumption of flow or no flow groundwater conditions in the basements are important considerations in the BFM *insitu* conceptual model. The source term represented by the volume of contaminated fill in all basements, after release and uniform mixing (total wall/floor surface area multiplied by the 1 m mix distance), as well as the assumption of groundwater flow conditions are used in RESRAD to calculate dose to the AMCG. The justification for the selection and use of these inputs in the RESRAD nondispersion groundwater model is provided in Section 6.11.3.

The vadose zone thickness of 1.1 m in the site-wide conceptual model (Section 6.6) is also considered for the BFM *insitu* scenario. Assuming a 1.1 m vadose zone thickness leads to the conclusion that when the basements are in the as-left geometry at the time of license termination (*insitu* geometry) all of the walls/floors, other than the top 0.18 m of the walls, are within the saturated zone. As conservative and simplifying assumption, the 0.18 m portion of wall is also assumed to be in the saturated zone. This leads to the vadose zone being a 0.92 m clean cover. There is no unsaturated zone present in the *insitu* scenario conceptual model. The excavation and drilling spoils scenarios are not affected by vadose zone thickness.

The conceptual model for the BFM excavation scenario includes the excavation of at least the top 2 m of concrete walls (or steel liner in Containment) which is approximately 3 m bgs. The minimum wall thickness in any basement (2 ft) is used and the concrete is assumed to be spread over a 1 m depth on the ground surface. A typical excavation process for a backfilled structure would entail using a medium sized excavator with a 1.0 to 1.5 cubic yard bucket to excavate and stockpile fill. After removing the fill to the planned excavation depth, a hoe-ram would be used to pound out the concrete walls. The concrete would be segregated, the rebar removed, and remaining concrete size reduced. The BFM excavation scenario assumes that the size reduced concrete is used as onsite fill.

The BFM excavation scenario applies to concrete. A check calculation is performed assuming that the activity is released to the fill as opposed to remaining in the concrete (see Section 6.11.11). The concentrations in the excavated fill are shown to be less than the concentrations in the excavated concrete.

6.11.3 BFM *insitu* Scenario Mathematical Model

RESRAD is used to calculate DSRs (mrem/yr per pCi/g) for the BFM *insitu* scenario. There are three conceptual model assumptions that require additional discussion to demonstrate that the RESRAD nondispersion model is applicable and conservative for the *insitu* scenario: 1) source term geometry after release and mixing in fill, 2) assuming all walls/floors are in the saturated zone, and 3) assuming groundwater flow through the basements. After confirming that these three assumptions can be conservatively applied in RESRAD, the converted source term (after release from concrete and mixing in fill) is used in a standard manner in RESRAD to calculate dose from all exposure pathways in the resident farmer scenario.

As discussed above, the well location and mixing distance are correlated and create the source term geometry. Instant release and mixing are conservatively assumed. After release from the wall by diffusion, the contamination moves to the well in accordance with the natural gradient as augmented by the gradient induced by the well pumping. Any mixing time other than instant release will lower the concentration in the well.

Source Term Geometry

The source term geometry in fill after release from walls is a function of the well location and corresponding mixing distance in fill. Assuming the well is located 1 m from the wall, the source term geometry is a 1 m thick layer of contaminated fill adjacent to all walls. The mixing distance from the floors is also assumed to be 1 m for consistency but, as discussed below, the well water concentration from floor contamination is not sensitive to the mixing distance. Figure 6-18 provides a simplified plan view of the wall source term geometry for the Auxiliary Building basement.

The RESRAD nondispersion groundwater model is used to calculate well water concentrations for the BFM *insitu* scenario. However, the geometry in Figure 6-18 cannot be modeled directly in RESRAD which requires a single contiguous source with a given area and depth. As a simplifying assumption, a source term is constructed for use in RESRAD which has a volume equal to the total surface area of all walls and floors in all basements multiplied by the assumed 1 m mix distance in fill. This represents the conceptual volume of contaminated fill after release from all basement surfaces and mixing into a 1 m fill thickness (i.e., SA/V of 1).

The total basement surface area is 16,700 m^2 which corresponds to a source volume of 16,700 m^3 [12]. A nominal source term thickness of 4 m is assumed which is the approximate height of basement walls at the 989 foot elevation Auxiliary Building floor and 990 foot elevation of the Turbine Building floor. The conceptual source is assumed to be 60 m wide (the approximate width of the Auxiliary and Turbine Building basements) and the length parallel to aquifer flow is assumed to be 70 m (16,700 m^3/ (60 m x 4 m)). The well intake depth is set to the same value as the source thickness, i.e., 4 m.

The primary performance objective of the conceptual model source term configuration is to ensure that the well water concentration calculated by the RESRAD nondispersion model is at the theoretical maximum. This is accomplished with the 16,700 m^2 area and 4 m depth as discussed below and demonstrated using Equation 6-2. But it should be noted that this could also be accomplished with essentially any conceptual source term configuration if the well depth is equal to or less than the conceptual source term depth.

As discussed in Section 6.11.2, the entire source term is assumed to be in the saturated zone. RESRAD ONSITE Version 7.2 allows the source to be entirely submerged in the saturated zone but the use of RESRAD ONSITE Version 7.2 under saturated conditions requires justification (USNRC, "E-Mail from Jack Parrot, NRC, to Corey Cameron, OPPD," January 25, 2021) [14]. The performance of the RESRAD nondispersion model with a fully submerged source term was tested. The code was run with Cs-137 and Sr-90 using a 16,700 m^2, 4 m deep, fully submerged source term with a length parallel to flow of 70 m. The well intake depth is set to the same value as the source thickness, i.e., 4 m. The remaining RESRAD parameters were set to the soil DCGL values with modifications as described in Section 6.11.5. The well water concentrations calculated by RESRAD were compared to the theoretical maximum concentrations in the fill pore space water, after instant release and mixing, using Equation 6-2. Achieving a well water concentration that is equal to the theoretical maximum in the fill pore space water is considered a conservative and acceptable result.

To determine the theoretical maximum water concentrations, Equation 6-2 was used with a unit concentration of 1 pCi/g in the fill, and K_ds of 158 and 6.6, for Cs-137 and Sr-90, respectively (25^{th} percentiles of saturated zone sand K_d PDFs listed in Attachment 6-1). A unit source volume (fill) of 1 m^3 was assumed for the calculation along with the saturated zone parameters listed in Attachment 6-1.

Equation 6-2

$$C = \frac{I}{[V(\theta + \rho Kd)]}$$

where:

C= concentration in water (pCi/L)
I = inventory (pCi) in fill = $1.49x10^6$ pCi
V = mixing volume (L) = 1000 L
Θ = total porosity of the saturated zone = 0.45
ρ = bulk density (g/cm^3) = 1.49 g/cm^3
K_d = distribution coefficient (cm^3/g)

and,

$$I = V_f 1x10^6 \rho_f C_f$$

where:

$V_{f\,=}$ volume of fill = 1 m^3
$1x10^6$ = conversion factor cm^3/m^3
ρ_f = bulk density of fill = 1.49 g/m^3
C_f = unit concentration in fill = 1 pCi/g

The theoretical maximum pore space water concentrations calculated by Equation 6-2 are 6.32 pCi/L and 144.9 pCi/L for Cs-137 and Sr-90, respectively. The well water concentrations calculated by RESRAD at time = 0 are the same at 6.23 pCi/l and 144.9 pCi/L, respectively. The performance of the RESRAD nondispersion model, when the source term is fully submerged, is therefore considered conservative and acceptable. In addition, a sensitivity analysis was performed to confirm that the RESRAD calculated well water concentrations, when the source term is fully submerged, are not sensitive to well pumping rate. The concentrations remained at the theoretical maximum values over a wide range of well pumping rates.

A second assessment evaluated whether the maximum water concentration in fill, as calculated by RESRAD, assuming instant release and a 1 m mix distance (SA/V=1), is conservative when compared to the well water concentrations calculated using the actual source term geometry as shown in Figure 6-18. Groundwater Vistas is a more complex groundwater modeling package that is used to directly evaluate the actual source term geometry in Figure 6-18. Well water concentrations calculated by RESRAD, using the conceptual model source term described above, and by Groundwater Vistas using the actual source term geometry are compared. The 3-D groundwater flow modeling was performed using MODFLOW-2005, particle tracing was calculated using MODPATH version 7, and radionuclide transport was completed using MT3DMS [1]. A detailed description of the methods and results is provided in [1]. A summary is provided here.

The MT3DMS model assumes a unit water concentration of Cs-137 and Sr-90 in fill in the geometry shown in Figure 6-18 and under fully saturated conditions. The source term is a 1 m thick contaminated zone on the walls and floors. The unit concentration represents the theoretical maximum well water concentration after instant release and mixing over a 1 m distance as discussed above. The hydrogeological parameters are the same as those used in RESRAD.

The well is placed in a variety of locations under flow conditions (assuming walls/floors do not obstruct flow) and no flow conditions (bathtub model). Several well locations are evaluated including 1 m from the center of a wall and 1 m from a wall corner both upstream and downstream of groundwater flow direction (immediately adjacent to the 1 m thick source term). The corner location is selected to ensure that when the well is in proximity to two walls, that the well water concentration is not higher than that calculated by RESRAD using a fill concentration based on a SA/V ratio of one (1 m well distance from a single wall). A well location at the center of the basement is also tested. A 1 m thick layer of contamination on the floor is directly accounted for in the model. Various source thicknesses and well depths are also tested.

The basement is assumed to be a nominal 60 m x 60 m square which are roughly the dimensions of the Auxiliary Building basement (including Containment). Well depths of 4 m and 9 m are tested which are the depths of the 971 and 989 foot elevations in the Auxiliary Building basement (top of wall elevation is at 1001 foot AMSL). Assuming well depths of 4 m and 9 m is unrealistic and conservative. The average depth of actual domestic and irrigation wells in the vicinity of FCS is 34 m [1]. An MT3D test of the realistic 34 m well depth is also performed to determine the extent of conservatism.

The no flow condition is tested because it is unlikely that groundwater will flow unimpeded through the basement walls although some in-leakage is expected to occur over time. MT3D is therefore used to test the various well locations in the no-flow condition, i.e., “bathtub” model, to ensure that the groundwater concentrations would not exceed the concentrations calculated by RESRAD which requires flow.

The results of the MT3DMS tests are provided in Table 6-9. The results represent the ratio of the MT3D model well water concentrations to the RESRAD nondispersion model concentrations. Particle transport diagrams and concentration curves as a function of time for the upstream corner, downstream corner, and downstream center locations under flow conditions, and upstream and downstream corners under no-flow (bathtub) conditions, are provided in Attachment 6.2. The shapes of the no-flow particle transport diagrams (“crowning affect”) are an artifact of setting the zero flow boundary conditions in the model and do not affect the results [1]. The particle flow diagrams and concentration curves for all other well locations and conditions listed in Table 6-9 are provided in [1].

Table 6-9 MT3D Results Under Groundwater Flow and No Flow Conditions in Basements

Well Location[1]	Ratio MT3D/RESRAD 1.0 m source thickness groundwater flow		Ratio MT3D/RESRAD 1.0 m source thickness, no flow (bathtub)	
	Cs-137	Sr-90	Cs-137	Sr-90
Upstream Center	0.45	0.45	-	-
Upstream Corner	0.66	0.65	0.61	0.62
Downstream Center (Inside)	0.40	0.40	-	-
Downstream Corner	0.66	0.66	0.61	0.61
Downstream Center (Outside)	0.37	0.37	-	-
Center of Basement	0.11	0.11	0.12	0.12
4m Wall, Downstream Corner	0.70	0.70	0.65	0.66
35 m well depth Downstream Corner	0.29	0.30	0.31	0.32
30 x 30, Downstream Corner	0.61	0.62	0.55	0.56
Downstream Corner 400 m^3/yr well pump rate	0.60	0.63	NA	NA
Downstream Corner 10,000 m^3/yr well pump rate	0.66	0.66	NA	NA

1) Wall height and corresponding well depth is 9 m unless otherwise note

In all cases the groundwater concentrations using the actual source term geometry and the more sophisticated MT3DMS transport model are less than the groundwater concentrations calculated by RESRAD using a simplified conceptual model. The maximum ratio for both Cs-137 and Sr-90 is 0.66 in the downstream corner under flow conditions when the well depth is set to 9 m. Placing the well in the center of the basement reduces the ratio to 0.11 and 0.12, respectively. The reduced concentrations with MT3DMS are expected because the well is drawing from a 360° radius as opposed to 100% from the fill immediately adjacent to the well as is assumed in RESRAD. The mixing distance from angles other than zero degrees is effectively longer than 1 m resulting in additional mixing and reduction in groundwater concentration. In addition, the well also draws from areas of clean fill that further dilutes the concentrations in the well.

The ratio is 0.70 in the downstream corner when a well depth of 4 m is assumed. The concentration ratio of Cs-137 when the well is located in the downstream corner and the well depth is equal to the local mean value of 34 m is reduced to 0.29 under flow conditions and 0.31 with no flow. The ratios for Sr-90 under these conditions are 0.30 and 0.32, respectively.

A sensitivity analysis was also performed using a smaller area of 30 m x 30 m and the downstream corner location. The smaller area is intended to generally represent the Auxiliary Building subbasement at 971 foot elevation or the Auxiliary Building basement area at 989 foot elevation that is adjacent to Containment. The ratios for the smaller area are lower than those calculated for the 60 x 60 m area. A sensitivity analysis was also performed using MT3D for well pumping rates of 400 m^3/yr and 10,000 m^3/yr. These pump rates correspond to a water well for a resident that does not irrigate and a resident farmer that irrigates a large 40,000 m^2 land area, respectively. As seen in Table 6-9, there is essentially no sensitivity to pumping rate.

One last observation from the MT3DMS modeling relates to well water concentration when the source is on the floor only. From Attachment 6-2, it is seen that the water concentration is a simple ratio of the mixing distance (1 m) and the well pump intake depth. This relationship is also seen in the RESRAD well water concentrations when the source term thickness is varied while maintaining a constant well pump intake depth.

In conclusion, MT3DMS groundwater modeling of a more realistic basement source term geometry confirms that the well water concentrations calculated using RESRAD with a source term comprised of a contiguous volume that is 16,700 m^2 in area and 4 m deep, and is fully submerged in the saturated zone with a 4 m well pump intake depth, will provide conservative well water concentrations. Coupling the MT3DMS results with the assumptions that the well is located at a distance of 1 m from the wall (which has a 2.3% probability of occurrence as discussed above), that the activity is instantly released, and that the well is installed immediately after license termination, provides high confidence that the BFM *insitu* scenario conceptual model will provide well water concentrations that are less than would actually be encountered by future site occupants.

6.11.4 BFM *insitu* Scenario RESRAD Uncertainty Analysis

The process used for the uncertainty analysis is the same as described in 6.9.4 for soil with the exception that the number of observations for certain actinides were reduced to 300 due to long run times. The parameters used for the BFM *insitu* scenario uncertainty analyses are the same as those used for soil (see Attachment 6.1) with the changes listed in Table 6-10. The well pump intake depth is removed from the analysis because the *insitu* scenario conceptual model requires the well depth to be the same as the thickness of the contaminated zone, i.e., 4 m. A PDF for the cover erosion rate is added due to the addition of a cover. The contaminated zone erosion rate is set to 0 because all of the contamination is assumed to be submerged.

The contaminated zone parameters in Table 6-10 are changed to match the saturated zone as opposed to the unsaturated zone because the contaminated zone is fully submerged. There is no unsaturated zone present. The K_d PDFs in Table 6-12, for both the contaminated zone and saturated zone, are assumed to be the sand soil type found in the saturated zone. In the BFM, the contaminated zone represents the fill material. However, the intake structure and circulating water tunnel basements are to be filled with grout (see LTP Chapter 3). As seen in Table 6-12, the K_ds for the majority of radionuclides are negatively correlated. None are positively correlated. The K_d values for Cesium, Cobalt, Strontium, and Nickel for low density grout were determined by laboratory analysis at Brookhaven National laboratory as reported in [15]. In all cases, the grout K_ds were higher than the K_d values selected for the BFM *insitu* scenario in Table 6-12. Using grout as fill would therefore result in lower well water concentrations, and lower dose, than the dose calculated for the BFM *insitu* scenario using the K_d values in Table 6-12.

The deterministic parameters in Table 6-10 are used for the insitu scenario DCGL calculations in place of the corresponding values in Attachment 6.1. The deterministic parameters selected through uncertainty analysis, and listed in Table 6-11 and Table 6-12, are also applied. The deterministic values in Attachment 6.1 are used for all parameters not listed in Table 6-10, Table 6-11, and Table 6-12.

Table 6-10 Changes to the Soil DCGL Deterministic Parameters and PDFs for BFM Wall/Floor *insitu* Scenario Uncertainty Analysis

Parameter	BFM *insitu* Scenario
Cover depth (m)	0.92
Cover erosion rate (m/y)	RESRAD Default PDF
Contaminated zone erosion rate[1]	0
Area of contaminated Zone (m^2)	16,700
Thickness of contaminated zone[1] (m)	4
Length parallel to aquifer flow (m)	70
Contaminated fraction below the water table	1
Contaminated zone density (g/cm^3)	1.49
Contaminated zone total porosity (unitless)	0.45
Contaminated zone field capacity (unitless)	0.24
Contaminated zone hydraulic conductivity (m/yr)	4350
Well pump intake depth (m)	4
Well pump intake depth (m)	Remove PDF from uncertainty Analysis
Contaminated zone deterministic K_d values for daughter products changed to the 50^{th} percentile of PDF in Table 2.13.1 (sand) or 2.13.10 (generic) [10]	Radionuclide Dependent
Contaminated zone K_d PDFs changed to the distributions in Table 2.13.1 (sand) or 2.13.10 (generic) [10]	Radionuclide Dependent

1) contaminated zone is fully submerged

6.11.5 BFM *insitu* Scenario Uncertainty Analysis Results and Deterministic Parameter Selection

The results of the uncertainty analysis and the selected deterministic parameters are provided in Table 6-11 and Table 6-12.

Table 6-11 BFM Wall/Floor *insitu* Scenario RESRAD Parameter Uncertainty Analysis Results for Non-Nuclide Specific Parameters and Selected Deterministic Values

Parameter	Correlation to Dose	Radionuclide	Basis of Deterministic Parameter Selection	Selected Deterministic Value
Cover erosion rate (m/y)	positive	Am-241, C-14, Eu-152, Ni-59, Pu-239, Pu-240	75th percentile	2.92E-03
Contaminated zone b parameter (unitless)	NS[1]	NA	50th percentile	3.6
Evapotranspiration coefficient (unitless)	positive	H-3, Tc-99	75th percentile	0.87
Wind Speed (m/s)	NS	NA	50th percentile	3.75
Runoff coefficient (unitless)	positive	H-3, Tc-99[2]	75th percentile	0.63
Depth of roots (m)	positive	Am-241, C-14, Ce-144, Cm-243, Cm-244, Cs-134, Cs-137, Co-58, Co-60, Eu-152, Eu-154, Eu-155, Fe-55, H-3, Ni-59, Ni-63, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Sb-125, Sr-90, Tc-99	75th percentile	3.08
Mass loading for inhalation (g/m^3)	NS	NA	50th percentile	2.35E-05
Indoor dust filtration factor (unitless)	NS	NA	50th percentile	0.55
Depth of Soil Mixing Layer (m)	NS	NA	50th percentile	0.23
Wet foliar interception fraction of leafy vegetables (unitless)	positive	Ce-144, Fe-55, Np-237, Pu-241[2], Pu-239[2], Pu-240	75th percentile	0.70
Weathering removal constant all vegetation (unitless)	negative	Ce-144, Fe-55, Ni-59[2], Ni-63[2], Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Tc-99	25th percentile	21.5
Wet weight crop yield of fruit, grain, and non-leafy vegetable (kg/m^2)	negative	Ce-144, Fe-55, Np-237[2], Pu-241[2], Pu-239[2], Pu-240	25th percentile	1.27
Humidity in air (H-3 only)	NS	NA	50th percentile	7.24

1) not sensitive (NS)
2) PRCC is less than |0.25| but correlation is indicated. Confirmed to have slight correlation to dose by single parameter sensitivity analysis.

Table 6-12 BFM Wall/Floor *insitu* Scenario RESRAD Parameter Uncertainty Analysis Results for K_d of Contaminated Zone and Unsaturated Zone and Selected Deterministic Values

Nuclide	Correlation to Dose	Basis of Deterministic Parameter Selection	Selected Deterministic Value (cm^3/g)
Am-241	negative	25th percentile	2.69E+02
C-14	negative	25th percentile	1.26E+00
Ce-144	NS	50th percentile	3.99E+02
Cm-243	negative	25th percentile	5.72E+02
Cm-244	negative	25th percentile	5.72E+02
Co-58	negative	25th percentile	3.70E+01
Co-60	negative	25th percentile	3.70E+01
Cs-134	negative	25th percentile	1.58E+02
Cs-137	negative	25th percentile	1.58E+02
Eu-152	negative	25th percentile	9.45E+01
Eu-154	negative	25th percentile	9.45E+01
Eu-155	negative	25th percentile	9.45E+01
Fe-55	NS	50th percentile	3.21E+02
H-3	negative	25th percentile	4.30E-02
Ni-59	negative	25th percentile	2.76E+01
Ni-63	negative	25th percentile	2.76E+01
Np-237	negative	25th percentile	5.49E+00
Pu-238	negative	25th percentile	1.56E+02
Pu-239	negative	25th percentile	1.56E+02
Pu-240	negative	25th percentile	1.56E+02
Pu-241	negative	25th percentile	1.56E+02
Sb-125	negative	25th percentile	5.07E+00
Sr-90	negative	25th percentile	6.57E+00
Tc-99	negative	25th percentile	1.90E-02

1) not sensitive (NS)

6.11.6 BFM *insitu* Scenario Initial Suite DCGL

The BFM *insitu* scenario DCGL ($DCGL_i$) is calculated in two steps. First, the unit concentration of activity in the fill (pCi/g per pCi/m^2) is calculated in Excel using Equation 6-3 [13]. As discussed in Section 6.11.2, the well location (mixing distance in fill) is assumed to be 1 m from the walls and floors. The activity in a 1 m^2 area of the wall instantly releases and mixes with a 1 m^3 volume of fill. The $DCGL_i$ are calculated using the unit concentration from Equation 6-3 and the DSRs calculated with the parameters developed in Section 6.11.5. The $DCGL_i$ values are calculated in Excel using Equation 6-4 [13]. The DSRs and $DCGL_i$ values are listed in Table 6-13. The values in Table 6-13 are not corrected for IC dose.

Equation 6-3

$$C_{f,u} = \frac{A_{c,u}}{V_{f,u} 1x10^6 \rho_f}$$

where:

$C_{f,u}$ = unit concentration in fill (pCi/g per pCi/m^2)
$A_{c,u}$ = unit activity of 1 pCi over a 1 m^2 area of concrete (pCi)
$V_{f,u}$ = unit fill volume of 1 m^3
Conversion Factor = 1x10^6 cm^3 per m^3
ρ_f = bulk density of fill (assumed to be sand)

Equation 6-4

$$DCGL_{i,j} = \frac{25}{DSR_{i,j} C_{f,u}}$$

where:

$DCGL_{i,j}$ = *insitu* scenario DCGL for radionuclide j (pCi/m^2)
$C_{f,u}$ = unit concentration in fill (pCi/g per pCi/m^2) from Equation 6-3.
$DSR_{i,j}$ = BFM *insitu* dose to source ratio for radionuclide j (mrem/yr per pCi/g)
25 = 25 mrem/yr dose criterion

Table 6-13 BFM *insitu* Scenario Initial Suite DSRs and DCGLs (No IC Dose Correction)

Radionuclide	DSR (mrem/yr per pCi/g)	$DCGL_i$ (pCi/m^2)
Am-241	8.025E+00	4.642E+06
C-14	1.591E+00	2.341E+07
Ce-144	2.441E-02	1.526E+09
Cm-243	2.800E+00	1.330E+07
Cm-244	2.238E+00	1.664E+07
Co-58	5.019E-02	7.422E+08
Co-60	1.299E+00	2.868E+07
Cs-134	1.854E+00	2.009E+07
Cs-137	1.472E+00	2.531E+07
Eu-152	4.061E-02	9.173E+08
Eu-154	5.906E-02	6.307E+08
Eu-155	9.177E-03	4.059E+09
Fe-55	1.382E-03	2.695E+10
H-3	1.366E-01	2.727E+08
Ni-59	1.852E-02	2.011E+09
Ni-63	5.072E-02	7.344E+08
Np-237	4.406E+02	8.454E+04
Pu-238	1.173E+01	3.176E+06
Pu-239	1.302E+01	2.861E+06
Pu-240	1.302E+01	2.861E+06

Radionuclide	DSR (mrem/yr per pCi/g)	$DCGL_i$ (pCi/m^2)
Pu-241	2.577E-01	1.445E+08
Sb-125	3.042E-01	1.225E+08
Sr-90	2.694E+01	1.383E+06
Tc-99	4.512E+00	8.256E+06

6.11.7 BFM Drilling Spoils Scenario Mathematical Model

The mathematical model for the drilling spoils scenario includes the use of Excel and RESRAD. The spoils are treated as soil after spreading over a 0.15 m depth on the ground surface. There are five steps to calculating drilling spoils DCGLs:

1) calculate the volume of the drilling spoils brought to the ground surface (m^3) using Equation 6-5,
2) calculate the concentration in the concrete (pCi/m^2) required to produce 1 pCi/g in the drilling spoils (pCi/m^2 per pCi/g) using Equation 6-6
3) calculate the area over which the drilling spoils are spread on the ground surface by dividing the volume from Equation 6-5 by the assumed 0.15 m spread depth (m^2),
4) calculate DSRs (mrem/yr per pCi/g) with RESRAD, using the drilling spoils spread area and depth as the source term (see Section 6.11.8),
5) calculate drilling spoils DCGLs (pCi/m^2) using Equation 6-7 (see Section 6.11.9).

The diameter of the drill borehole is assumed to be 8 inches (0.203 m). The minimum depth to a floor in the Auxiliary Building, Turbine Building, or Containment Building basement is 8.5 feet (2.59 m) which is the distance from the ground surface (1004 foot elevation) to the 995.5 foot elevation of the spent fuel pit floor in the Auxiliary basement. The minimum distance maximizes the radionuclide concentration in the drilling spoils. The drill is assumed travel 1 inch (0.0254 m) into the floor before drilling is stopped due to meeting refusal from the concrete. The total drilling depth, including fill and concrete, is therefore 2.62 m. The spoils are assumed to be spread on the ground at a thickness of 0.15 m. The area of the drilling spoils after spreading is the volume divided by 0.15 m. The volume of drilling spoils is calculated in Excel using Equation 6-5 [13].

Equation 6-5

$$V_{ds} = \pi (r)^2 \, D_{bh}$$

where:

V_{ds} = volume of the drilling spoils on ground surface (m^3)
π = pi constant
r = radius of borehole (m)
D_{bh} = depth of borehole (m)

A unit concentration of 1 pCi/g is assumed to be contained in the drilling spoils. The concentration (pCi/m^2) that is required to be in the concrete to produce an average concentration of 1 pCi/g in the drilling spoils is calculated in Excel using Equation 6-6 [13]. The difference between the density of the fill and the concrete is ignored due to the low percentage of concrete in the fill.

Equation 6-6

$$C_{c,u} = \frac{V_{ds}\ 1x10^6 \rho_f C_{ds,u}}{\pi (r)^2}$$

where:

$C_{c,u}$ = concrete concentration (pCi/m^2) required to produce a unit concentration of 1 pCi/g concentration in drilling spoils (pCi/m^2 per pCi/g)
V_{ds} = volume of drilling spoils calculated by Equation 6-5 (m^3)
$1x10^6$ = conversion factor cm^3/m^3
ρ_f = density of fill (drilling spoils) (g/cm^3)
$C_{ds,u}$ = unit concentration in drilling spoils (1 pCi/g)
π = pi constant
r = radius of borehole (m)

6.11.8 BFM Drilling Spoils Scenario RESRAD Deterministic Parameters

The drilling spoils source term, after being spread on the ground surface, is treated as surface soil in RESRAD. The RESRAD deterministic parameters used to calculate soil DCGLs (see Section 6.9.5) are used to calculate drilling spoils DSRs with the three changes shown in Table 6-14. Additional parameter uncertainty analysis is not required.

Table 6-14 Changes to the Soil DCGL Deterministic Parameter Set for BFM Drilling Spoils DCGL Calculation

Parameter	Buried Pipe *insitu* Scenario
Area of contaminated zone (m^2)	0.566
Thickness of contaminated zone[1] (m)	0.15
Length parallel to aquifer flow (m)	0.85

The values in Table 6-14 are derived as follows:

- The area of contaminated zone is calculated by dividing the source volume from Equation 6-5 (V_{ds} = 8.48E-02 m^3) by the 0.15 m thickness that the spoils are spread on the ground surface (8.48E-02/0.15),
- length parallel to aquifer flow is the diameter of 0.566 m^2 (0.85 m).

The unit concrete concentration (pCi/m^2 per pCi/g) from Equation 6-6 is used in conjunction with the DSRs to calculate drilling spoils DCGLs using Equation 6-7 in Section 6.11.9.

6.11.9 BFM Drilling Spoils Scenario Initial Suite DCGLs

The BFM drilling spoils DCGLs are calculated in Excel using Equation 6-7.

Equation 6-7

$$DCGL_{ds,i} = \frac{25\ C_{c,u}}{DSR_i}$$

where:

$DCGL_{ds,i}$ = drilling spoils DCGL for radionuclide i (pCi/m^2)
$C_{c,u}$ = concrete concentration required to produce 1 pCi/g average concentration in drilling spoils (Equation 6-6, Section 6.11.7) (pCi/m^2 per pCi/g)
DSR_i = dose to source ratio for radionuclide i (mrem/yr per pCi/g)
25 = 25 mrem/yr dose criterion

The drilling spoils DSRs and DCGLs ($DCGL_{ds}$), with no IC dose correction, are listed in Table 6-15.

Table 6-15 BFM Drilling Spoils Initial Suite DSRs and Base Case DCGLs (No IC Dose Correction)

Radionuclide	DSR (mrem/yr per pCi/g)	$DCGL_{ds}$ (pCi/m^2)	Radionuclide	DSR (mrem/yr per pCi/g)	$DCGL_{ds}$ (pCi/m^2)
Am-241	3.627E-03	2.687E+10	Fe-55	4.148E-08	2.349E+15
C-14	6.142E-07	1.587E+14	H-3	8.452E-07	1.153E+14
Ce-144	4.556E-03	2.139E+10	Ni-59	2.433E-07	4.005E+14
Cm-243	1.636E-02	5.957E+09	Ni-63	6.613E-07	1.474E+14
Cm-244	1.169E-03	8.336E+10	Np-237	3.059E-02	3.186E+09
Co-58	3.390E-02	2.875E+09	Pu-238	1.874E-03	5.200E+10
Co-60	2.902E-01	3.358E+08	Pu-239	2.061E-03	4.728E+10
Cs-134	1.698E-01	5.739E+08	Pu-240	2.058E-03	4.735E+10
Cs-137	7.161E-02	1.361E+09	Pu-241	8.992E-05	1.084E+12
Eu-152	1.371E-01	7.108E+08	Sb-125	4.713E-02	2.068E+09
Eu-154	1.460E-01	6.675E+08	Sr-90	1.380E-03	7.062E+10
Eu-155	5.241E-03	1.859E+10	Tc-99	9.713E-05	1.003E+12

6.11.10 BFM Excavation Scenario Mathematical Model

As discussed in Section 6.11.2, the BFM excavation scenario conceptual model assumes that the wall with the minimum thickness is excavated and spread over a 1 m depth on the ground surface. Mixing of the contaminated and uncontaminated portions of the concrete occurs during excavation, removing rebar, and sizing the concrete to allow for use as onsite backfill. The minimum wall thickness for all basements is 2 ft [12]. The concrete is conservatively assumed to have the characteristics of soil. The soil DCGLs are therefore used to calculate dose from excavated concrete and corresponding concrete DCGLs in units of pCi/m^2. The surface area of the excavated concrete, after spreading on the ground surface, is assumed to be large such that area factors do not apply.

The concentration in the excavated concrete (pCi/g per pCi/m^2), assuming a unit activity of 1 pCi per 1 m^2 in or on the concrete, is calculated in Excel using Equation 6-8.

Equation 6-8

$$C_{ec,u} = \frac{A_{c,u}}{t_w(30.48)UA_c(1x10^4)\rho_c}$$

where:

$C_{ec,u}$ = unit concentration in excavated concrete (pCi/g per pCi/m^2)
$A_{c,u}$ = unit activity of 1 pCi over a 1 m^2 area of concrete (pCi)
t_w = minimum wall thickness in all basements (ft)
30.48 = cm/ft
UA_c = unit area of concrete wall (1 m^2)
$1x10^4$ = cm^2/m^2
ρ_c = density of concrete (2.2 g/cm^2)

The BFM concrete excavation DCGLs ($DCGL_{ec}$) are calculated in Excel using Equation 6-9 [13].

Equation 6-9

$$DCGL_{ec,i} = \frac{DCGL_{s,i}}{C_{ec,u}}$$

where:

$DCGL_{ec,i}$ = concrete excavation DCGL for radionuclide i (pCi/m^2)
$DCGL_{s,i}$ = soil DCGL (1 m thickness) for radionuclide i (pCi/g)
$C_{ec,u}$ = unit concentration in excavated concrete from Equation 6-8 (pCi/g per pCi/m^2)

The DCGL for the excavated steel liner ($DCGL_{el}$) requires a different mathematical model than that used for concrete. Mixing within the volume of the steel liner would not occur during excavation. After excavation and placement on the ground surface, the activity on the liner surface is assumed to instantly release and mix with the underlying soil over an area equal to the surface area of the liner. The excavated liner surface area, and corresponding underlying soil area is assumed to be large such that area factors do not apply. The mixing depth in soil after release from the liner may be less than 1 m. Therefore, a sensitivity analysis is performed to evaluate the $DCGL_{el}$ with soil mixing depths of 0.15 m and 1 m (see Section 6.11.10.1).

A unitized activity of 1 pCi over a 1 m^2 area of the excavated liner is assumed to mix within the underlying 1 m^2 area of soil at each depth (0.15 m or 1 m). The resulting unit concentration in soil (pCi/g per pCi/m^2) for each depth is calculated in Excel using Equation 6-10 [13].

Equation 6-10

$$C_{s,t,u} = \frac{A_u}{tUA_l 1x10^6 \rho_s}$$

where:

$C_{s,t,u}$ = unit concentration in soil for thickness t (pCi/g per pCi/m^2)
A_u = unit activity of 1 pCi over a liner area of 1 m^2 (1 pCi)
t = thickness of soil mixing zone (0.15 m or 1 m)
UA_l = unit area of liner and underlying soil (1 m^2)
1x10^6 = cm^3/m^3
ρ_s = density of soil (1.5 g/cm^3)

The liner excavation DCGLs for both the 0.15 m and 1.0 m soil mixing depths are calculated in Excel using Equation 6-11 [13].

Equation 6-11

$$DCGL_{el,i} = \frac{DCGL_{s,t,i}}{C_{s,t,u}}$$

where:

$DCGL_{el,i}$ = liner excavation DCGL for radionuclide i (pCi/m^2)
$DCGL_{s,t,i}$ = soil DCGL for thickness t (0.15 or 1 m) for radionuclide i (pCi/g)
$C_{s,t,u}$ = unit concentration in soil for thickness t (0.15 m or 1.0 m) from Equation 6-10 (pCi/g per pCi/m^2)

6.11.10.1 BFM Excavation Scenario Initial Suite DCGL

The concrete excavation DCGLs ($DCGL_{ec}$) are calculated in Excel using Equation 6-8 and Equation 6-9 [13]. The $DCGL_{ec}$ are listed in Table 6-16. The values in Table 6-16 are not corrected for IC dose.

Table 6-16 Concrete Excavation DCGLs ($DCGL_{ec}$) (No IC Dose Correction)

Radionuclide	$DCGL_{ec}$ (pCi/m^2)
Am-241	4.094E+07
C-14	1.367E+07
Ce-144	3.110E+08
Cm-243	4.104E+07
Cm-244	7.733E+07
Co-58	4.195E+07
Co-60	4.139E+06
Cs-134	5.682E+06
Cs-137	1.027E+07
Eu-152	1.039E+07
Eu-154	9.613E+06
Eu-155	4.060E+08
Fe-55	2.846E+10
H-3	1.161E+09
Ni-59	3.094E+09
Ni-63	1.130E+09
Np-237	1.022E+06
Pu-238	4.742E+07
Pu-239	4.270E+07
Pu-240	4.271E+07
Pu-241	1.395E+09
Sb-125	3.149E+07
Sr-90	2.321E+06
Tc-99	2.068E+07

The steel liner excavation scenario DCGL ($DCGL_{el}$) calculations and sensitivity analyses are performed in Excel for both the 0.15 and 1 m mixing thickness [13]. The 0.15 m soil mixing thickness resulted in the minimum $DCGL_{el}$ for all radionuclides and are therefore assigned as listed in Table 6-17. The concrete excavation DCGLs ($DCGL_{ec}$) are calculated in Excel using Equation 6-10 and Equation 6-11 [13]. The values in Table 6-17 are not corrected for IC dose.

Table 6-17 Liner Excavation DCGLs ($DCGL_{el}$) (No IC Dose Correction)

Radionuclide	$DCGL_{el}$ (pCi/m^2)	Radionuclide	$DCGL_{el}$ (pCi/m^2)
Am-241	3.155E+07	Fe-55	8.235E+09
C-14	1.349E+07	H-3	1.298E+09
Ce-144	6.179E+07	Ni-59	2.538E+09
Cm-243	1.518E+07	Ni-63	9.270E+08
Cm-244	6.624E+07	Np-237	1.063E+06
Co-58	8.170E+06	Pu-238	3.942E+07
Co-60	8.933E+05	Pu-239	3.551E+07
Cs-134	1.445E+06	Pu-240	3.551E+07
Cs-137	3.092E+06	Pu-241	1.275E+09
Eu-152	1.993E+06	Sb-125	5.990E+06
Eu-154	1.850E+06	Sr-90	2.500E+06
Eu-155	6.932E+07	Tc-99	2.313E+07

6.11.11 Fill Excavation Check Calculation

The BFM excavation scenario discussed in 6.11.10 assumes that the source term is excavated concrete. The fill excavation check assumes that 100% of the residual radioactivity in the concrete is instantly released and uniformly mixed with the fill during basement excavation. Therefore, the source term is in the fill and not in the concrete.

The concentrations in the fill are calculated assuming that the mixing volume is 1 m^3 based on a typical fill excavation process which entails using a 1.0 to 1.5 cubic yard bucket (see Section 6.11.2). The activity in a 1 m^2 area of concrete is released and mixed in a single 1 m^3 bucket load (1 m distance from the wall, i.e., SA/V ratio of 1). A SA/V ratio of 1 is the same mixing assumption used in the BFM *insitu* scenario conceptual model (see Section 6.11.2).

The objective of the calculation is to ensure that the concentration in excavated fill after 100% of the activity on the walls is instantly released to the fill is less than the concentration in excavated concrete assuming no release. The concentration in the fill is calculated using Equation 6-12.

Equation 6-12

$$C_{ef,u} = \frac{A_{c,u}}{t_f UA_c (1x10^6) \rho_f}$$

where:

$C_{ef,u}$ = unit concentration in excavated fill (pCi/g per pCi/m^2)
$A_{c,u}$ = unit activity of 1 pCi over a 1 m^2 area of concrete (pCi)
t_f = mixing thickness in fill (1m)
UA_c = unit area of concrete wall (1 m^2)
$1x10^6$ = cm^3/m^3
ρ_f = density of fill (1.49 g/cm^2)

The concentration calculated for fill using Equation 6-12 is $6.71x10^{-7}$ pCi/g per pCi/m^2. The concentration calculated for concrete using Equation 6-8 is 7.46 $x10^{-7}$ pCi/g per pCi/m^2. The concentration in the fill is less than the concentration in concrete and therefore assuming that concrete is the source term in the BFM excavation scenario is conservative.

6.12 BFM Wall/Floor Initial Suite DCGL

The BFM wall/floor DCGL is the summation of the dose from the *insitu* scenario, drilling spoils scenario, and excavation scenario. The same *insitu* and drilling spoils DCGLs apply to all basements. There is a different set of excavation scenario DCGLs for the Containment Building and all other basements due to the presence of the steel liner in Containment. Therefore, the BFM wall/floor DCGLs for the Containment Building differ from the other basements (Auxiliary Building, Turbine Building, Circulating Water Tunnels, and Intake Structure). The BFM wall/floor DCGLs are calculated in Excel using Equation 6-13 [13]. The resulting BFM wall/floor DCGLs are listed in Table 6-18. The DCGLs listed in Table 6-18 are not corrected for IC dose.

Equation 6-13

$$DCGL_{wf} = \frac{1}{(\frac{1}{DCGL_i} + \frac{1}{DCGL_{ds}} + \frac{1}{DCGL_e})}$$

where:

$DCGL_{wf}$ = BFM wall/floor DCGL
$DCGL_i$ = BFM *insitu* scenario DCGL
$DCGL_{ds}$ = BFM drilling spoils scenario DCGL
$DCGL_e$ = BFM excavation scenario DCGL

Table 6-18 BFM Wall/Floor Initial Suite DCGLs (No IC Dose Correction)

Radionuclide	BFM Wall/Floor DCGL ($DCGL_{wf}$) (Auxiliary, Turbine, Circulating Water Tunnels, Intake Structure) (pCi/m^2)	BFM Wall/Floor DCGL ($DCGL_{wf}$) (Containment) (pCi/m^2)
Am-241	4.168E+06	4.046E+06
C-14	8.629E+06	8.559E+06
Ce-144	2.553E+08	5.922E+07
Cm-243	1.003E+07	7.082E+06
Cm-244	1.369E+07	1.330E+07
Co-58	3.916E+07	8.058E+06
Co-60	3.578E+06	8.640E+05
Cs-134	4.396E+06	1.345E+06
Cs-137	7.265E+06	2.749E+06
Eu-152	1.013E+07	1.983E+06
Eu-154	9.336E+06	1.839E+06
Eu-155	3.619E+08	6.791E+07
Fe-55	1.384E+10	6.308E+09
H-3	2.208E+08	2.254E+08
Ni-59	1.219E+09	1.122E+09
Ni-63	4.451E+08	4.098E+08
Np-237	7.808E+04	7.831E+04
Pu-238	2.976E+06	2.939E+06
Pu-239	2.681E+06	2.647E+06
Pu-240	2.681E+06	2.647E+06
Pu-241	1.310E+08	1.298E+08
Sb-125	2.475E+07	5.694E+06
Sr-90	8.666E+05	8.903E+05
Tc-99	5.900E+06	6.084E+06

The dose percentage attributable to the *insitu*, drilling spoils, and excavation scenarios from the BFM wall/floor media can be calculated using Equation 6-14. Using Cs-137 as an example, the dose percentages attributable to the three scenarios for basements other than the Containment Building are 29%, 1%, and 70%, respectively.

Equation 6-14

$$DF_{s_j,i} = \frac{DCGL_{wf,i}}{DCGL_{s_j,i}}$$

where:

$DF_{sj,i}$ = fraction of total BFM wall/floor dose attributable to scenario j for radionuclide i

$DCGL_{wf,i}$ = BFM wall/floor DCGL for radionuclide i (Table 6-18)

$DCGL_{sj,i}$ = DCGL for wall/floor scenario j and radionuclide i (Table 6-13, Table 6-15, Table 6-17)

6.13 BFM Embedded Pipe Source Term, Conceptual Model, and Mathematical Model

The Auxiliary Building basement and Turbine Building basement contain floor drains that are embedded in concrete (embedded pipe). Embedded pipe is defined as pipe that runs vertically through a concrete wall or horizontally through a concrete floor. A detailed description of the embedded pipe including location, diameter, length, and internal surface area is provided in [4]. The floor drains in the Turbine Building basement do not meet the standard definition of embedded pipe. The drains are open to the basement but run under the basement slab. The conceptual model for Auxiliary Building and Turbine Building drains is described below.

The survey of embedded pipe was not included in the characterization program due to access issues. The Auxiliary floor drains are known to contain elevated levels of contamination. The Turbine drains are expected to contain minimal contamination. A radiological assessment survey of Auxiliary Building drains will be performed after the initial pass of decontamination by high pressure hydrolazing. The characterization of the Turbine Building drains will be performed as a part of the continuing characterization program. Decontamination is not expected to be required for the Turbine Building drains.

The conceptual and mathematical models for the embedded pipe dose assessment are fundamentally the same as the BFM wall/floor *insitu* model described in Section 6.11.2 and Section 6.11.3. The activity on the internal surface of the pipes is assumed to instantly release and mix into a 1 m thick layer of fill above the floor.

There are three floor elevations that contain embedded pipe: the 989 and 971 foot elevations of the Auxiliary basement and the 990 foot elevation of the Turbine basement. These three floor elevations are treated as separate areas for the embedded pipe DCGL calculations because the DCGL is dependent on the ratio of pipe internal surface area to the floor area as shown in Equation 6-15. The floor surface areas from [12] and the embedded pipe internal surface areas from [4] are provided in Table 6-19.

Table 6-19 Basement Floor Area and Embedded Pipe Internal Surface Area for the Three Basement Floor Elevations

Floor Elevation	Basement Floor Surface Area (ft^2)	Embedded Pipe Internal Surface Area (ft^2)
Auxiliary Floor 971′	6334	260
Auxiliary Floor 989′	25,109	1320
Turbine Floor 990′	26,235	2009

The DCGL is calculated in three steps: 1) calculate the concentration in fill after release from the pipe (pCi/g per pCi/m^2) in Excel using Equation 6-15, 2) calculate DSRs with RESRAD using the deterministic parameters developed for the BFM *insitu* scenario in Section 6.11.5 with the changes listed in Table 6-20, and 3) calculate the embedded pipe DCGLs in Excel using Equation 6-16.

Equation 6-15

$$C_{f,e_i} = \frac{A_{ep,u} SA_{ep,e_i} 0.0929}{SA_{f,e_i} 0.0929 D_m 1x10^6 \rho_f}$$

where:

$C_{f,ei}$ = concentration in fill from release of activity from embedded pipe at floor elevation i (pCi/g per pCi/m^2)
$A_{ep,u}$ = unit activity in embedded pipe (1 pCi per m^2),
$SA_{ep,ei}$ = embedded pipe internal surface area in floor elevation i (ft^2),
0.0929 = Conversion Factor (m^2/ft^2)
$SA_{f,ei}$ = floor surface area at elevation i (ft^2),
D_m = mix distance in fill (1 m)
$1x10^6$ = Conversion factor (g/cm^3)
ρ_f = bulk density of fill (assumed to be sand) (g/cm^3)

The embedded pipe DSRs are calculated using the deterministic parameters developed for the BFM insitu scenario in Section 6.11.5 with the changes listed in Table 6-20. The uncertainty analysis conducted for the BFM insitu scenario is assumed to apply to embedded pipe given that both involve a fully submerged source term in a basement under a clean cover. The total depth of the fill is assumed to be the same as for the BFM *insitu* scenario, i.e., 4 m. The cover depth for the embedded pipe scenario is 3.92 m which is the depth of the clean cover over the backfilled basements (0.92 m) plus the 3 m of clean fill assumed to be above the 1 m floor mixing zone.

The area of contaminated zone requires additional discussion. The largest area of the three floor elevations after unit conversion is 2437 m^2 which would be a conservative RESRAD contaminated area, i.e., the largest, for all floor elevations. However, using a contaminated area of 2437 m^2 reduces the ingestion rate of meat and milk by a factor of 2437 m^2/20,000 m^2. The plant ingestion rate is maximized for any area greater than 1,000 m^2. The meat and milk dose from the water independent pathway is zero regardless of the contaminated area size because the 3.92 m cover depth exceeds the 75^{th} percentile root depth of 3.08 m. But there is meat and milk dose from the water dependent pathway due to irrigation. The conceptual model can be interpreted as requiring the well to irrigate the entire site which exceeds 20,000 m^2. Therefore, the reduction in the water dependent meat and milk dose due to the small basement floor area source terms could be interpreted as nonconservative. Note that the well water concentration is at the theoretical maximum for both the 2437 m^2 and 20,000 m^2 contaminated areas. Therefore, to ensure that dose is not underestimated, the contaminated area for the embedded pipe DSR calculation is set to an area of 20,000 m^2. The length parallel to aquifer flow is maintained at the area corresponding 2437 m^2 which is 49 m (square root of 2437 m^2). The well water concentration is insensitive to the length parallel to aquifer flow when the source is fully submerged.

Table 6-20 Changes to BFM *insitu* Scenario Deterministic Parameters for the Embedded Pipe DSR Calculation.

Parameter	Embedded Pipe
Area of contaminated zone (m^2)	20,000
Thickness of contaminated zone (m)	1 m
Length parallel to aquifer flow (m)	49
Cover depth (m)	3.92

The conceptual model applies directly to the Auxiliary Building basement embedded pipe which runs within the concrete foundation. However, the Turbine Building basement floor drains run below the slab and have the attributes of buried pipe as well as embedded pipe. Treating the Turbine Building floor drains as buried pipe leads to the release of the activity from the internal surface of the pipe to the surrounding soil under the slab (see Section 6.14 for details regarding buried pipe conceptual model). Treating the Turbine Building floor drains as embedded pipe leads to release of activity from the pipe through openings in the floor into the basement.

The buried pipe conceptual model assumes a mix distance of 0.0254 m and a well depth of 21.4 m (see Section 6.14.2). Both models assume the source term is fully submerged in the saturated zone. Therefore, the differences between the buried pipe and embedded pipe conceptual models are essentially reduced to: 1) the mixing distance into the soil or fill after release from the pipe (0.0254 m for buried pipe and 1 m for embedded pipe) and 2) the assumed depth of the well (21.4 m for buried pipe and 4 m for embedded pipe). A check calculation was conducted using the RESRAD parameter set developed for embedded pipe with the buried pipe mixing depth and well depth of 0.0254 m and 21.4 m, respectively. The DSRs with the embedded pipe mixing depth and well depth are higher than the DSRs with the buried pipe mixing and well depths for all radionuclides and are therefore applied in the embedded pipe DSR calculation for the Turbine basement embedded pipe.

The embedded pipe DCGLs are calculated for each floor elevation in Excel using Equation 6-16. The resulting embedded pipe DSRs and DCGLs are listed in Section 6.13.1.

Equation 6-16

$$DCGL_{ep,e_i,j} = \frac{25}{DSR_{ep,j} C_{f,e_i}}$$

Where:

$DCGL_{ep,ei,j}$ = embedded pipe DCGL at floor elevation i for radionuclide j (pCi/m^2)
25 = 25 mrem/yr dose criterion
$DSR_{ep,j}$ = embedded pipe DSR for radionuclide j (mrem/yr per pCi/g)
$C_{f,ei}$ = concentration in fill from release of activity from embedded pipe at floor elevation i (Equation 6-15) (pCi/g per pCi/m^2)

6.13.1 Embedded Pipe Initial Suite DCGL

The DSRs and the results of the embedded pipe DCGL calculations using Equation 6-15 and Equation 6-16 are provided in Table 6-21. The DCGLs in Table 6-21 are not corrected for IC dose.

Table 6-21 Embedded Pipe Initial Suite DSRs and DCGL (No IC Dose Correction)

Radionuclide	DSR (mrem/yr per pCi/g)	$DCGL_{ep}$ 971′ Auxiliary Floor Drains (pCi/m^2)	$DCGL_{ep}$ 989′ Auxiliary Floor Drains (pCi/m^2)	$DCGL_{ep}$ 990′ Turbine Floor Drains (pCi/m^2)
Am-241	1.851E+00	4.900E+08	3.83E+08	2.63E+08
C-14	5.787E-01	1.568E+09	1.224E+09	8.406E+08
Ce-144	4.811E-03	1.886E+11	1.473E+11	1.011E+11
Cm-243	5.938E-01	1.528E+09	1.193E+09	8.192E+08
Cm-244	4.747E-01	1.912E+09	1.493E+09	1.025E+09
Co-58	6.528E-03	1.390E+11	1.085E+11	7.452E+10
Co-60	1.691E-01	5.366E+09	4.190E+09	2.877E+09
Cs-134	1.365E-01	6.648E+09	5.191E+09	3.564E+09
Cs-137	1.084E-01	8.371E+09	6.537E+09	4.487E+09
Eu-152	9.568E-03	9.484E+10	7.406E+10	5.084E+10
Eu-154	1.392E-02	6.519E+10	5.090E+10	3.495E+10
Eu-155	2.164E-03	4.193E+11	3.274E+11	2.248E+11
Fe-55	3.283E-04	2.764E+12	2.158E+12	1.482E+12
H-3	3.725E-02	2.436E+10	1.902E+10	1.306E+10
Ni-59	4.373E-03	2.075E+11	1.620E+11	1.112E+11
Ni-63	8.698E-03	1.043E+11	8.146E+10	5.593E+10
Np-237	1.476E+02	6.148E+06	4.801E+06	3.296E+06
Pu-238	2.796E+00	3.246E+08	2.534E+08	1.740E+08
Pu-239	4.086E+00	2.221E+08	1.734E+08	1.190E+08
Pu-240	3.782E+00	2.399E+08	1.874E+08	1.286E+08
Pu-241	6.079E-02	1.493E+10	1.166E+10	8.002E+09
Sb-125	6.833E-02	1.328E+10	1.037E+10	7.119E+09
Sr-90	4.010E+00	2.263E+08	1.767E+08	1.213E+08
Tc-99	1.151E+00	7.884E+08	6.156E+08	4.226E+08

6.14 Buried Pipe DCGL

6.14.1 Buried Pipe Source Term Configuration and Spatial Variability

Buried piping is defined as below ground pipe located outside of structures and basements. Note that buried electrical or structural commodities are not categorized as buried pipe. See LTP Chapter 5 for the FSS approach to such commodities.

A detailed description and breakdown of the buried pipe, including maps and the site buildings or areas that are served by each pipe, is provided in [4]. The buried pipe to remain at license termination is comprised predominantly of storm drains. There is also one service water piping system to remain that currently serves the maintenance shop. The total length and interior surface area the End State buried pipe are listed in Table 6-22. The internal diameters of the storm drains range from 8 inches to 84 inches. The internal diameter of the service water pipe is 2.9 inches.

Table 6-22 Total Length and Surface Area of Buried Piping

Piping	Length (m)	Interior Surface Area (m^2)
Storm Drain	955.9	2,167.8
Service Water	54.9	12.7
Total	1,010.8	2,181

The buried pipe was not surveyed during characterization. The pipe will be surveyed later as a part of "continuing characterization" [9]. However, these systems are not expected to contain residual radioactivity beyond perhaps the occasional presence of low concentrations near the detection limits.

6.14.2 Buried Pipe Conceptual Model, Source Term Abstraction and Release

The buried pipe conceptual model includes two exposure scenarios: 1) the buried pipe remains in the as-left geometry (*insitu* scenario), and 2) the buried pipe is excavated and placed on the ground surface (excavation scenario). The doses from the two scenarios are summed to calculate the buried pipe DCGL. The conceptual model conservatively assumes that all of the pipe is present in both the *insitu* and excavation scenarios simultaneously. The environmental and exposure pathways applicable to buried pipe are listed in Table 6-2.

The conceptual model for the *insitu* scenario assumes that the residual radioactivity on the internal surfaces of the pipe is instantly released and mixed into a 0.0254 m layer of soil over a contiguous area equal to the total internal surface area of the buried pipe (2,181 m^2). No credit is taken for the presence of the pipe to reduce environmental transport. A thin 0.0254 m mixing layer is justified because there are no mechanical mixing mechanisms when the pipes are left buried and undisturbed. The pipes are assumed to be located in the soil immediately below the 1.1 m thick vadose zone (see Section 6.14.2) and 100% submerged in groundwater. The pipe thicknesses are ignored. The source term is therefore a 0.0254 m layer of soil in the saturated zone with a 1.1 m clean cover.

The conceptual model for the excavation scenario assumes that the pipes are excavated and brought to the ground surface. The residual radioactivity on the internal surfaces of the pipe is instantly released and mixed into a layer of soil on the ground surface over a contiguous area equal to the total internal surface area of the buried pipe (2,181 m^2). A sensitivity analysis is performed using 0.15 m and 1 m mixing depths (see Section 6.14.3). Either depth is reasonable given the extensive disturbance of the ground surface during the large-scale excavation required to remove the pipe. Assuming a vadose zone thickness of 1.1 from the site general conceptual model (see Section 6.7), the unsaturated zone thickness for the 0.15 m and 1 m mixing depths are 0.95 m and 0.1 m, respectively.

A mixing thickness of 0.15 m results in the highest dose for all initial suite radionuclides except C-14, H-3 and Tc-99 [13]. A mixing thickness of 1.0 m is therefore used for C-14, H-3 and Tc-99 and a thickness of 0.15 m for all other radionuclides.

6.14.3 Buried Pipe Mathematical Model

The units of the buried pipe DCGLs are dpm/100 cm^2 to match the output of the FSS instrumentation use for buried pipe surveys. The first step in the buried pipe DCGL calculation is to calculate the concentration in soil after the release of a unit amount of activity from the pipe (1 dpm). The unitized source terms (pCi/g per dpm/cm^2) for the *insitu* and excavation scenarios are calculated in Excel using Equation 6-17 [13].

Equation 6-17

$$C_{s,u,i} = \frac{A_{bp,u}/2.22}{(SA_{bp,u} t_{m,s} \rho_s)}$$

where:

$C_{s,u.i}$ = unitized soil concentration for buried pipe scenario i (pCi/g per dpm/cm^2)
$A_{bp,u}$ = unit activity in pipe over a 1 cm^2 area (1 dpm)
2.22 = conversion factor (dpm/pCi)
$SA_{bp,u}$ = unit surface area of buried pipe (1 cm^2)
$t_{m,i}$ = thickness of soil mixing zone for buried pipe scenario *i* (*insitu* scenario 2.54 cm or excavation scenario 15 cm and 100 cm)
ρ_s = density of soil (g/cm^3)

RESRAD is used to calculate DSRs (mrem/yr per pCi/g) for both the *insitu* and excavation scenarios. The selection of the RESRAD input parameters for calculating the DSRs is discussed Section 6.14.4. The unitized source terms from Equation 6-17 are used in Equation 6-19 (in Section 6.14.5) to calculate the buried pipe DCGLs.

6.14.4 Buried Pipe RESRAD Uncertainty Analysis and Deterministic Parameter Selection

6.14.4.1 Buried Pipe Excavation Scenario

The RESRAD deterministic parameters used to calculate soil DCGLs (see Section 6.9.5) are used to calculate buried pipe excavation scenario DSRs with three changes as listed in Table 6-23. The results of the soil uncertainty analysis in Section 6.9.5 are considered applicable because the excavation scenario source term is a layer of soil on the ground surface which is the same as the soil conceptual model. The area of primary contamination is changed to 2,181 m^2 and the depth of contamination to 0.15 m or 1 m. The length parallel to aquifer flow is also changed to 26 m to reflect the 2181 m^2 area of primary contamination. The lowest DCGL, from either the 0.15 m or 1.0 m mixing distance is selected (see Section 6.14.5).

Table 6-23 Parameter Changes to RESRAD Soil DCGL Parameter Set for Buried Pipe Excavation Scenario DSR Calculation

Parameter	Buried Pipe Excavation
Area of Contaminated Zone (m^2)	2,181
Thickness of Contaminated zone[1] (m)	0.15 m or 1 m
Length Parallel to Aquifer Flow (m)	26
Thickness of unsaturated zone[2]	0.1 m for 1 m mix thickness 0.95 m for 0.15 m mix thickness

1) Thickness of contaminated zone is the mixing depth
2) The site conceptual model vadose zone thickness is 1.1 m (see Section 6.7). The unsaturated zone thickness is 1.1 m minus the mixing depth.

6.14.4.2 Buried Pipe Insitu Scenario

The conceptual model for the buried pipe *insitu* scenario is substantially different from the soil conceptual model. A 1.1 m cover is present and the source term is a thin layer that is fully submerged. Therefore, unlike the buried pipe excavation scenario, a separate uncertainty analysis is performed for the *insitu* scenario.

The process for performing the uncertainty analysis is the same as described in Section 6.9.4 for soil. The soil deterministic parameters and PDFs listed in Attachment 6-1 are used with modifications as shown in Table 6-24. The deterministic parameters in Table 6-24 except for the primary radionuclide K_d values) are also used in the RESRAD DSR calculations.

Table 6-24 Changes to Soil Uncertainty Analysis Parameter Set in Attachment 6- Required for Buried Pipe *insitu* Scenario Uncertainty Analysis

Parameter	Buried Pipe *insitu* Scenario
Cover depth (m)	1.1
Cover erosion rate (m/y)	RESRAD Default PDF
Contaminated zone erosion rate[1]	0
Area of contaminated Zone (m^2)	2,181
Thickness of contaminated zone (m)	0.0254 m
Length parallel to aquifer flow (m)	26
Contaminated fraction below the water table	1
Contaminated zone density (g/cm^3)	1.49
Contaminated zone total porosity (unitless)	0.45
Contaminated zone field capacity (unitless)	0.24
Contaminated zone hydraulic conductivity (m/yr)	4350
Contaminated zone deterministic K_d values changed to the 50^{th} percentile of PDF in Table 2.13.1 or 2.13.10 of [10]	Radionuclide Dependent
Contaminated zone K_d PDFs changed to the distributions in Table 2.13.1 or 2.13.10 of [10]	Radionuclide Dependent PDF

1) Contaminated zone is completely submerged in saturated zone

The changes to the contaminated zone hydrogeological parameters and K_d values listed in Table 6-24 are required to satisfy the conceptual model which assumes that the buried pipe is fully submerged in the saturated zone as opposed to being in the vadose zone.

Table 6-25 and Table 6-26 list the parameters from Attachment 6-1 that are stochastic and require assessment by uncertainty analysis for the buried pipe *insitu* scenario. The uncertainty analysis results and selected deterministic parameters to be applied in the RESRAD DSR calculation are also listed in Table 6-25 and Table 6-26.

Table 6-25 Buried Pipe *insitu* Scenario RESRAD Uncertainty Analysis Results for Non-Nuclide Specific Parameters and Selected Deterministic Values

Parameter	Correlation to Dose	Radionuclide	Basis of Deterministic Parameter Selection	Selected Deterministic Value
Cover erosion rate (m/yr)	positive	Am-241, Co-60, Cs-134, Cs-137, Eu-152, Eu-154, Eu-155, Pu-238, Pu-239, Pu-240, Pu-241, Sb-125		2.92E-03
Contaminated zone b parameter (unitless)	NS[1]	NA		3.6
Evapotranspiration coefficient (unitless)	positive	C-14[2], Pu-241[2], H-3, Tc-99		0.87
Wind Speed (m/s)				3.75
Runoff coefficient (unitless)	positive	H-3[2], Tc-99		0.63
Depth of roots (m)	negative	Am-241, Ce-144, Cm-244, Pu-239, Pu-241, Pu-241, Fe-55[2], Ni-59,		1.23
Well pump intake depth (m)	negative	Am-241, C-14, Cm-243[2], Cm-244, Pu-238, Pu-239, Pu-240, Pu-241, Fe-55, H-3, Np-237, Ni-59[2], Tc-99		21.4
Mass loading for inhalation (g/m^3)	NS	NA		2.35E-05
Indoor dust filtration factor (unitless)	NS	NA		0.55
Depth of Soil Mixing Layer (m)	NS	NA		0.23
Wet foliar interception fraction of leafy vegetables (unitless)	NS	NA		0.58
Weathering removal constant all vegetation (unitless)	NS	NA		33
Wet weight crop yield of fruit, grain, and mon-leafy vegetable (kg/m^2)	NS	NA		1.75

1) not sensitive (NS)
2) PRCC is less than |0.25| but some degree of correlation is indicated. Confirmed to be correlated by single parameter sensitivity analysis.

Table 6-26 Buried Pipe *insitu* Scenario RESRAD K_d Parameter Uncertainty Analysis Results for Contaminated Zone and Saturated Zone and Selected Deterministic Values[1]

Radionuclide	Correlation to Dose	Basis of Deterministic Parameter Selection	Selected Deterministic Value (cm^3/g)
Am-241	negative	25th percentile	269
C-14	negative	25th percentile	1.26
Ce-144	NS[2]	50th percentile	399
Cm-243	negative	25th percentile	572
Cm-244	negative	25th percentile	572
Co-58	NS	50th percentile	260
Co-60	NS	50th percentile	260
Cs-134	NS	50th percentile	528
Cs-137	NS	50th percentile	528
Eu-152	NS	50th percentile	829
Eu-154	NS	50th percentile	829
Eu-155	NS	50th percentile	829
Fe-55	NS	50th percentile	321
H-3	negative	25th percentile	0.043
Ni-59	NS	50th percentile	130
Ni-63	NS	50th percentile	130
Np-237	negative	25th percentile	5.49
Pu-238	negative	25th percentile	156
Pu-239	negative	25th percentile	156
Pu-240	negative	25th percentile	156
Pu-241	negative	25th percentile	156
Sb-125	NS	50th percentile	16.9
Sr-90	NS	50th percentile	22
Tc-99	negative	25th percentile	0.019

1) The buried pipe is fully submerged in the saturated zone. An unsaturated zone is not present.
2) not sensitive (NS)

The parameters in Attachment 6-1, as modified by the deterministic parameters listed in Table 6-24 and those selected through uncertainty analysis and listed in Table 6-25 and Table 6-26, are used in RESRAD to calculate the buried pipe *insitu* scenario DSRs. Before the RESRAD runs were made, two additional sensitivity analyses were performed using the DSR parameter set.

First, the performance of the RESRAD nondispersion model for the buried pipe *insitu* scenario, which assumes that the source term is fully submerged, was checked to ensure the results are applicable and reasonable. The expectation is that the well water concentration will be the theoretical maximum, as calculated using Equation 6-2, multiplied by the ratio of the source thickness divided by the well pump intake depth. The RESRAD Cs-137 well water concentration at time = 0 years is 2.247×10^{-3} pCi/L using the *insitu* buried pipe scenario source thickness of 0.0254 m, the K_d of 528 cm^3/g, a source term concentration of 1 pCi/g, and well intake depth of 21.4 m. The theoretical maximum calculated using Equation 6-2 is 1.893 pCi/L. Using Equation 6-18, it is seen that the Cs-137 well water concentration from the insitu scenario is the value expected. This validates the use of the RESRAD nondispersion groundwater model for the fully submerged buried pipe *insitu* scenario source term.

Equation 6-18

$$C_{ww} = 1.893\left(\frac{0.0254}{21.4}\right) = 2.247x10^{-3}$$

where:

C_{ww} = well water concentration (pCi/L)
1.893 = theoretical maximum well water concentration (pCi/L)
0.0254 = source thickness for buried pipe insitu scenario (m)
21.4 = well intake depth (m)

A second sensitivity analysis was performed to ensure that placing the source in the saturated zone is conservative. As discussed in Section 6.2.2, the water table elevation varies seasonally with the stage of the Missouri river. Most of the year the water table elevation is expected to be lower than 1.1 m bgs. The buried pipe could therefore be in the vadose zone for most of the year. A sensitivity analysis was performed by placing the 0.0254 m thick source into the vadose zone (i.e., setting the contaminated zone fraction below the water table to 0). The thickness of the unsaturated zone was set to a thin layer of 0.1 m. Using Cs-137 as Sr-90 as examples, the DSRs with a fully submerged source term are slightly higher (less than 2%). The fully submerged source term is conservative albeit to a minor extent.

6.14.5 Buried Pipe Initial Suite DCGL

DCGLs for the buried pipe excavation scenario are calculated for both a 0.15 and 1.0 m mixing distance. The DSR increases with increasing thickness of the contaminated zone (mixing thickness) when the source term concentration is constant but the concentration decreases as a function of mixing thickness. The lowest DCGL, from either the 0.15 m or 1.0 m mixing distance is selected. For the insitu scenario, the DCGL is based on a thickness of 0.0254 m.

The DCGLs for both the *insitu* and excavation scenarios are calculated in Excel using Equation 6-19 [13].

Equation 6-19

$$DCGL_{bp,s,i} = \left(\frac{25}{C_{bp,u,s} DSR_{bp,i}}\right)100$$

where:

$DCGL_{bp,s,i}$ = buried pipe DCGL for scenario *s* and radionuclide *i* (dpm/100 cm^2)
25 = 25 mrem/yr dose criterion
$C_{bp,u,s}$ = unitized soil concentration for buried pipe scenario *s* calculated using Equation 6-17 (pCi/g per dpm/cm^2)
$DSR_{bp,i}$ = buried pipe DSR for radionuclide i (mrem/yr per pCi/g)
100 = 100 cm^2 to calculate the DCGL in units of dpm/100 cm^2.

For the excavation scenario, a mixing thickness of 0.15 m results in the highest dose for all initial suite radionuclides except C-14, H-3, and Tc-99 [13]. A mixing thickness of 1.0 m is therefore used to calculate excavation DCGLs for C-14, H-3, and Tc-99 and a thickness of 0.15 m is used for all other radionuclides. The buried pipe insitu and excavation DCGLs are listed in Table 6-27.

Table 6-27 Buried Pipe Initial Suite Excavation and *insitu* Scenario DCGLs (No IC Dose Correction)

Radionuclide	Buried Pipe Excavation Scenario DCGL (dpm/100 cm^2)	Buried Pipe *insitu* Scenario DCGL (dpm/100 cm^2)
Am-241	7.442E+05	7.905E+05
C-14	3.596E+06	8.874E+06
Ce-144	1.430E+06	1.213E+08
Cm-243	3.511E+05	1.387E+06
Cm-244	1.548E+06	1.736E+06
Co-58	1.898E+05	4.746E+07
Co-60	2.103E+04	1.836E+06
Cs-134	3.670E+04	1.238E+06
Cs-137	8.432E+04	1.558E+06
Eu-152	4.610E+04	2.653E+08
Eu-154	4.282E+04	1.826E+08
Eu-155	1.611E+06	1.185E+09
Fe-55	1.341E+09	5.346E+09
H-3	5.409E+07	6.443E+07
Ni-59	2.400E+08	2.817E+08
Ni-63	8.763E+07	1.029E+08
Np-237	3.955E+04	1.987E+04
Pu-238	9.576E+05	7.360E+05
Pu-239	8.624E+05	6.626E+05
Pu-240	8.630E+05	6.626E+05
Pu-241	3.018E+07	2.766E+07
Sb-125	1.382E+05	2.024E+07
Sr-90	8.093E+04	8.279E+04
Tc-99	5.938E+05	1.475E+06

The excavation scenario and insitu scenario DCGLs are summed to calculate the final buried pipe DCGL in Excel using Equation 6-20. The resulting DCGLs are listed in Table 6-28. The DCGLs in Table 6-28 are not corrected for IC dose.

Equation 6-20

$$DCGL_{bp,i} = \frac{1}{\left({}^{1}/_{DCGL_{bpi,i}} + {}^{1}/_{DCGL_{bpe,i}}\right)}$$

where:

$DCGL_{bp,i}$ = Buried pipe DCGL for radionuclide i
$DCGL_{bpi,i}$ = Buried pipe *insitu* scenario DCGL for radionuclide i
$DCGL_{bpe,i}$ = Buried pipe excavation scenario DCGL for radionuclide i

Table 6-28 Buried Pipe Initial Suite DCGLs (No IC Dose Correction)

Radionuclide	$DCGL_{bp}$ (dpm/100 cm^2)	Radionuclide	$DCGL_{bp}$ (dpm/100 cm^2)
Am-241	3.833E+05	Fe-55	1.072E+09
C-14	2.559E+06	H-3	2.941E+07
Ce-144	1.413E+06	Ni-59	1.296E+08
Cm-243	2.802E+05	Ni-63	4.732E+07
Cm-244	8.182E+05	Np-237	1.323E+04
Co-58	1.890E+05	Pu-238	4.161E+05
Co-60	2.079E+04	Pu-239	3.747E+05
Cs-134	3.564E+04	Pu-240	3.748E+05
Cs-137	7.999E+04	Pu-241	1.443E+07
Eu-152	4.609E+04	Sb-125	1.372E+05
Eu-154	4.281E+04	Sr-90	4.093E+04
Eu-155	1.609E+06	Tc-99	4.234E+05

6.15 Insignificant Contributor Radionuclide Dose and Selection of ROC

In accordance with NUREG-1757 [16], radionuclides that contribute, in aggregate, less than 10% of the 25 mrem/yr dose criteria are considered to be “insignificant contributors” (IC). The IC radionuclides may be eliminated from further consideration during FSS but the aggregate dose from the IC radionuclides must be accounted for in the compliance demonstration. The remaining radionuclides, after the IC radionuclides are removed, and the associated mixture fractions, comprise the radionuclides of concern (ROC) that will undergo detailed evaluation during FSS. The detailed analysis and evaluation of the relative doses, the IC dose, and the selection of the ROC is provided in [8]. A summary is provided here.

The relative dose fraction for each initial suite radionuclide is calculated for each DCGL (Table 6-7, Table 6-13, Table 6-15, Table 6-16, Table 6-17, Table 6-18, Table 6-21, and Table 6-28) and mixture (Containment and AB/TB/RWPB). The IC radionuclide dose from the 25 mrem/yr concentrations for the LLBP scenarios were also evaluated. For each DCGL/mixture pair, the evaluation uses the radionuclide mixture fractions of the initial suite of radionuclides to select the radionuclides that are ICs and determine the aggregate dose from the eliminated radionuclides. The radionuclides remaining after the IC radionuclides are eliminated are the ROCs that will undergo detailed assessment during FSS. The IC dose fraction will be accounted for by adjusting the DCGLs for each ROC.

For basement walls/floors and embedded pipe, the ROCs were found to be Cs-137, Co-60, Eu-152, Sr-90, and C-14. The aggregate dose percentage from the IC radionuclides in basement floors/walls ranges from 0.52% (IC dose fraction of 5.2E-03) to 1.25% (IC dose fraction of 1.25E-02). The IC dose fraction for embedded pipe is 4.94% (IC dose fraction of 4.94E-02).

The ROCs for soil and buried pipe are Cs-137, Co-60, Eu-152, and C-14. The aggregate dose percentage from the IC radionuclides in soil ranges from 0.82% (IC dose fraction of 8.2E-03) to 2.39% (IC dose fraction of 2.39E-02). The maximum value of 2.39% applies to the soil DCGL for a 1 m source thickness. Note that there is no indication that subsurface soil contamination is present at FCS. The 1 m thickness soil DCGLs are included primarily to evaluate confirmation sampling of the absence of subsurface sampling during FSS and is not a considered a currently contaminated media at FCS. The 0.15 soil thickness DCGL represents the soil media at FCS. The IC dose for buried pipe is 1.05% (IC dose fraction of 1.05E-02). The ROCs for soil are also assigned to above ground buildings. The ROCs for fill material are also Cs-137, Co-60, Eu-152, and C-14 (see Section 6.19 for discussion of fill DCGLs) with an IC radionuclide dose fraction of 8.88% (IC dose fraction of 8.88E-02).

The renormalized mixture fractions for the ROC only are provided in Table 6-29.

Table 6-29 ROC and Renormalized Mixture Fractions

Radionuclide	CB Mix Fraction	AB/TB/RWPB Mix Fraction
C-14	9.27E-01	8.27E-02
Co-60	1.87E-03	1.85E-02
Cs-137	6.74E-02	8.94E-01
Eu-152	3.31E-03	1.15E-03
Sr-90	2.91E-04	4.04E-03
Sum	1.0	1.0

6.16 ROC DCGLs for Soil, BFM Wall/Floor, BFM Embedded Pipe, and Buried Pipe

For the basement walls/floors, embedded pipe, soil, and buried pipe a dose fraction of 0.05 is assigned to the IC radionuclides that are eliminated from detailed assessment during FSS. For embedded pipe and fill an IC dose of 0.1 is assigned. The ROC selected in Section 6.15 are adjusted to account for the IC dose by multiplying the ROC DCGLs for basement walls/floors, soil, and buried pipe by 0.95 (1.0 – 0.05). The ROC DCGLs for embedded pipe and fill are adjusted to account for the IC dose by multiplying the ROC DCGLs by 0.9 (1.0 – 0.1). The IC adjustment factors exceed the actual IC dose fractions in order to provide a margin to account for uncertainty in the radionuclide mixtures. The IC adjusted DCGLs for soil, BFM wall/floor, embedded pipe, and buried pipe are provided in Table 6-30, Table 6-31, Table 6-32, and Table 6-33.

Table 6-30 Soil DCGL for ROC (Adjusted for IC Dose)

Radionuclide	Soil DCGL (pCi/g) 0.15 m	Soil DCGL (pCi/g) 1.0 m
C-14	5.70E+01	9.68E+00
Co-60	3.77E+00	2.93E+00
Cs-137	1.31E+01	7.27E+00
Eu-152	8.41E+00	7.36E+00

Table 6-31 BFM Wall/Floor DCGL for ROC (Adjusted for IC Dose)

Radionuclide	BFM Wall/Floor DCGL (pCi/m2)	
	Auxiliary, Turbine, Circulating Water Tunnels, Intake Structure	Containment
C-14	8.20E+06	8.13E+06
Co-60	3.40E+06	8.21E+05
Cs-137	6.90E+06	2.61E+06
Eu-152	9.62E+06	1.88E+06
Sr-90	8.23E+05	8.46E+05

Table 6-32 BFM Embedded Pipe DCGL for ROC (Adjusted for IC Dose)

Radionuclide	BFM Embedded Pipe DCGL (pCi/m2)		
	Auxiliary Floor 971' elevation (pCi/m2)	Auxiliary Floor 989' elevation (pCi/m2)	Turbine Floor 990' elevation (pCi/m2)
C-14	1.41E+09	1.10E+09	7.57E+08
Co-60	4.83E+09	3.77E+09	2.59E+09
Cs-137	7.53E+09	5.88E+09	4.04E+09
Eu-152	8.54E+10	6.67E+10	4.58E+10
Sr-90	2.04E+08	1.59E+08	1.09E+08

Table 6-33 Buried Pipe DCGL for ROC (Adjusted for IC Dose)

Radionuclide	Buried Pipe DCGL (dpm/100 cm^2)
C-14	2.43E+06
Co-60	1.98E+04
Cs-137	7.60E+04
Eu-152	4.38E+04

6.17 Above Ground Building DCGL For ROC

As indicated in LTP Chapter 3, all impacted FCS buildings other than those listed below, will be demolished and removed to a depth of at least 3 feet below grade or removed in their entirety. The above ground buildings to remain are:

- Training Building,
- FLEX Building,
- Owner Controlled Area Entrance Building,
- 3451 Old Building,
- 3451 New Building, and
- 1251 Control and Switchgear Building.

Scan surveys and static measurements were performed in the buildings during characterization. There were no scan alarms during the survey. The gross beta static measurement and swab results were all at low levels and likely at or near background levels. However, a reference area was not established, or deemed necessary, for the characterization effort and net measurements were not determined. The classification for the buildings to remain is Class 3.

The site conditions listed in Section H.2.2 of [16] apply to the remaining above ground FCS buildings. The screening values in NUREG-1757, Vol 2, Table H-1 [16] are therefore applicable to the FSS of above ground buildings and will be used. Table H-1 does not include a screening value for Eu-152. The Eu-152 screening value applied is the "Pcrit 0.90" value from Table 5.19 of NUREG/CR-5512, Volume 3 [11]. The IC adjusted above ground building DCGLs are listed in Table 6-34. The IC dose fraction assigned to soil, i.e., 0.05, is also assigned to above ground buildings.

Table 6-34 Above Ground Building DCGL for ROC (Adjusted for IC Dose)

Radionuclide	Above Ground Building DCGL dpm/100 cm^2	Above Ground Building DCGL Adjusted for IC Dose dpm/100 cm^2
C-14	3.70E+06	3.52E+06
Co-60	7.10E+03	6.75E+03
Cs-137	2.80E+04	2.66E+04
Eu-152	1.27E+04	1.21E+04

6.18 Existing Groundwater Dose Conversion Factors For ROC

The existing groundwater (GW) dose conversion factors (DCF) (mrem/yr per pCi/L) are derived from the results of the FCS embedded pipe DSR RESRAD analysis. The scenario used is immaterial as long as the contaminated zone area is at least 20,000 m^2 but the RESRAD results need to include non-negligible well water concentrations. The water dependent dose includes exposure from ingestion of drinking water, ingestion of plants subject to irrigation, and ingestion of meat and milk from cows that consume well water directly and consume fodder that is subject to irrigation. The existing GW DCFs are calculated in Excel using Equation 6-21 [13] and listed in Table 6-35.

Equation 6-21

$$DCF_{egw,i} = \frac{D_{wd,t,i}}{C_{ww,t,i}}$$

where:

$DCF_{egw,i}$ = dose conversion factor for radionuclide i (mrem/yr per pCi/L)
$D_{wd,t,i}$ = water dependent dose at time t for radionuclide i (mrem/yr)
$C_{ww,t,i}$ = well water concentration at time t for radionuclide i (pCi/L)

Table 6-35 Existing Groundwater Dose Conversion Factors for ROC

Radionuclide	Water Dependent Dose[1] at t=1 yr (mrem/yr)	Well Water Concentration[2] at t = 1 yr (pCi/L)	Dose Conversion Factor mrem/yr per pCi/L
C-14	0.45	1.663E+02	2.68E-03
Co-60	0.15	5.886E+00	2.52E-02
Cs-137	0.11	1.544E+00	6.86E-02
Eu-152	0.01	2.505E+00	3.63E-03
Sr-90	3.95	3.585E+01	1.10E-01

1) RESRAD Report "FCS Embedded Pipe DSR"
2) RESRAD Report "FCS Embedded Pipe DSR Concentration Report"

6.19 Basement Fill Material DCGLs

The material to be used as basement backfill is the soil that was excavated as part of the rail spur expansion project (see LTP Chapter 3). Approximately 132,000 cubic yards of spoils produced from the excavation of the rail spur area are planned to be used on-site as fill material in basement structures after FSS of the structure surfaces (and embedded pipe). Because the fill material was excavated from an impacted area of the FCS site, albeit Class 3, the dose from residual radioactivity in the fill must be assessed.

The basement will be filled with the soil up to site grade (1004 feet AMSL). The backfill material will therefore have two components: the "cover" soil which is in the vadose zone above the basement structure and the "fill" soil which is within the basement structure and assumed to be in the saturated zone per the BFM conceptual model (see Section 6.11.2). The dose from the "cover" soil is accounted for by the dose calculated for the Class 3 open land area represented by the fill. The dose from the "fill" soil within the basements is assumed to be fully submerged in the saturated zone consistent with the BFM conceptual model). The "cover" soil is assumed to be 0.92 m (3 ft) deep and the fill soil 4 m deep consistent with the BFM conceptual model.

The dose from the fill material within the basement is calculated as a stand-alone media, i.e., no other residual radioactivity is assumed to be within the basements. The fill material is therefore a stand-alone dose component in the compliance dose calculation (see Equation 6-26). The fill material is assumed to contain uniform concentrations of residual radioactivity as determined through the FSS of the Class 3 open land area represented by the rail spur excavation area. Therefore, the concentrations of residual radioactivity in the "cover" component of the fill, i.e., at site grade (1004 foot elevation), is equivalent to the concentration of residual radioactivity at depth within the basements. Therefore, if the fill were excavated from the basements, the concentration of residual radioactivity in the excavated material would be equivalent to that contained at site grade. Because the dose from the soil/fill at site grade is already accounted for by the dose assessment of the open land survey unit represented by the fill no additional consideration of fill excavation, or drilling spoils, from the basement is required. However, the dose from the fully submerged fill material within the basement is a separate environmental pathway that requires assessment.

The BFM conceptual model assumes that all of the residual radioactivity is released from the walls/floors and is then contained in the fill material. The geometry of the fill source term after release from the walls/floors in the BFM model is equivalent to the geometry of the fill source term after placement in the basement. Therefore, the BFM *insitu* scenario conceptual and mathematical models described in Sections 6.11.2 and 6.11.3 apply to the fill material. One difference in the fill geometry, as compared to the BFM *insitu* geometry, is that the source term area would exceed the 16,700 m^2 assumed in the BFM *insitu* scenario. For areas less than 20,000 m^2 (and greater than 1,000 m^2), RESRAD decreases the dose from the meat and milk pathways. Although the differences are minor, a separate DCGL was calculated for the fill using the deterministic parameters selected in Section 6.11.5 for the BFM *insitu* scenario with the area of contamination changed to 20,000 m^2 and the length parallel to aquifer flow changed to 83 m (20,000 m^2 / (60 m x 4 m)).

The fill DCGL calculation assumes that all of the soil originating from the rail spur area contains uniform residual radioactivity at all depths. This is a conservative assumption for a Class 3 area where the residual radioactivity is expected to be in the 0.15 m surface layer at or near background concentrations.

The resulting DCGLs for the initial suite radionuclides and the IC adjusted DCGLs for the ROC are provided in Table 6-36.

Table 6-36 Fill *insitu* Scenario DCGLs (no IC Dose Correction)

Radionuclide	Fill *insitu* DCGL mrem/yr per pCi/g
Am-241	3.115E+00
C-14	1.433E+01
Ce-144	1.023E+03
Cm-243	8.928E+00
Cm-244	1.117E+01
Co-58	4.575E+02
Co-60	1.767E+01
Cs-134	1.197E+01
Cs-137	1.508E+01
Eu-152	6.108E+02
Eu-154	4.200E+02
Eu-155	2.703E+03

Radionuclide	Fill *insitu* DCGL mrem/yr per pCi/g
Fe-55	1.716E+04
H-3	1.733E+02
Ni-59	1.190E+03
Ni-63	4.345E+02
Np-237	5.655E-02
Pu-238	2.131E+00
Pu-239	1.919E+00
Pu-240	1.919E+00
Pu-241	9.916E+01
Sb-125	8.165E+01
Sr-90	8.859E-01
Tc-99	5.399E+00

Fill *insitu* Scenario ROC DCGLs (IC Dose Corrected)

Radionuclide	Fill *insitu* DCGL Adjusted for IC Dose mrem/yr per pCi/g
C-14	1.29E+01
Co-60	1.59E+01
Cs-137	1.36E+01
Eu-152	5.50E+02

6.20 Soil Area Factors for ROC

Soil area factors are calculated for the ROC in two steps. First, calculate the a DCGL using the RESRAD deterministic parameters described in Section 6.9.5 with the area of contaminated zone changed to the areas listed in Table 6-37 and the length parallel to aquifer flow changed to the square root of the area. The second step is to divide the DCGLs for each area by the DCGL for the given radionuclide listed in Table 6-7. The soil area factors are calculated in Excel [13] and are listed in Table 6-37 and Table 6-38.

Table 6-37 Soil Area Factors for ROC 0.15 m Thickness

Radionuclide	Soil Area Factors					
	$1m^2$	$2m^2$	$5m^2$	$10m^2$	$100m^2$	$143\ m^2$
C-14	3.42E+05	1.43E+05	4.22E+04	1.62E+04	5.89E+02	3.48E+02
Co-60	1.23E+01	6.98E+00	3.76E+00	2.48E+00	1.29E+00	1.24E+00
Cs-137	1.44E+01	8.25E+00	4.47E+00	2.94E+00	1.56E+00	1.50E+00
Eu-152	1.17E+01	6.64E+00	3.59E+00	2.36E+00	1.25E+00	1.20E+00

Table 6-38 Soil Area Factors for ROC 1.0 m Thickness

Radionuclide	Soil Area Factors					
	$1m^2$	$2m^2$	$5m^2$	$10m^2$	$100m^2$	$143\ m^2$
C-14	1.69E+05	7.57E+04	2.49E+04	1.03E+04	4.52E+02	2.72E+02
Co-60	1.11E+01	6.55E+00	3.61E+00	2.39E+00	1.39E+00	1.34E+00
Cs-137	1.95E+01	1.16E+01	6.44E+00	4.28E+00	2.45E+00	2.35E+00
Eu-152	9.70E+00	5.74E+00	3.17E+00	2.11E+00	1.23E+00	1.20E+00

6.21 Less Likely But Plausible Scenario Assessment

6.21.1 Drilling Spoils After Contact with Basement Embedded Pipe

The embedded pipe drilling spoils LLBP scenario assumes that the source term for the drilling spoils scenario described in Section 6.11.1 and Section 6.11.7 is embedded pipe as opposed to concrete. The well installer is assumed to continue drilling through the floor, after initially encountering concrete, down to 21.4 m which is the 25th percentile of the well screen depths in the vicinity of FCS (see Attachment 6.1). While moving through the floor the drill encounters embedded pipe which run from approximately 2 to 16 feet below the floor elevations in the Auxiliary Building basement and 2.5 to 3.5 feet below the Turbine Building floor [4]. The pipe cuttings are brought to the ground surface and mixed with the drilling spoils from the entire 21.4 m borehole. Based on the precedent in the approved LTP for the Zion Nuclear Power Station for the same scenario, the well is assumed to be drilled 30 years after license termination. The half-lives applied are from Table 3.1 of the Users Manual for RESRAD Version 6 [17].

The radionuclide concentrations in the embedded pipe that that would result in 25 mrem/yr to the resident farmer AMCG through the embedded pipe drilling spoils pathway are calculated. The concentration resulting in 25 mrem/yr from embedded pipe LLBP drilling spoils is calculated for each radionuclide and compared to the BFM embedded pipe DCGLs listed in Table 6-21. A qualitative evaluation of worker dose concludes that the worker dose is less than the resident farmer dose due to decreased exposure time and elimination of drinking water and food pathways.

The same mathematical model used in Section 6.11.7 to calculate DCGLs for the drilling spoils compliance scenario (after drilling through concrete) is used for the LLBP scenario of drilling through embedded pipe with three adjustments as described below. The volume of drilling spoils is calculated in Excel using Equation 6-22 [13].

Equation 6-22

$$V_{ds} = \pi(0.0254d/2)^2 \, D_{bh}$$

where:

V_{ds} = volume of the drilling spoils on ground surface (m^3)
π = pi constant
0.0254 = conversion factor m/in
d = diameter of borehole (in) = 8 in
D_{bh} = depth of borehole (m) = 21.4 m

The drilling spoils volume (V_{ds} from Equation 6-22) is divided by the 0.15 m spread depth to calculate the area of contaminated zone [13]. The length parallel to aquifer flow is the square root of the contaminate zone area [13]. See Table 6-39. All other RESRAD parameters used to calculate drilling spoils DSRs in Section 6.11.7 are used to calculate DSRs for the LLBP embedded pipe drilling spoils scenario without change.

Table 6-39 Changes to the BFM Drilling Spoils Deterministic Parameter Set for Embedded Pipe LLBP Drilling Spoils DSR Calculation

Parameter	LLBP Embedded Pipe Drilling Spoils Scenario
Area of contaminated zone (m^2)	4.63
Length parallel to aquifer flow (m)	2.15

A unit concentration of 1 pCi/g is assumed to be contained in the drilling spoils. The concentration (pCi/m^2) that is required to be on the internal surface of the embedded pipe to produce an average concentration of 1 pCi/g in the drilling spoils is calculated in Excel using Equation 6-23 [13].

The actual mass of the drilling spoils is the weighted average of the fill/soil density (1.5 g/cm^3) and basement foundation concrete density (2.2 g/cm^2) based on the thickness of each material that the drill traverses. The unit concentration (pCi/m^2 per pCi/g), and the corresponding concentration in the pipe (pCi/m^2) decreases with decreasing density. Therefore, to simplify the calculation and ensure conservatism, the concrete density is ignored and the fill density (1.5 g/cm^3) is applied to both the Auxiliary and Turbine Building basements.

The embedded pipe internal surface area contacted by the drill (denominator in Equation 6-23, i.e., A_{ep}) depends on the internal diameter of the pipe. In the Auxiliary Building basement there are two embedded pipe diameters: 4.03 in and 18.81 in [4]. The internal surface area of the 4.03 in pipe that is contacted by the drill is calculated by determining the pipe circumference and multiplying by the drill borehole diameter (8 in). The use of borehole diameter as the length of pipe contacted is conservative because the curvature of the borehole is ignored. The surface area of the 18.81 in diameter pipe contacted by the drill is calculated by ignoring the curvature and assuming the drill cuts through a top and bottom piece equal to the circular area of the 8 in diameter borehole. The Turbine Building basement contains embedded pipes with internal diameters ranging from 3.51 in to 9.4 in [4]. The 9.4 in diameter pipe is treated the same way as the 4.03 in pipe to simplify the calculation, i.e., assuming that the entire pipe diameter is captured by the drill. This conservative calculation bounds the surface areas of the smaller pipes in the Turbine basement.

Equation 6-23

$$C_{ep,u} = \frac{V_{ds}\ 1x10^{6}\rho_{f/s}C_{ds,u}}{A_{ep}}$$

where:

$C_{ep,u}$ = embedded pipe concentration (pCi/m^2) required to produce a unit concentration of 1 pCi/g concentration in drilling spoils (pCi/m^2 per pCi/g)
V_{ds} = volume of drilling spoils calculated by Equation 6-22 (m^3)
$1x10^6$ = conversion factor cm^3/m^3
$\rho_{f/s}$ = density of fill/soil (g/cm^3)
$C_{ds,u}$ = unit concentration in drilling spoils (1 pCi/g)
A_{ep} = internal surface area of embedded pipe contacted by drill (m^2)

and:

$$A_{ep,4.03} = \pi 4.03d(6.45x10^{-4})$$

$$A_{ep,18.81} = 2\pi(d/2)^2 6.45x10^{-4}$$

$$A_{ep,9.4} = \pi 9.4d(6.45x10^{-4})$$

where:

$A_{ep,4.03}$ = internal surface area of 4.03 inch internal diameter pipe that is contacted by 8 in diameter drill
d = diameter of drill borehole = 8 in
$6.45x10^{-4}$ = conversion factor m^2/in^2
$A_{ep,18.81}$ = internal surface area of 18.81inch internal diameter pipe that is contacted by 8 inch diameter drill
$A_{ep,9.4}$ = internal surface area of 9.4 inch internal diameter pipe that is contacted by 8 inch diameter drill

The concentrations on the internal surface of embedded pipe that results in 25 mrem/yr through the LLBP embedded pipe drilling spoils scenario are calculated in Excel using Equation 6-24. See Table 6-40.

Equation 6-24

$$C_{ep,i} = \frac{25\ C_{ep,u}}{DSR_{ep,i}}$$

where:

$C_{ep,i}$ = concentration in embedded pipe that results in 25 mrem/yr through the LLBP drilling spoils scenario for radionuclide i (pCi/m2)
$C_{ep,u}$ = embedded pipe concentration required to produce 1 pCi/g average concentration in drilling spoils (Equation 6-23) (pCi/m2 per pCi/g)
$DSR_{ep,i}$ = dose to source ratio for LLBP embedded pipe drilling spoils for radionuclide i (mrem/yr per pCi/g)
25 = 25 mrem/yr dose criterion

All of the initial suite radionuclide LLBP embedded pipe drilling spoils scenario 25 mrem/yr concentrations are less than the embedded pipe DCGLs listed in Table 6-21 except Cs-137, Eu-152, and Eu-154. The ratios of the Auxiliary Building basement concentrations to the embedded pipe DCGLs in Table 6-21 are 4.08, 37.54, and 12.32, respectively [13]. The ratios for the Turbine Building basement are 5.10, 46.94, and 15.40, respectively [13]. A ratio greater than one indicates that the dose from the LLBP embedded pipe drilling spoils scenario will exceed 25 mrem/yr for Cs-137, Eu-152, and Eu-154 when the concentrations are equal to the embedded pipe DCGLs in Table 6-21. The dose is the ratio multiplied by 25 mrem/yr. The dose from LLBP embedded pipe drilling spoils will be evaluated during the FSS of embedded pipe to ensure that the dose, if it exceeds 25 mrem/yr, is not significant. See LTP Chapter 5, for the process that will be used during FSS to evaluate the LLBP embedded pipe drilling spoils dose.

6.21.2 Offsite Processing/Recycling of Excavated Basement Concrete and Containment Steel Liner for ROC

The BFM excavation scenario evaluated in Section 6.11 assumes that the basement concrete walls are excavated, size reduced, and used as onsite fill. The activity on the Containment steel liner is released to onsite soil after excavation. The AMCG for the BFM excavation scenario is a resident farmer. The environmental pathways and exposure pathways applicable to the resident farmer scenario (see Table 6-2) are assumed to represent a bounding set of pathways. The BFM excavation scenario assumes that a somewhat limited basement excavation occurs. However, it is possible that a large-scale industrial project may proceed on the site at some time in the future that includes the excavation of the majority of the walls and the liner. There are currently no plans to conduct a large-scale excavation project on the site and a period of 30 years is assumed to elapse before such a project is undertaken.

The large volumes of excavated concrete and steel are assumed to be recycled or disposed of offsite. As discussed in Section 6.4.4, there are multiple worker exposure scenarios in the recycle/disposal scenario which includes activities such as dismantling and handling while decommissioning, exposure to waste piles, transport to the recycle or disposal site, and disposal at a landfill. There is also the potential for public exposure from the commercial reuse of the materials. Offsite recycle and disposal scenarios are evaluated in NUREG-1640 [5]. Table 2.1 of [5] provides dose factors for concrete and steel in units of Sv/yr per Bq/cm^2. The dose factors are converted to concentrations (pCi/m^2) that result in a dose of 25 mrem/yr in Excel using Equation 6-25 [13]. The concentrations are provided in Table 6-41.

Table 6-40 LLBP Embedded Pipe Drilling Spoils Concentrations That Result in 25 mrem/yr

Radionuclide	Auxiliary EP 4.03 in ID	Auxiliary EP 18.81 in ID	Turbine EP 9.4 in ID
	EP LLBP Drilling Spoils Concentration (pCi/m^2 per 25 mrem/yr)	EP LLBP Drilling Spoils Concentration (pCi/m^2 per 25 mrem/yr)	EP LLBP Drilling Spoils Concentration (pCi/m^2 per 25 mrem/yr)
Am-241	4.066E+10	4.10E+10	1.743E+10
C-14	4.493E+13	4.526E+13	1.926E+13
Ce-144	6.470E+21	6.518E+21	2.774E+21
Cm-243	1.020E+10	1.028E+10	4.374E+09
Cm-244	6.995E+11	7.047E+11	2.999E+11
Co-58	2.604E+56	2.623E+56	1.116E+56
Co-60	1.290E+10	1.299E+10	5.529E+09
Cs-134	1.035E+13	1.043E+13	4.438E+12
Cs-137	2.052E+09	2.067E+09	8.795E+08
Eu-152	2.527E+09	2.546E+09	1.083E+09
Eu-154	5.294E+09	5.333E+09	2.269E+09
Eu-155	1.014E+12	1.021E+12	4.345E+11
Fe-55	3.335E+18	3.360E+18	1.430E+18
H-3	3.699E+14	3.727E+14	1.586E+14
Ni-59	2.092E+14	2.107E+14	8.968E+13
Ni-63	9.495E+13	9.566E+13	4.071E+13
Np-237	2.541E+09	2.560E+09	1.089E+09
Pu-238	1.758E+11	1.771E+11	7.537E+10
Pu-239	1.257E+11	1.266E+11	5.389E+10
Pu-240	1.266E+11	1.275E+11	5.427E+10
Pu-241	6.279E+12	6.326E+12	2.692E+12
Sb-125	2.834E+12	2.856E+12	1.215E+12
Sr-90	8.305E+10	8.367E+10	3.561E+10
Tc-99	5.105E+11	5.143E+11	2.189E+11

Equation 6-25

$$C_{r,m,i} = \frac{25(1x10^4)}{(DF_{m,i}0.0037e^{(-\lambda_i 30)})}$$

where:

$C_{r,m,i}$ = concentration in material *m* (concrete or steel) for radionuclide *i* that results in a recycle/disposal dose of 25 mrem/yr (pCi/m^2)
25 = 25 mrem/yr dose criterion
$1x10^4$ = conversion factor (cm^2/m^2)
$DF_{m,i}$ = NUREG-1640, Table 2,1 dose factor for material *m* and radionuclide *i* (Sv/yr per Bq/cm^2)
0.0037 = conversion factor (mrem/yr per pCi/cm^2 per Sv/yr per Bq/cm^2)
λ_i = radioactive decay constant for radionuclide *i* (0.693/$t_{1/2}$)
30 = years after license termination that large-scale excavation occurs (yr)

Table 6-41 LLBP Offsite Recycle/Disposal Concentrations That Result in 25 mrem/yr

Radionuclide	Concrete	Steel
	Recycle/Disposal LLBP Concentration (pCi/m^2 per 25 mrem/yr)	Recycle/Disposal LLBP Concentration (pCi/m^2 per 25 mrem/yr)
Am-241	3.22E+10	7.54E+06
C-14	1.654E+11	1.0 76E+10
Ce-144	1.414E+21	1.343E+20
Cm-243	1.274E+09	2.123E+07
Cm-244	3.328E+09	4.095E+07
Co-58	3.007E+55	2.460E+54
Co-60	3.489E+09	3.489E+08
Cs-134	2.787E+12	1.879E+11
Cs-137	6.142E+08	4.222E+07
Eu-152	7.305E+08	7.475E+07
Eu-154	1.464E+09	1.668E+08
Eu-155	4.062E+11	5.077E+10
Fe-55	1.963E+16	1.640E+15
H-3	1.516E+12	9.573E+10
Ni-59	8.893E+12	7.427E+11
Ni-63	1.134E+13	8.926E+11
Np-237	1.778E+07	1.090E+06
Pu-238	8.563E+08	1.404E+07
Pu-239	6.148E+08	1.025E+07
Pu-240	6.162E+08	1.027E+07
Pu-241	1.301E+11	2.602E+09
Sb-125	8.189E+11	8.773E+10
Sr-90	2.653E+10	1.150E+09
Tc-99	3.072E+09	2.112E+08

The resulting concentrations ($C_{r,m,i}$) are compared to the BFM wall/floor DCGLs in Table 6-18. All of the $C_{r,m,i}$ recycle/disposal concentrations are greater than the BFM wall/floor DCGLs [13]. Therefore, the dose from the LLBP recycle/reuse scenario will be less than the dose calculated for the BFM wall/floor compliance scenario.

6.22 Compliance Dose Summation

The final dose used to demonstrate compliance with the 25 mrem/yr criterion is the summation of the dose from all FCS site media including:

- soil,
- basement walls and floors,
- basement embedded pipe,
- basement fill,
- buried pipe,
- above ground buildings, and
- existing groundwater.

The compliance dose calculation is generally described by Equation 6-26. The specific details of the dose calculation for each of the terms in Equation 6-26 are provided in Chapter 5 of this LTP, Section 5.2.6.8.

Equation 6-26

$$D_c = D_s + D_{b,wf} + D_{b,ep} + D_f + D_{bp} + D_{agb} + D_{egw}$$

where:

D_c = compliance dose (mem/yr)
D_s = dose from soil (mrem/yr)
$D_{b,wf}$ = dose from basement walls and floors (mrem/yr)
$D_{b,ep}$ = dose from basement embedded pipe (mrem/yr)
D_f = dose from basement fill material (mrem/yr)
D_{bp} = dose from buried pipe (mrem/yr)
D_{agb} = dose from above ground buildings (mrem/yr)
D_{egw} = dose from existing groundwater (mrem/yr)

6.23 References

[1] Haley & Aldrich, "Hydrogeological Conceptual Site Model, Rev. 4, Fort Calhoun Station, Blair, NE," 2021.

[2] Radiation Safety and Control Services, "TSD 20-001, Historical Site Assessment for Fort Calhoun Station," 2020.

[3] Omaha Public Power District, Fort Calhoun Station Unit 1, “Defueled Safety Analysis Report”.

[4] Omaha Public Power District, "FC-21-002, Description of Embedded Piping, Penetrations, and Buried Pipe to Remain in Fort Calhoun End State".

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[6] Omaha Public Power District, "FC-18-002, Potential Radionuclides of Concern During the Decommissioning of Fort Calhoun Nuclear Station".

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[10] Argonne National Laboratory, "Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil and Building Structures, ANL/EVS/TM-14-4," 2015.

[11] U.S. Nuclear Regulatory Commission, "NUREG/CR-5512, Volume 3, Residual Radioactive Contamination From Decommissioning, Parametrer Analysis," 1999.

[12] Omaha Public Power District, "FC-20-006, Fort Calhoun Station End State Building Concrete Surface Areas and Volumes".

[13] Omaha Public Power District, "Excel Spreadsheet Calculations in Support of FCS LTP Chapter 6, Revision 0".

[14] U.S. Nuclear Regulatory Commission, "E-Mail from Jack Parrot, NRC, to Corey Cameron, OPPD," January 25, 2021.

[15] Brookhaven National Laboratory, "Recommended Values for the Distribution Coefficient (Kd) to be Used in Dose Assessments for Decommissioning the Zion Nuclear Power Plant, BNL-105442-2014-IR," 2014.

[16] U.S. Nuclear Regulatory Commission, "NUREG-1757, Volume 2, Revision 2, Consolodated Decommissioning Guidance, Charcaterization, Survey and Determination of Radiological Criteria - Draft Report for Comment," 2020.

[17] Argonne National Laboratory, "Users Manual for RESRAD Version 6," 2001.

[18] U.S. Nuclear Regulatory Commission, "NUREG/CR-6697, Development of Probabilistic RESRSAD 6.0 and RES-BUILD 3.0 Computer Codes," 2000.

[19] e. a. Napier, "Hanford Environmental Dosimetry Upgrade Project, GEN II The Hanford Environmental Radiation Dosimetry Software System, Volume 1 Conceptual Representation," 1988.

Figure 6-1 FCS Site Location

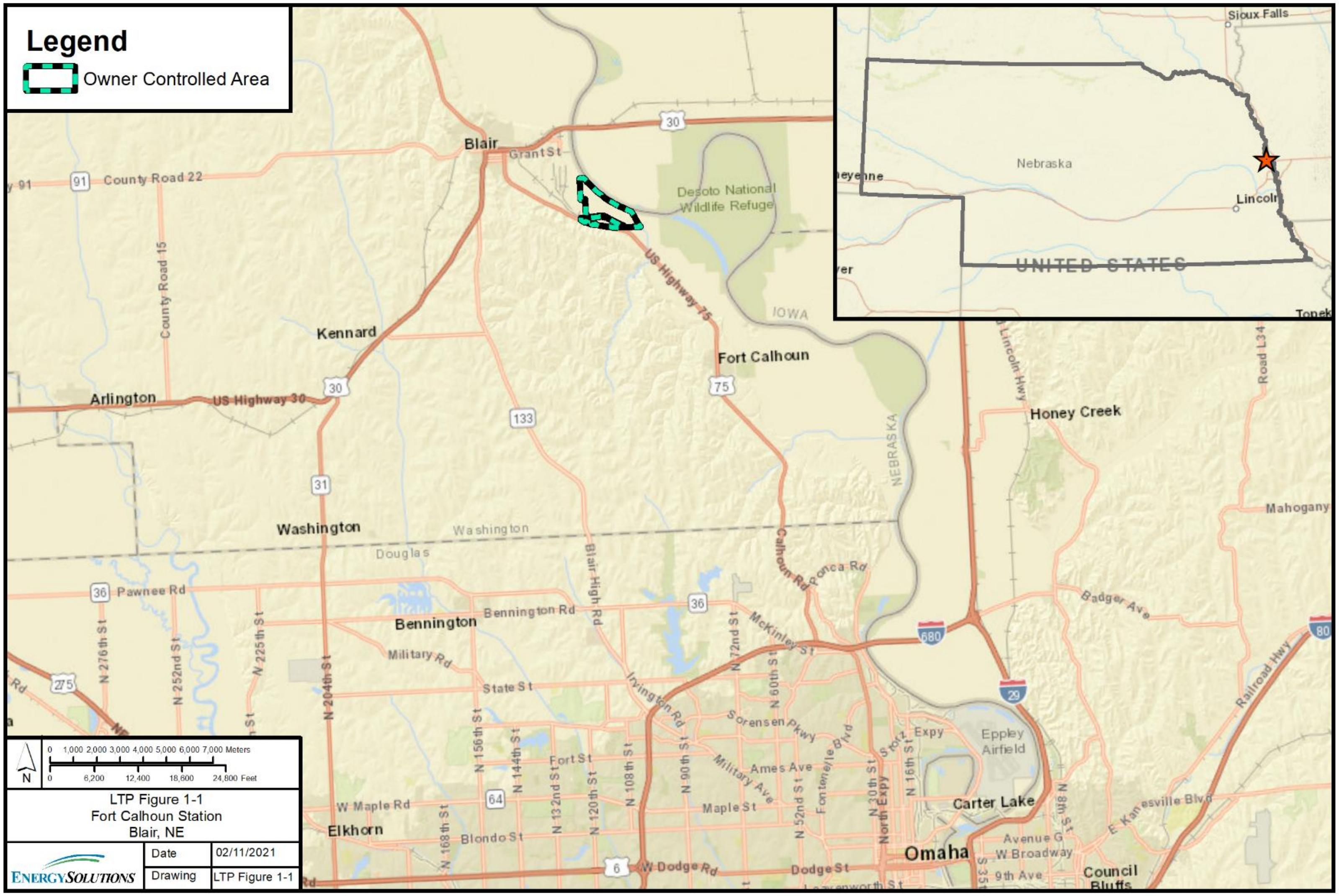


Figure 6-2 FCS Owner Controlled Areas

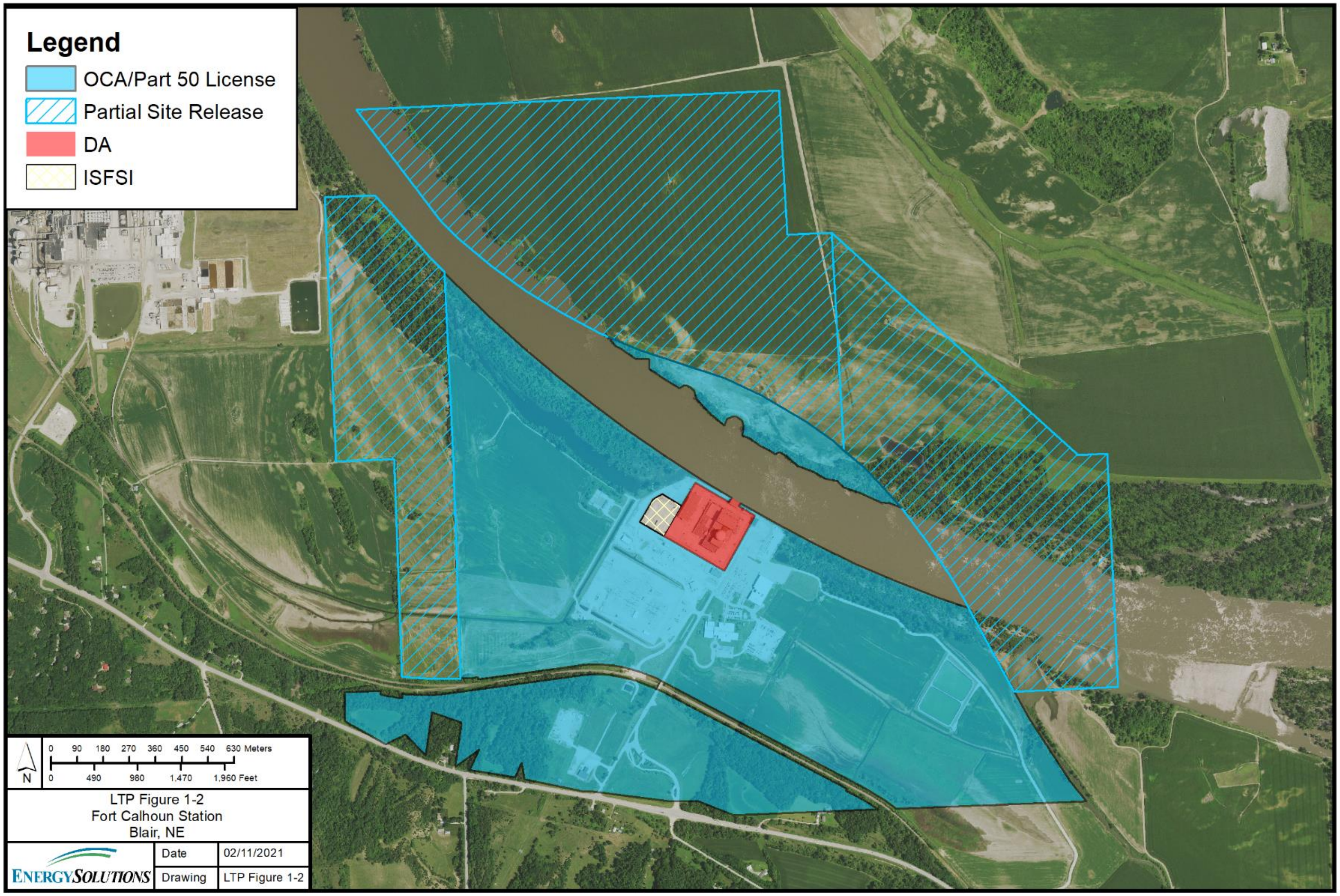


Figure 6-3 FCS Site Layout

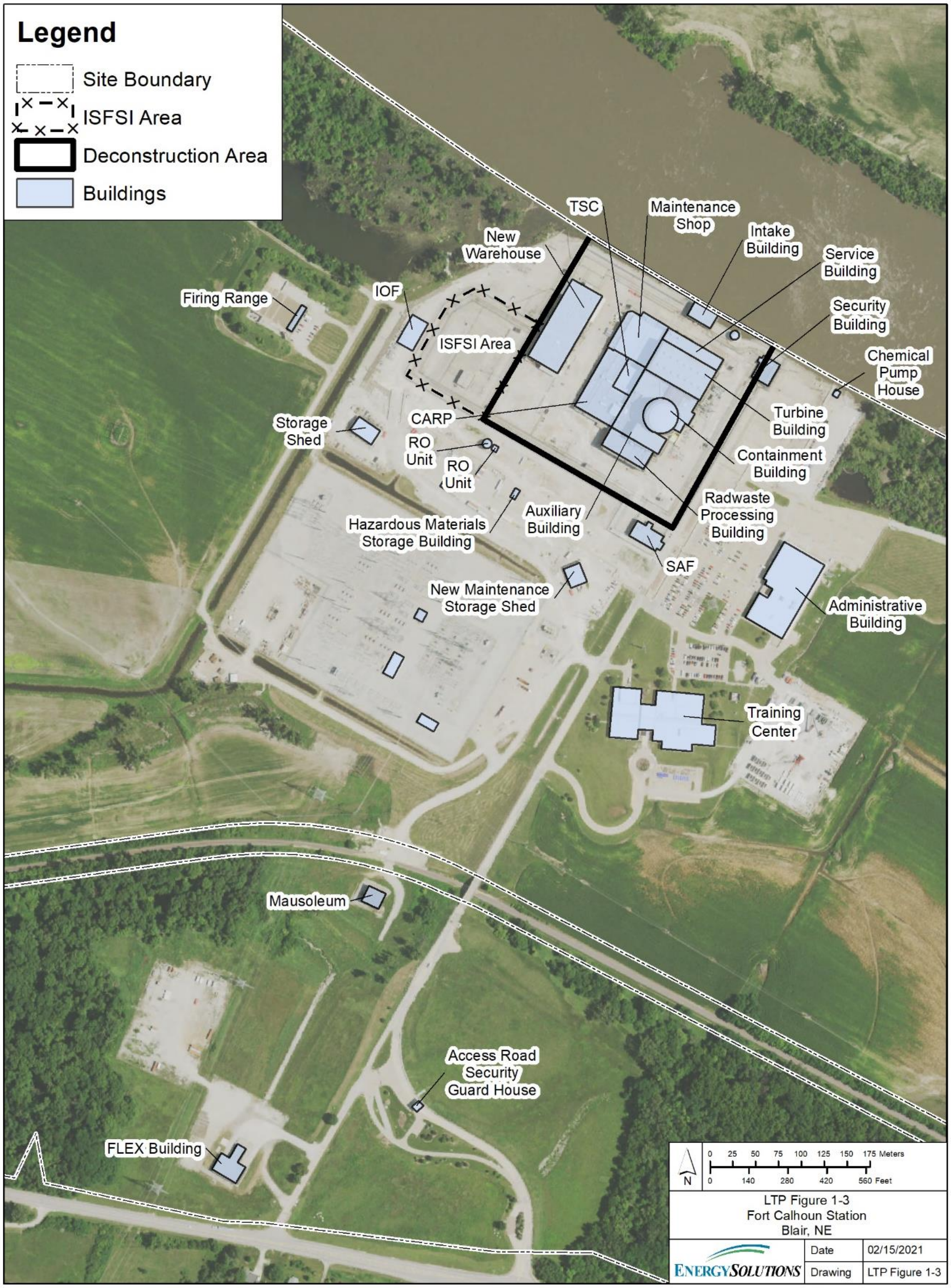


Figure 6-4 FCS Site Topography

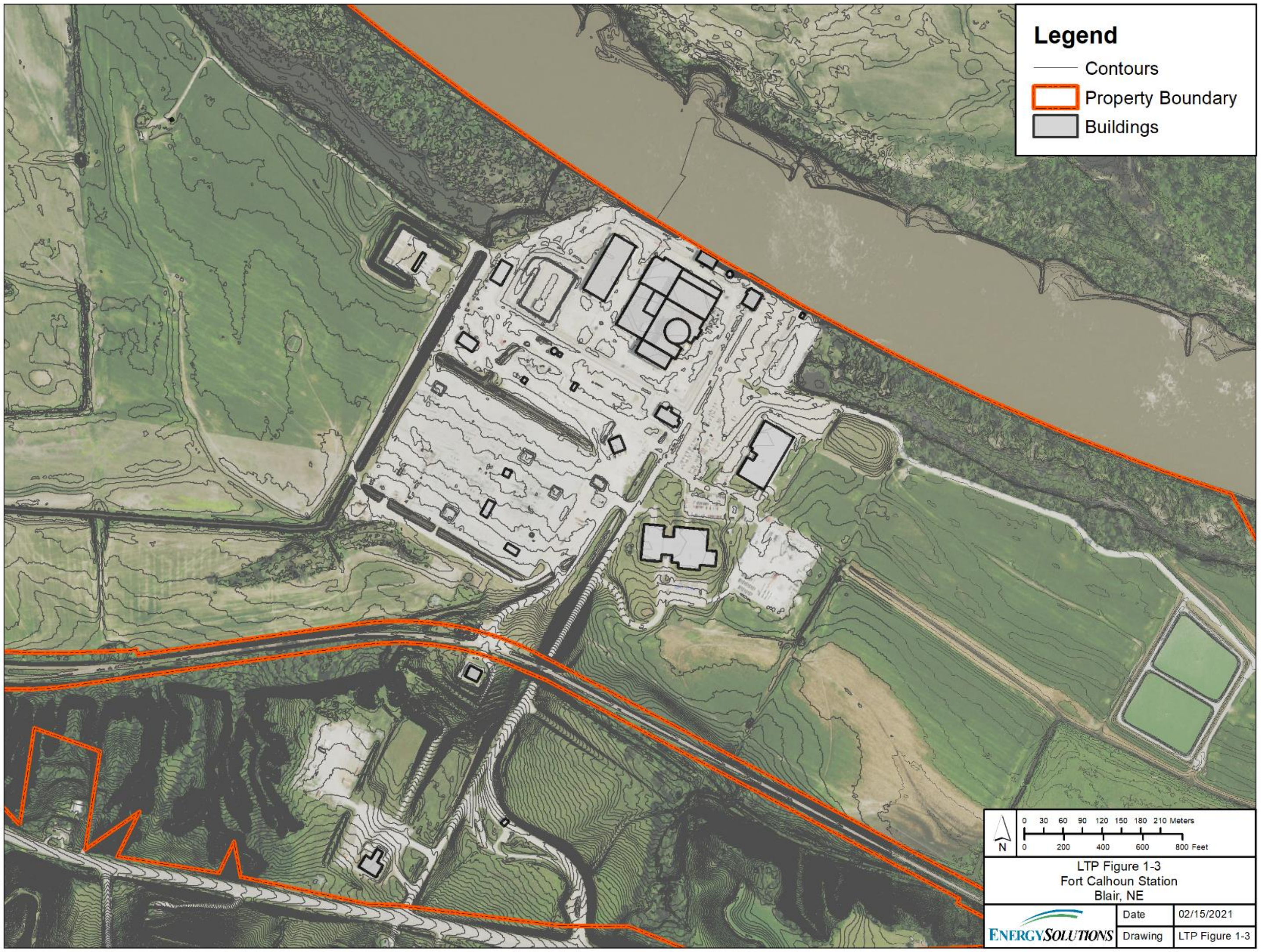


Figure 6-5 Shallow Groundwater Aquifer Contours

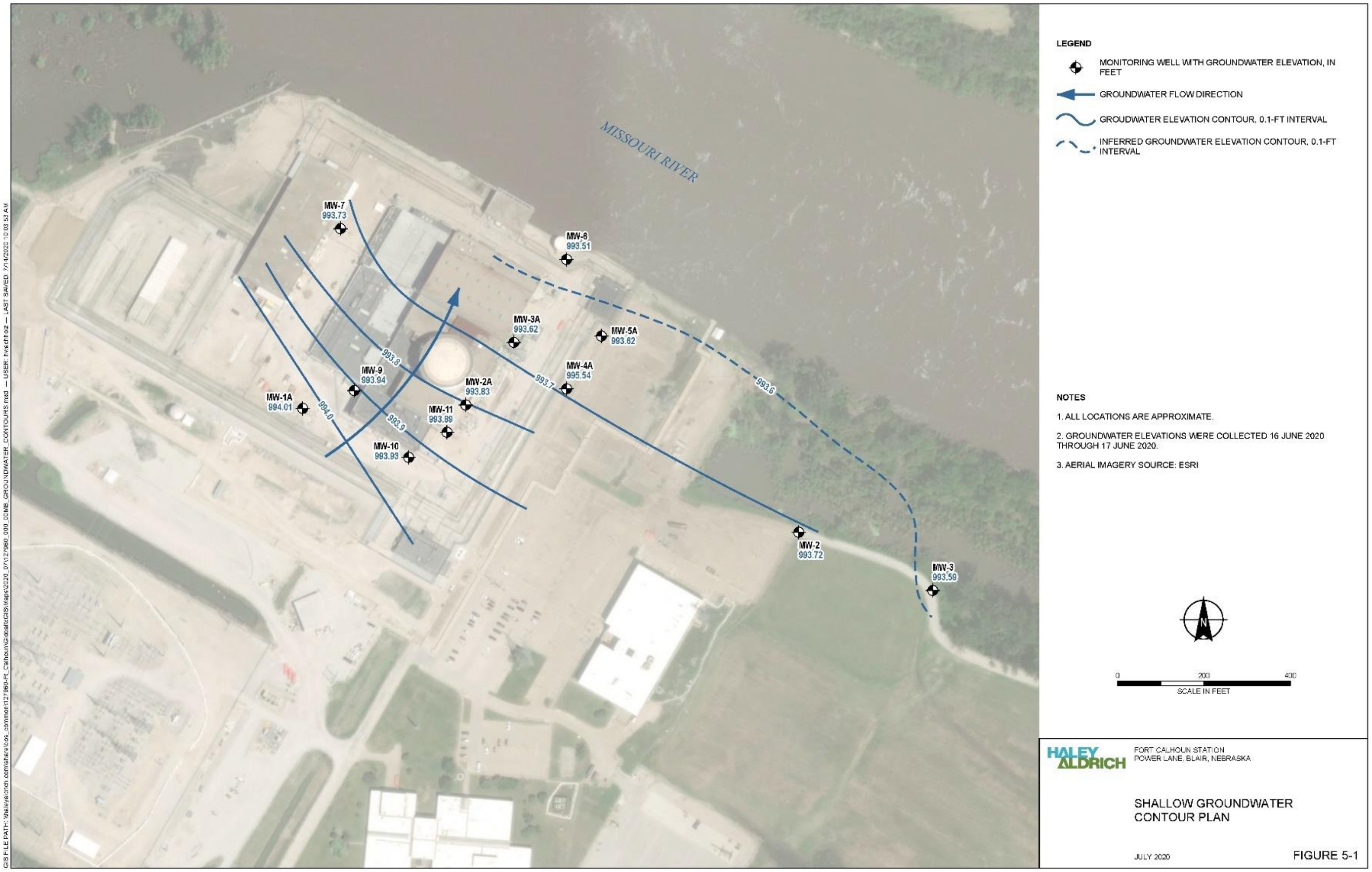


Figure 6-6 Deep Aquifer Groundwater Contours

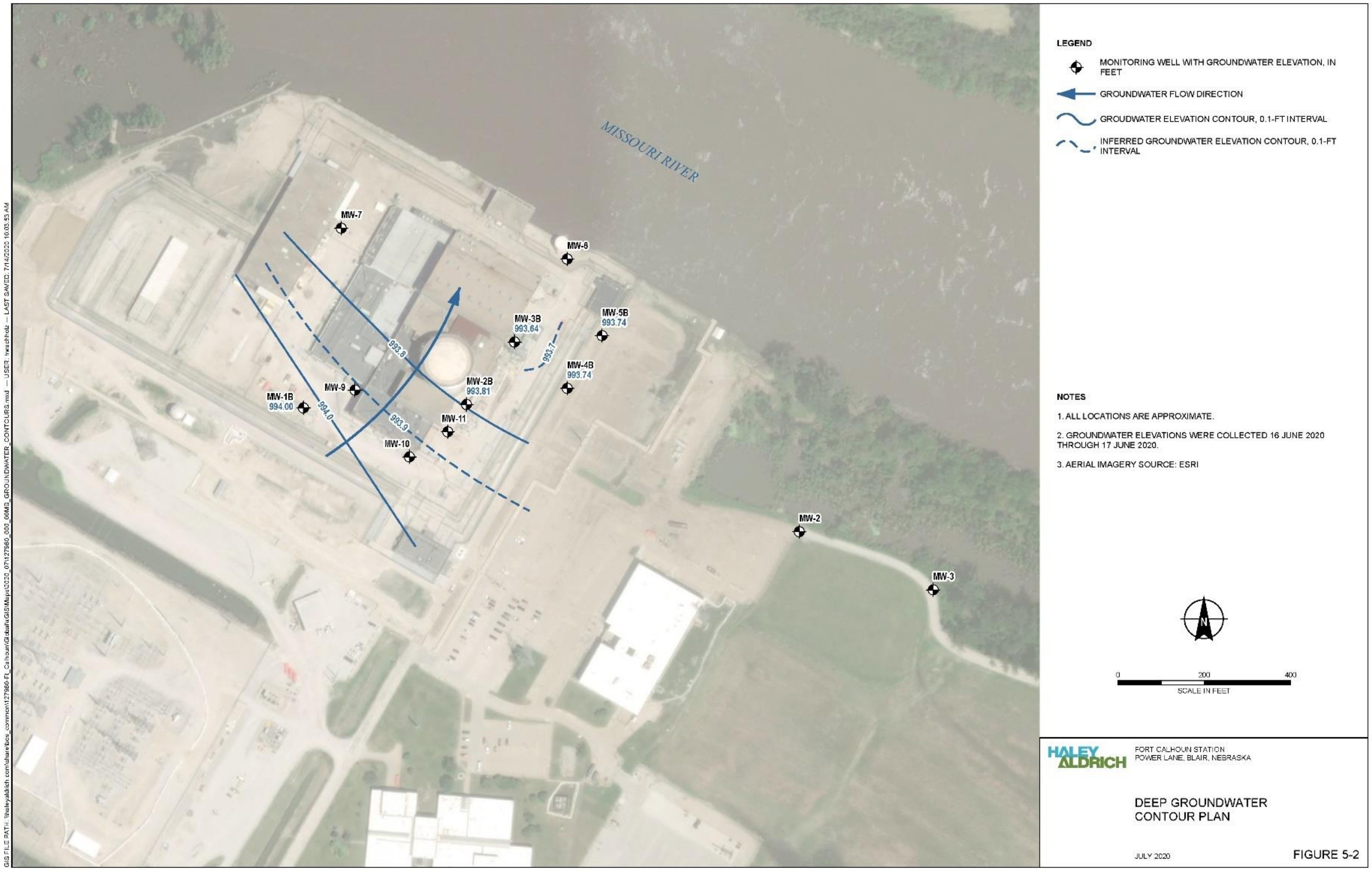


Figure 6-7 General Basement Plan View at 1001′ Demolition Elevation (Not Final End State Configuration)

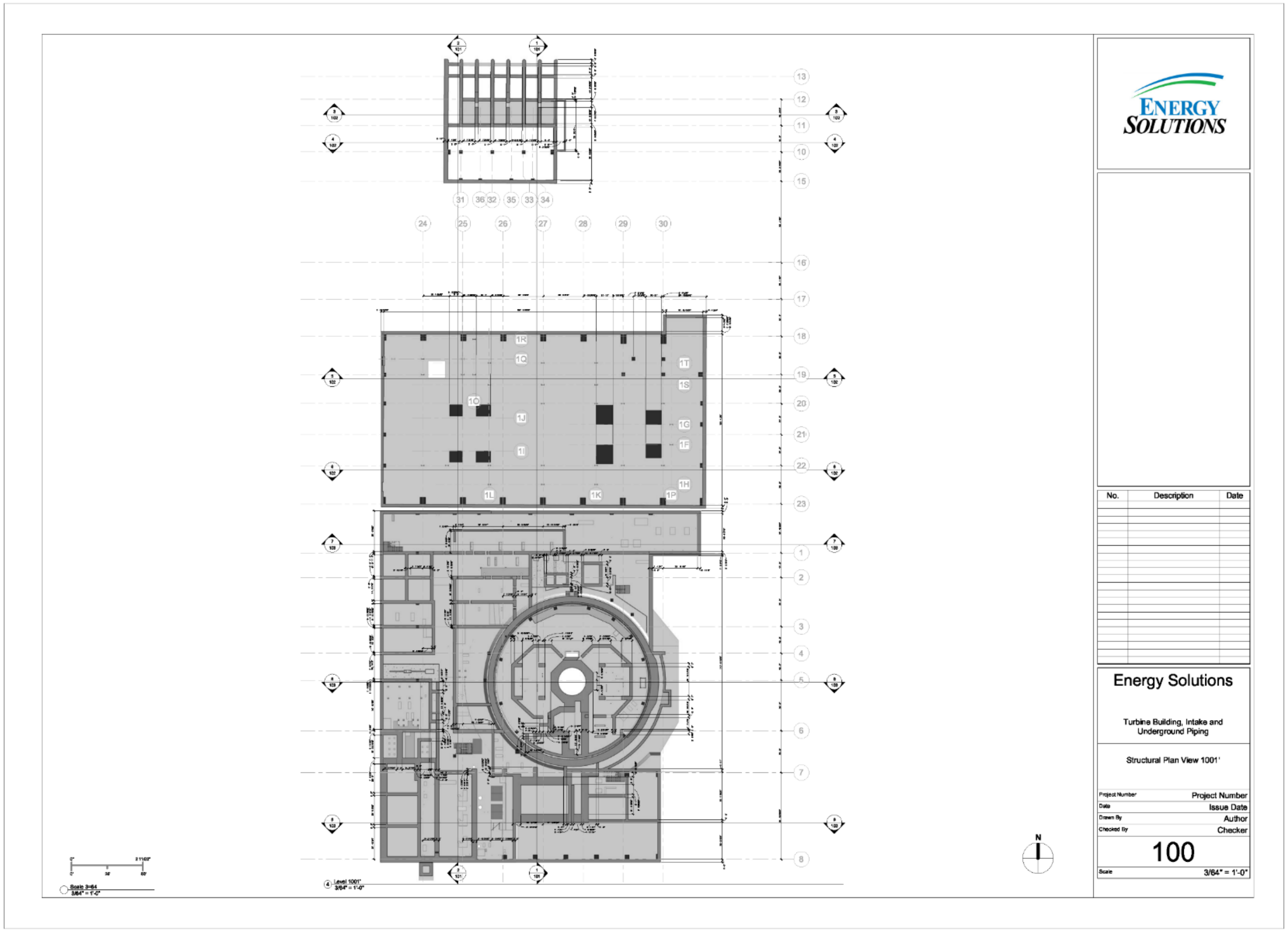


Figure 6-8 General FCS Plant Cross Sections (Not Final End State Configuration)

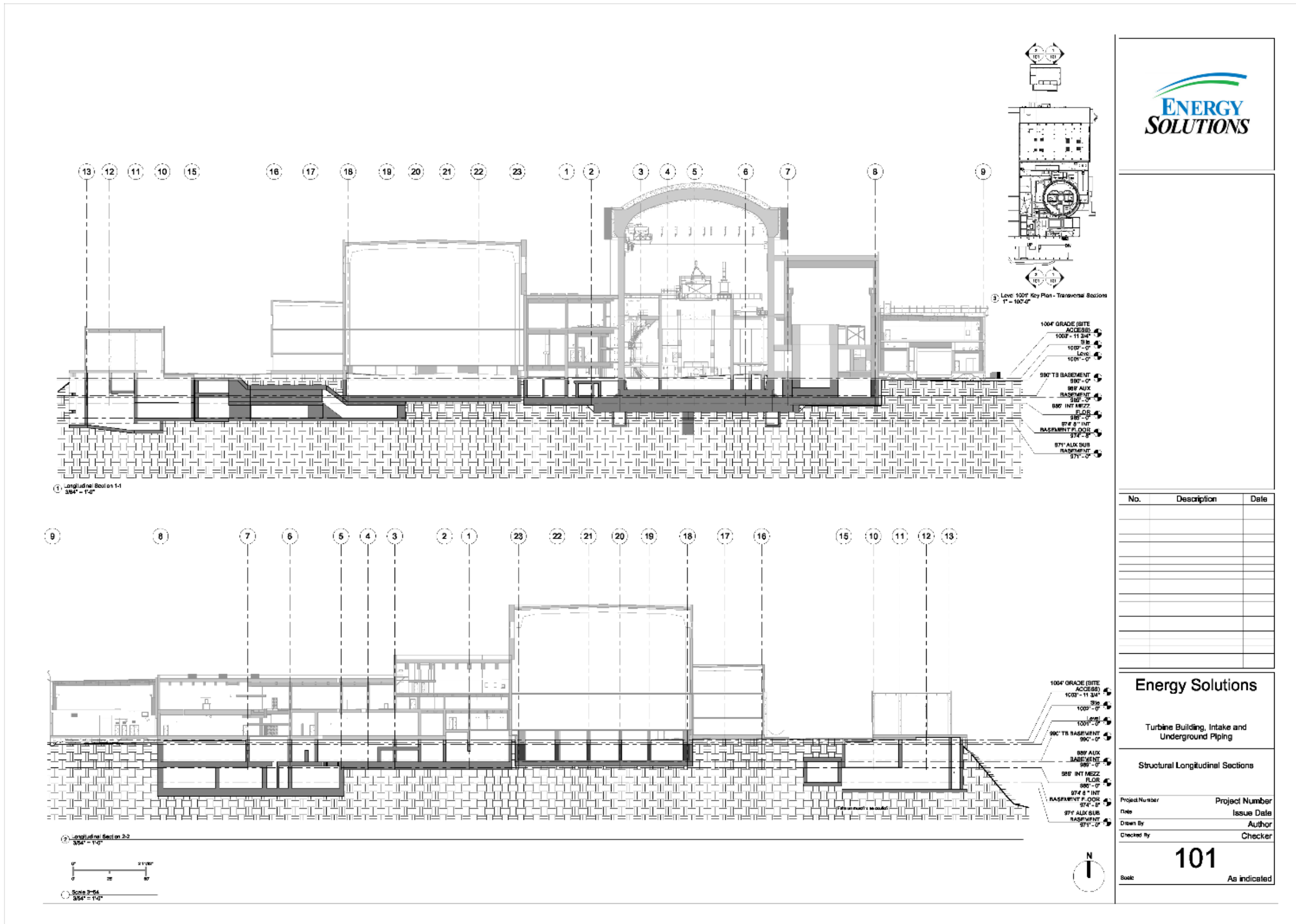


Figure 6-9 General FCS Plant Cross Sections (Not Final End State Configuration)

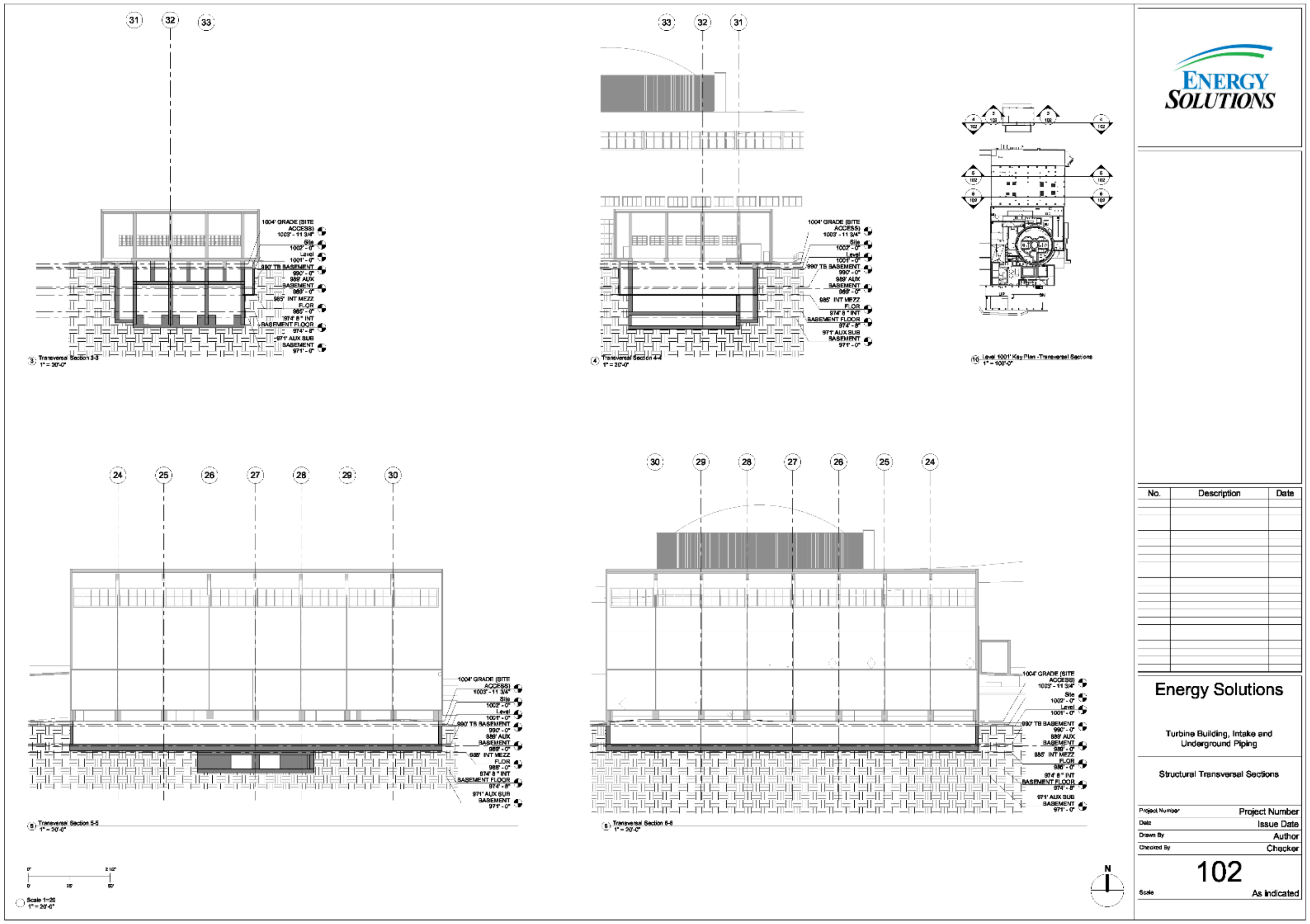


Figure 6-10 General FCS Plant Cross Sections (Not Final End State Configuration)

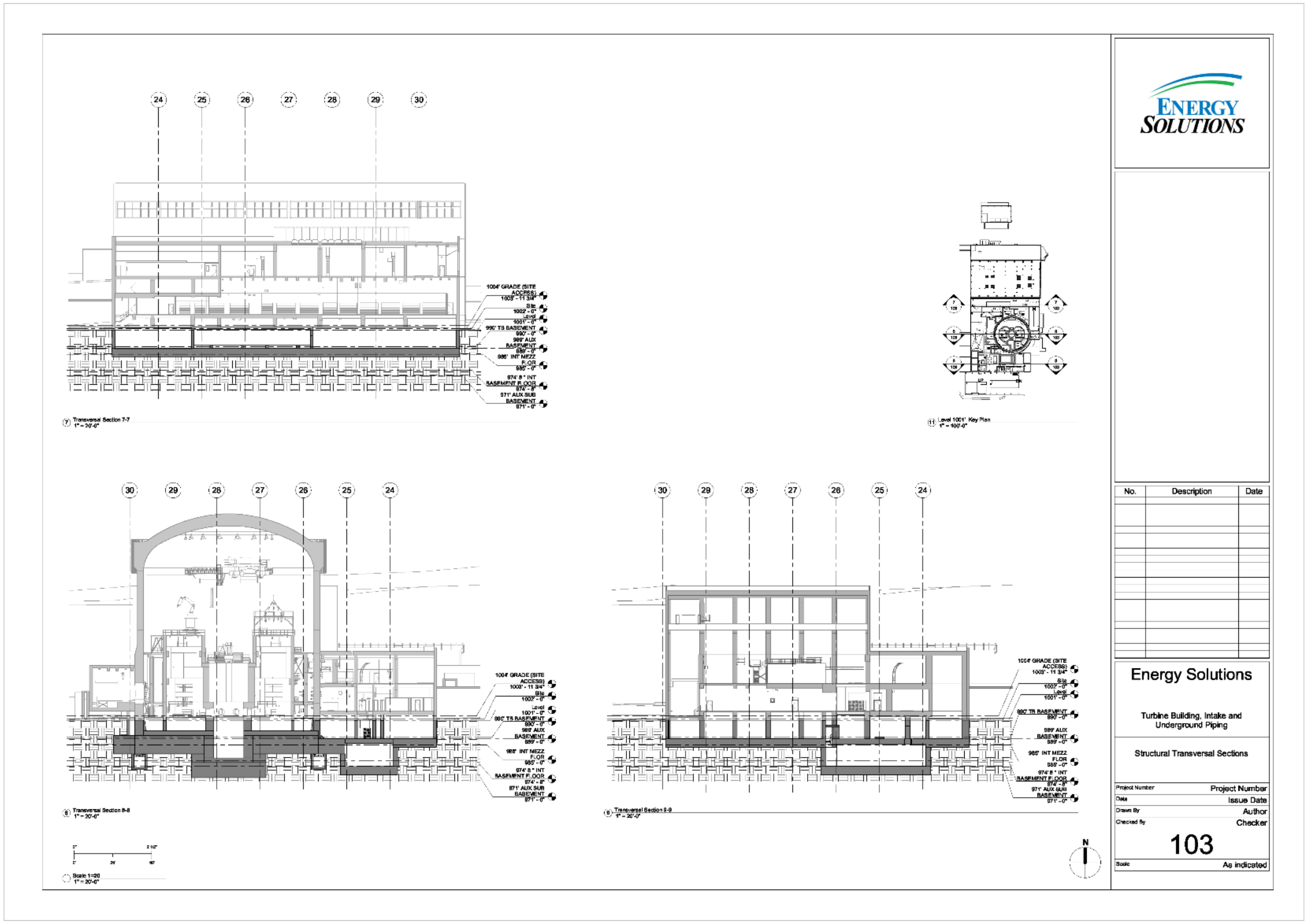


Figure 6-11 Auxiliary Basement End State Floor Elevations and Surface Areas

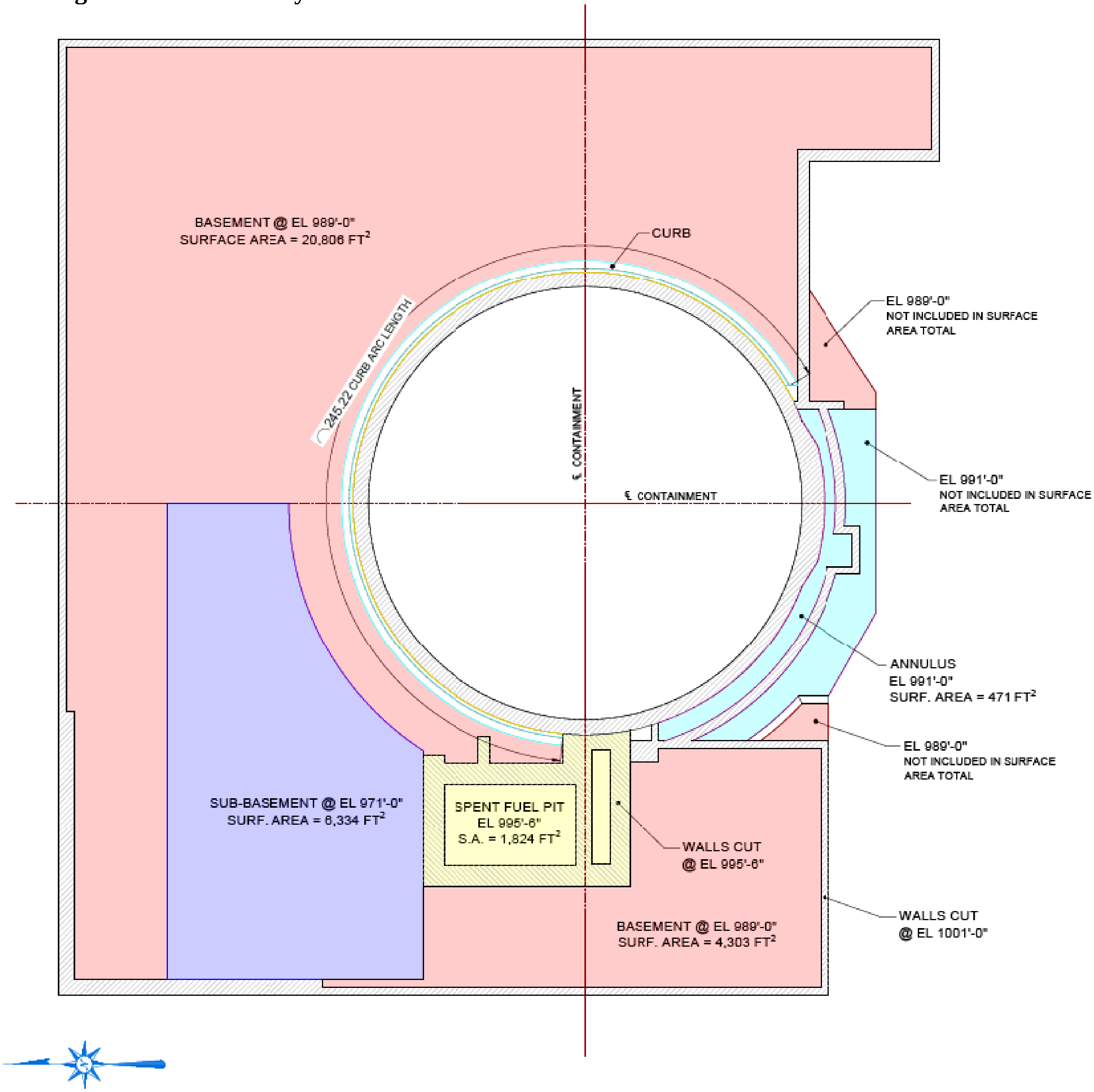


Figure 6-12 Containment End State General Arrangement Elevation View

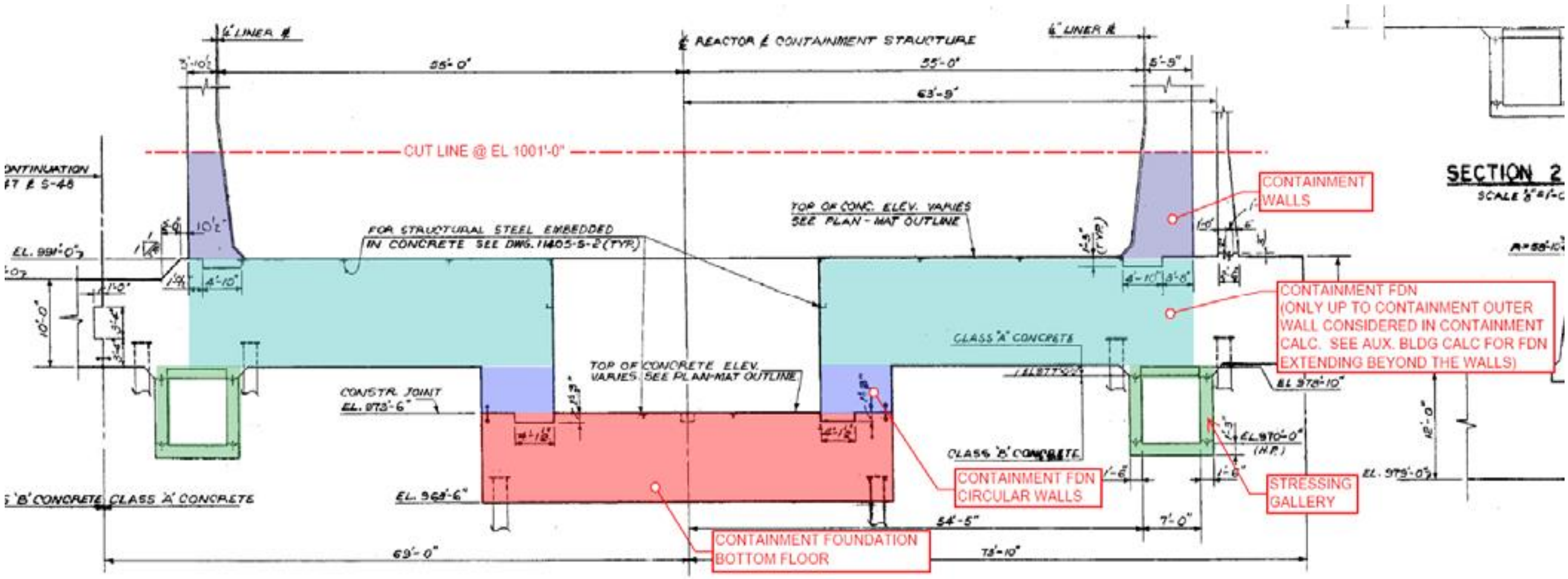


Figure 6-13 Turbine Building End State General Arrangement Plan View

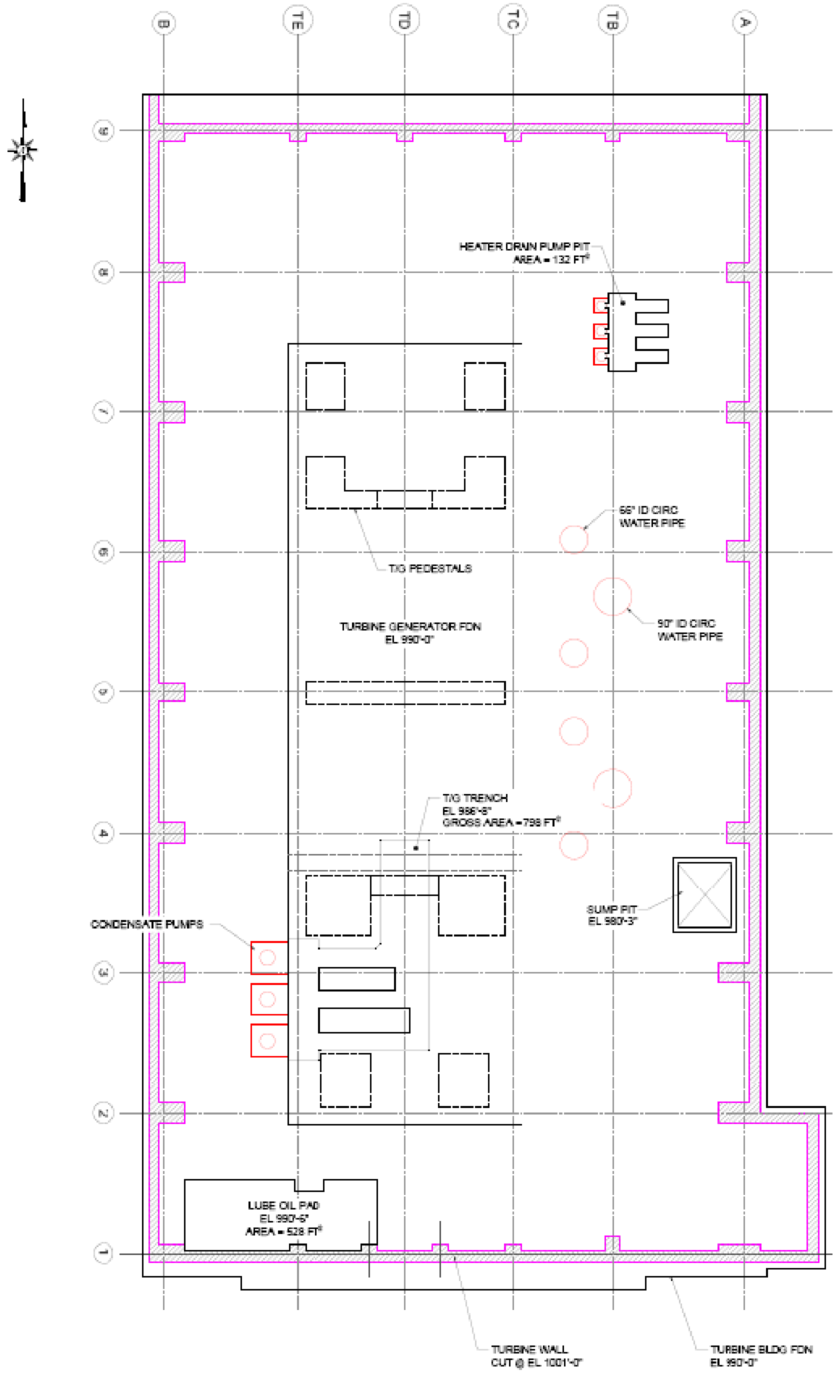


Figure 6-14 Turbine Basement End State Foundation, Wall Areas, and Pedestal Layout

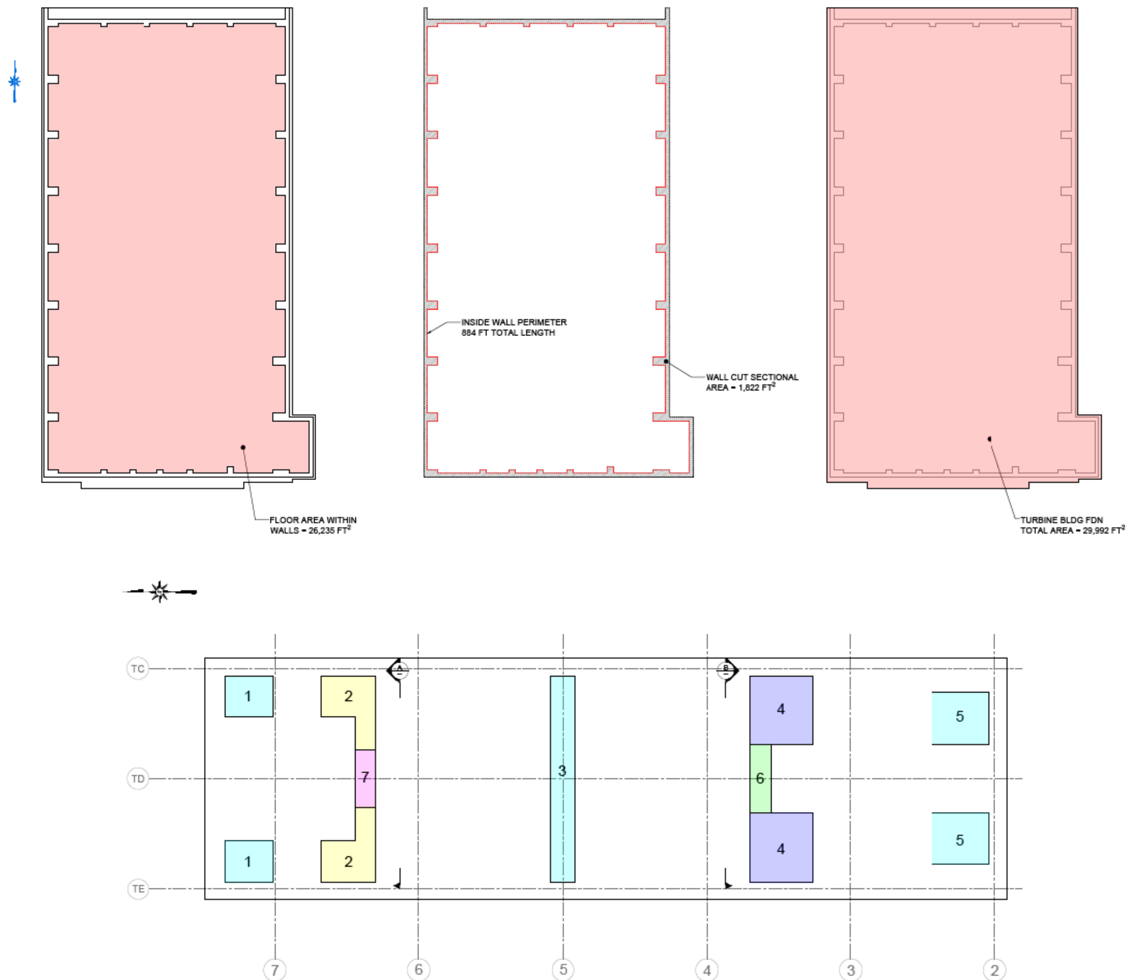


Figure 6-15 Circulating Water Discharge Tunnel End State at Turbine Basement

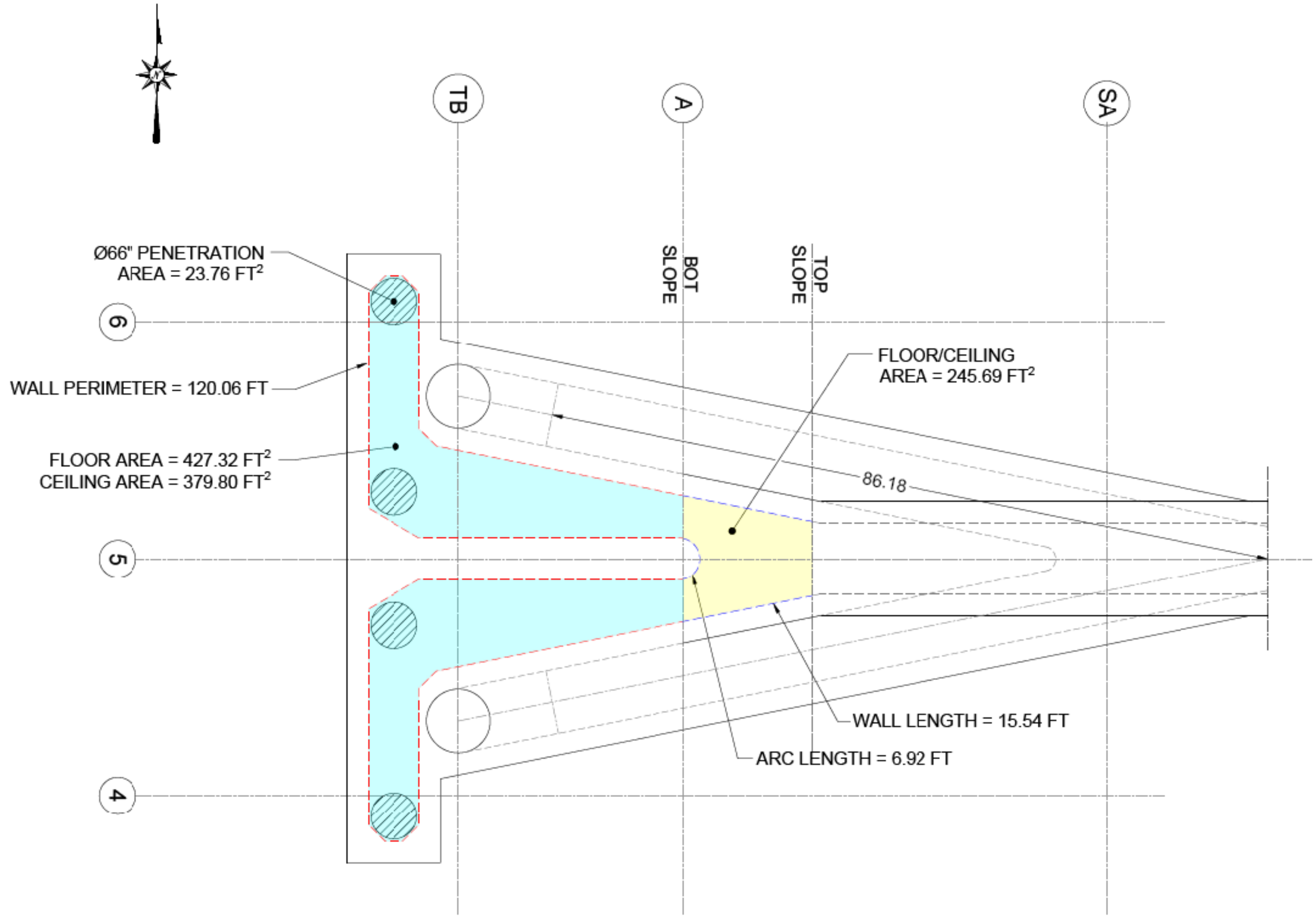


Figure 6-16 Intake Structure End State Elevations and Surface Areas

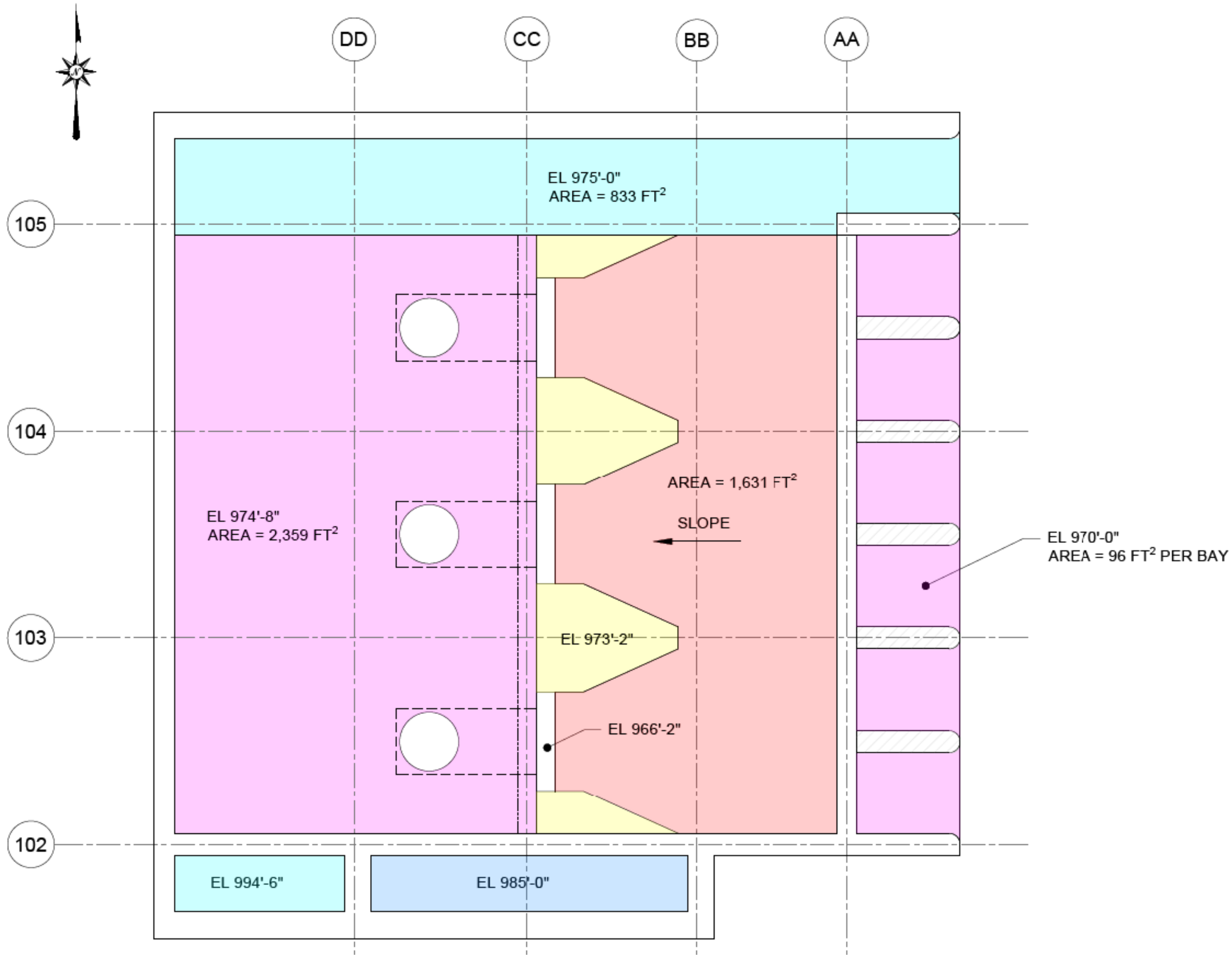


Figure 6-17 RESRAD Parameter Selection Process

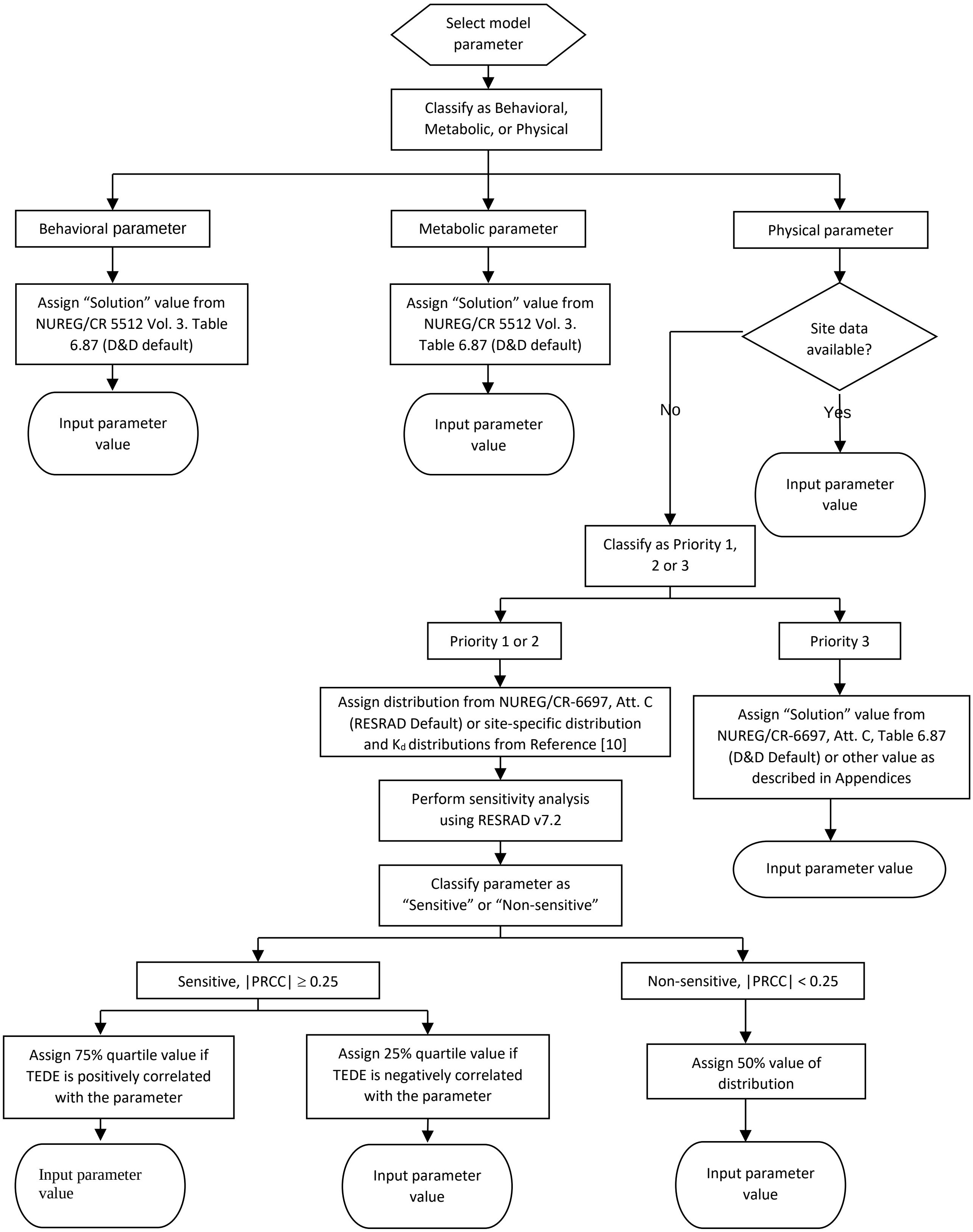
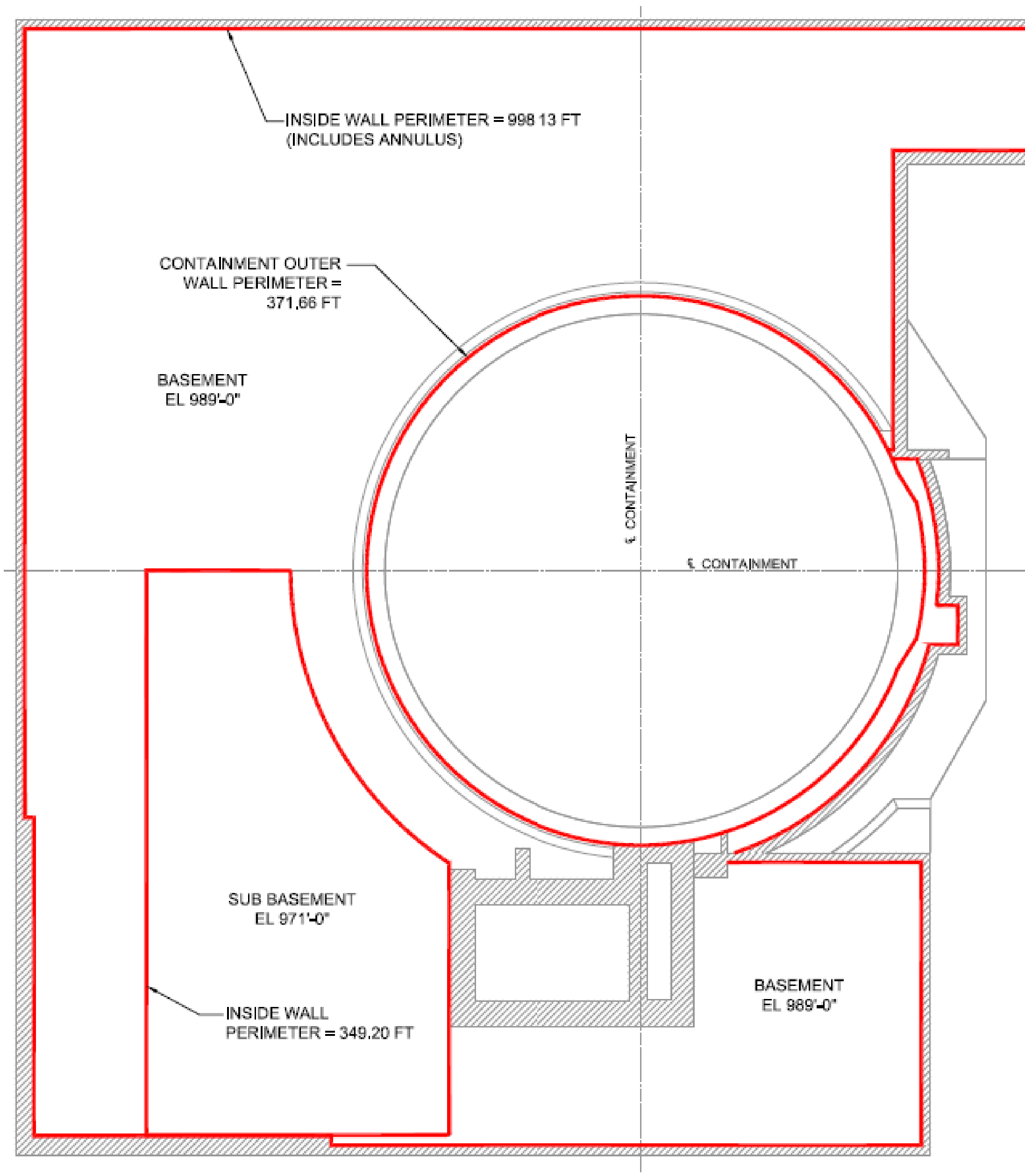


Figure 6-18 Actual Auxiliary Basement *insitu* Source Term Geometry. Red Lines Represent 1 m Thick Source in Fill Adjacent to Walls. Floor Source Term Not Shown



Attachment 6-1 RESRAD Parameters for Soil Uncertainty Analysis

Table A.1.1 RESRAD Parameters for Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Dose Conversion Factor File	NA	NA	NA	FCS FGR11	Plant, Meat, and Milk transfer factors set to the 75th percentile of the RESRAD default PDF				
Basic radiation dose limit (mrem/y)		3	D	25	10 CFR 20.1402	NR	NR	NR	NR
Initial principal radionuclide (pCi/g)	P	2	D	1	Unit Value	NR	NR	NR	NR
Distribution coefficients (contaminated and unsaturated zones) (cm^3/g)									
Am-241	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2 [10]	8.34	1.79	NA	NA
C-14	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[e] [10]	2.4	3.22	NA	NA
Ce-144	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	8.01	1.1	NA	NA
Cm-243/ 244	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	9.85	0.69	NA	NA
Co-58	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	6.7	2.71	NA	NA
Co-60	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	6.7	2.71	NA	NA
Cs-134	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	8.16	1.39	NA	NA
Cs-137	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	8.16	1.39	NA	NA
Eu-152	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[e]	6.72	3.22	NA	NA
Eu-154	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[e]	6.72	3.22	NA	NA
Eu-155	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[e]	6.72	3.22	NA	NA
Fe-55	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	6.79	0.69	NA	NA
H-3	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[e]	-2.81	0.5	NA	NA
Ni-59	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	5.19	1.61	NA	NA
Ni-63	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	5.19	0.69	NA	NA
Np-237	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	3.14	1.39	NA	NA
Pu-238	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	6.86	1.39	NA	NA
Pu-239/ 240	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	6.86	1.39	NA	NA
Pu-241	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	6.86	1.39	NA	NA
Sb-125	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	4.11	1.1	NA	NA
Sr-90	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	4.04	1.61	NA	NA
Tc-99	P	1	S	Lognormal-N	DCH 2015, Table 2.13.2	-2.66	1.1	NA	NA
Daughter Products									

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Actinium	P	NA	D	8.29E+02	Median of Distribution DCH 2015, Table 2.13.10[e] [10]				
Americium	P	NA	D	4.19E+03	Median of Distribution DCH 2015, Table 2.13.2 [10]				
Curium	P	NA	D	1.90E+04	Median of Distribution DCH 2015, Table 2.13.2				
Gadolinium	P	NA	D	8.29E+02	Median of Distribution DCH 2015, Table 2.13.10[e]				
Lead	P	NA	D	1.00E+04	Median of Distribution DCH 2015, Table 2.13.2				
Neptunium	P	NA	D	2.31E+01	Median of Distribution DCH 2015, Table 2.13.2				
Polonium	P	NA	D	2.30E+02	Median of Distribution DCH 2015, Table 2.13.2				
Protactinium	P	NA	D	3.80E+02	Median of Distribution DCH 2015, Table 2.13.10[e]				
Plutonium	P	NA	D	9.53E+02	Median of Distribution DCH 2015, Table 2.13.2				
Radium	P	NA	D	7.13E+02	Median of Distribution DCH 2015, Table 2.13.2				
Tellurium	P	NA	D	3.81E+01	Median of Distribution DCH 2015, Table 2.13.10[e]				
Thorium	P	NA	D	1.80E+04	Median of Distribution in DCH 2015, Table 2.13.2				
Uranium	P	NA	D	3.11E+02	Median of Distribution in DCH 2015, Table 2.13.2				
Distribution coefficients (saturated zone) (cm^3/g)									
Am-241	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1 [10]	6.91	1.95		
C-14	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[f] [10]	2.4	3.22		
Ce-144	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	5.99	0		
Cm-243/ 244	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	8.13	2.64		
Co-58	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	5.56	2.89		
Co-60	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	5.56	2.89		
Cs-134	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	6.27	1.79		
Cs-137	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	6.27	1.79		
Eu-152	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[f]	6.72	3.22		

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Eu-154	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[f]	6.72	3.22		
Eu-155	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[f]	6.72	3.22		
Fe-55	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	5.77	0		
H-3	P	1	S	Lognormal-N	DCH 2015, Table 2.13.10[f]	-2.81	0.5		
Ni-59	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	4.87	2.3		
Ni-63	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	4.87	2.3		
Np-237	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	2.64	1.39		
Pu-238	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	5.99	1.39		
Pu-239/ 240	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	5.99	1.39		
Pu-241	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	5.99	1.39		
Sb-125	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	2.83	1.79		
Sr-90	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	3.09	1.79		
Tc-99	P	1	S	Lognormal-N	DCH 2015, Table 2.13.1	-3.22	1.1		
Daughter Products									
Actinium	P	NA	D	8.29E+02	Median of Distribution DCH 2015, Table 2.13.10[f]				
Americium	P	NA	D	1.00E+03	Median of Distribution in DCH 2015, Table 2.13.1 [10]				
Curium	P	NA	D	3.40E+03	Median of Distribution in DCH 2015, Table 2.13.1				
Gadolinium	P	NA	D	8.29E+02	Median of Distribution DCH 2015, Table 2.13.10[f]				
Lead	P	NA	D	2.19E+02	Median of Distribution in DCH 2015, Table 2.13.1				
Neptunium	P	NA	D	1.40E+01	Median of Distribution in DCH 2015, Table 2.13.1				
Polonium	P	NA	D	1.00E+02	Median of Distribution in DCH 2015, Table 2.13.1				
Protactinium	P	NA	D	3.80E+02	Median of Distribution DCH 2015, Table 2.13.10[f]				
Plutonium	P	NA	D	3.99E+02	Median of Distribution in DCH 2015, Table 2.13.1				
Radium	P	NA	D	3.10E+03	Median of Distribution in DCH 2015, Table 2.13.1				

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Tellurium	P	NA	D	3.81E+01	Median of Distribution DCH 2015, Table 2.13.10[f]				
Thorium	P	NA	D	699	Median of Distribution in DCH 2015, Table 2.13.1				
Uranium	P	NA	D	110	Median of Distribution in DCH 2015, Table 2.13.1				
Initial concentration of radionuclides in groundwater (pCi/l)	P	3	D	0	No existing groundwater contamination	NR	NR	NR	NR
Calculation Times									
Time since placement of material (y)	P	3	D	0	Dose calculation applies to concentrations present at time of license termination	NR	NR	NR	NR
Time for calculations (y)	P	3	D	0, 1, 3, 10, 30, 100, 300, 1000	RESRAD default values acceptable times for FCS assessment	NR	NR	NR	NR
Contaminated Zone									
Area of contaminated zone (m^2)	P	2	D	79,600	Total area of Class 1 open land areas: DA and waste haul path [9]. Total area assumed to be a contiguous circle.	NR	NR	NR	NR
Thickness of contaminated zone (m)	P	2	D	1.0 or 0.15	DCGL calculated for standard 0.15 m soil contaminated zone thickness DCGL also calculated using a 1.0 m thickness to apply if thicker soil sources (> 0.15 m) are identified during remediation or FSS operations.	NR	NR	NR	NR
Length parallel to aquifer flow (m)	P	2	D	318	Diameter of 79,600 m^2 contaminated area.	NR	NR	NR	NR
Does the initial contamination penetrate the water table?	NA	NA	NA	No	No soil contamination has been identified in the saturated zone. See LTP Chapter 2	NA	NA	NA	NA
Contaminated fraction below water table	P	3	D	NA	NA	NR	NR	NR	NR
Cover and Contaminated Zone Hydrological Data									

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Cover depth (m)	P	2	D	0	No Cover	NA	NA	NA	NA
Density of cover (g/cm^3)	P	1	NA	NA	No Cover	NA	NA	NA	NA
Cover erosion rate (m/y)	P, B	2	NA	NA	No Cover	NA	NA	NA	NA
Density of contaminated zone (g/cm^3)	P	1	D	1.5	Site-specific mean [1]	NA	NA	NA	NA
Contaminated zone erosion rate (m/y)	P, B	2	S	Continuous Logarithmic	RESRAD Default PDF (NUREG/CR-6697 Att. C Table 3.8-1) [18]	See NUREG/CR-6697 Att. C, Table 3.8-1			
Contaminated zone total porosity	P	2	D	0.43	Site-specific mean [1]	NA	NA	NA	NA
Contaminated zone field capacity	P	3	D	0.28	Site-specific mean for unsaturated zone [1]	NA	NA	NA	NA
Contaminated zone hydraulic conductivity (m/y)	P	2	S	34.4	Site-specific mean for unsaturated zone [1]	NA	NA	NA	NA
Contaminated zone b parameter	P	2	S	Bounded Lognormal – N	NUREG 6697, Att. C Table 3.5-1 [18] distribution for site-specific unsaturated zone soil type – silt loam	1.28	0.334	1.28	10.1
Humidity in air (g/m^3)	P	3	D	Truncated Lognormal-N	Active for H-3 only RESRAD Default PDF (NUREG/CR-6697 Att. C) [18]	1.98	0.334	0.001	0.999
Evapotranspiration coefficient	P	2	S	Uniform	RESRAD default PDF (NUREG/CR-6697 Att. C) [18] with site specific adjustment Calculation provided in Section A.6.1.1	0.5	0.99	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Average annual wind speed (m/s)	P	2	S	uniform	Site-Specific range FCS DSAR [3] maximum and minimum. The DSAR data ends at year 1990. The wind speed data in the following web reference was reviewed to ensure wind speed after 1990 was within the DSAR range: https://www.weatherbase.com/weather/weatherall-print.php3?s=62652&cityname=&units= DSAR Range 6.3 mph – 10.6 mph converted to 2.8 m/s – 4.7 m/s. Equation for unit conversion provided in Section A.6.1.2	2.8	4.7	NR	NR
Precipitation (m/y)	P	2	D	0.76	Site-specific FCS DSAR [3]. The DSAR data ends at year 1990. The precipitation data reported by the Nebraska State Climate Office (https://nsco.unl.edu/) for years 1981-2010 at Eppley Airfield was reviewed to ensure no significant changes since 1990. Convert 29.9 inch/yr to 0.76 m/yr	NR	NR	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Irrigation (m/y)	B	3	D	0.19	Site-specific USDA National Agriculture Statistics Service, Census of Agriculture, Table 7 https://www.nass.usda.gov/Publications/AgCensus/2017/Online_Resources/Farm_and_Ranch_Irrigation_Survey/index.php Calculation provided in Section A.6.1.3	NR	NR	NR	NR
Irrigation mode	B	3	D	Overhead	Overhead irrigation is common practice in U. S.	NR	NR	NR	NR
Runoff coefficient	P	2	S	Uniform	RESRAD Default PDF (NUREG/CR-6697, Att. C) [18]	0.1	0.8	NR	NR
Watershed area for nearby stream or pond (m^2)	P	3	D	1.0E+06	RESRAD Default	NR	NR	NR	NR
Accuracy for water/soil computations	-	3	D	1.00E-03	RESRAD Default	NR	NR	NR	NR
Saturated Zone Hydrological Data									
Density of saturated zone (g/cm^3)	P	1	D	1.49	Site-specific [1]	NR	NR	NR	NR
Saturated zone total porosity	P	1	D	0.45	Site-specific [1]	NR	NR	NR	NR
Saturated zone effective porosity	P	1	D	0.20	Site-specific [1]	NR	NR	NR	NR
Saturated zone field capacity	P	3	D	0.24	Site-specific [1]	NR	NR	NR	NR
Saturated zone hydraulic conductivity (m/y)	P	1	D	4350	Site-specific [1]	NR	NR	NR	NR
Saturated zone hydraulic gradient	P	2	D	8.4E-04	Site-specific [1]	NR	NR	NR	NR
Saturated zone b parameter	P	2	D	NA saturated zone b not active because water table drop rate =0	NA	NR	NR	NR	NR
Water table drop rate (m/y)	P	3	D	0	Well pumping rate assumed small relative to water table volume.	NR	NR	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Well pump intake depth (m below water table)	P	2	S	Continuous Linear	Site-specific [1] Calculation provided in Section A.6.1.4	See Section A.6.1.4			
Model: Non-dispersion (ND) or Mass-Balance (MB)	P	3	D	Non-dispersion	Non-dispersion model applicable	NR	NR	NR	NR
Well pumping rate (m^3/y)	B, P	2	D	4550	Calculated using method in NUREG/CR-6697, Att. C, Section 3.10 [18]with Nebraska-specific irrigation rate, NUREG/CR-5512 vol. 3 human drinking water intake and livestock water intake. Irrigated land area of 22000 m^2 assumed (2,000 m^2 garden and 20,000 m^2 for livestock fodder) Calculation provided in Section A.6.1.5	NR	NR	NR	NR
Unsaturated Zone Hydrological Data									
Number of unsaturated zone strata	P	3	D	1	One unsaturated zone	NA	NA	NA	NA
Unsat. Zone thickness (m)	P	1	D	0.1	Basis provided in Section A.6.1.6	NA	NA	NA	NA
Unsat. Zone soil density (g/cm^3)	P	2	D	1.5	Site-specific [1]	NR	NR	NR	NR
Unsat. Zone total porosity	P	2	D	0.43	Site-specific [1]	NR	NR	NR	NR
Unsat. Zone effective porosity	P	2	D	0.16	Site-specific [1]	NR	NR	NR	NR
Unsat. Zone field capacity	P	3	D	0.28	Site-specific [1]	NR	NR	NR	NR
Unsat. Zone hydraulic conductivity (m/y)	P	2	D	34.4	Site-specific [1]	NR	NR	NR	NR
Unsat. Zone b parameter	P	2	S	Bounded Lognormal-N	NUREG 6697 distribution for site soil type – silt loam [18]	1.28	0.334	1.28	10.1
Occupancy									

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Inhalation rate (m^3/y)	M,B	3	D	8600	NUREG/CR-5512, Vol. 3 Table 6.87 [11] Occupancy time weighted average of Indoor, outdoor, and gardening breathing rates. Calculation provided in Section A.6.1.7	NR	NR	NR	NR
Mass loading for inhalation (g/m^3)	P,B	2	S	Continuous Linear	RESRAD Default PDF (NUREG/CR-6697, Att. C) [18]	See NUREG-6697, Att. C, Table 4.6-1	See NUREG-6697, Att. C, Table 4.6-1	See NUREG-6697, Att. C, Table 4.6-1	See NUREG-6697, Att. C, Table 4.6-1
Exposure duration	B	3	D	30	RESRAD Default (Parameter not used in dose calculation)	NR	NR	NR	NR
Indoor dust filtration factor	P,B	2	S	Uniform	NUREG/CR-6697, Att. C [18]	0.15	0.95		
Shielding factor, external gamma	P	2	D	0.552	NUREG/CR-5512, Vol. 3 Table 6.87 [11]. The D&D default of 0.552 value exceeds the 75th percentile of the RESRAD PDF (0.4) and is therefore applied as a prudently conservative deterministic value	NR	NR	NR	NR
Fraction of time spent indoors	B	3	D	0.66	NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Fraction of time spent outdoors (on site)	B	3	D	0.12	NUREG/CR-5512, Vol. 3 Table 6.87 (outdoors + gardening) [11]	NR	NR	NR	NR
Shape factor flag, external gamma	P	3	D	Circular	Circular contaminated zone assumed for modeling purposes	NR	NR	NR	NR
Ingestion, Dietary									

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Fruits, non-leafy vegetables, grain consumption (kg/y)	M,B	2	D	224	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 (other vegetables + fruits + grain) [11] Plant ingestion rate represents consumption of home-grown food which is grown in contaminated area. The 112 kg/y D&D default value was doubled to 224 in order to allow RESRAD to calculate correction factors to contaminated zone areas ≤1000 m^2. RESRAD default correction factor for onsite vegetable consumption is 0.5 for areas > 1,000 m^2	NR	NR	NR	NR
Leafy vegetable consumption (kg/y)	M,B	3	D	42.8	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11] Plant ingestion rate represents consumption of home-grown food which is grown in contaminated area. The 21.4 kg/y D&D default value was doubled to 42.8 in order to allow RESRAD to calculate correction factors to contaminated zone areas ≤ 1000 m^2. RESRAD default correction factor for onsite vegetable consumption is 0.5 for areas > 1,000 m^2	NR	NR	NR	NR
Milk consumption (L/y)	M,B	2	D	233	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Meat and poultry consumption (kg/y)	M,B	3	D	65.1	D&D Default NUREG/CR5512, Vol. 3 Table 6.87 (beef + poultry)	NR	NR	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Fish consumption (kg/y)	M,B	3	D	NA	NUREG/CR-5512, Vol. 3 Table 6.87 [11] Aquatic Pathway inactive	NR	NR	NR	NR
Other seafood consumption (kg/y)	M,B	3	D	NA	RESRAD User's Manual Table D.2 [17] Aquatic Pathway inactive	NR	NR	NR	NR
Soil ingestion rate (g/y)	M,B	2	D	18.3	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Drinking water intake (L/y)	M,B	2	D	478	NUREG/CR-5512, Vol. 3 Table 6.87 [11] 1.31 L/d converted to 478 L/y	NR	NR	NR	NR
Contamination fraction of drinking water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR
Contamination fraction of household water (if used)	B,P	3		NA	Applicable to radon pathway only				
Contamination fraction of livestock water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR
Contamination fraction of irrigation water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR
Contamination fraction of aquatic food	B,P	2	D	NA	Assumption that pond is constructed that intercepts contaminated water is not credible at FCS site	NR	NR	NR	NR
Contamination fraction of plant food	B,P	3	D	-1	RESRAD default - plant ingestion rate reduced for contaminated areas less than 1,000 m^2	NR	NR	NR	NR
Contamination fraction of meat	B,P	3	D	-1	RESRAD default - meat ingestion rate reduced for contaminated areas less than 20,000 m^2	NR	NR	NR	NR
Contamination fraction of milk	B,P	3	D	-1	RESRAD default - milk ingestion rate reduced for contaminated areas less than 20,000 m^2	NR	NR	NR	NR
Ingestion, Non-Dietary									

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Livestock fodder intake for meat (kg/day)	M	3	D	27.1	D&D Default NUREG/CR5512, Vol. 3 Table 6.87 (forage, grain and hay for beef cattle + poultry + layer hen) [11]	NR	NR	NR	NR
Livestock fodder intake for milk (kg/day)	M	3	D	63.21	D&D Default NUREG/CR5512, Vol. 3 Table 6.87 (forage + grain + hay) [11]	NR	NR	NR	NR
Livestock water intake for meat (L/day)	M	3	D	50.6	D&D Default NUREG/CR5512, Vol. 3 Table 6.87 (beef cattle + poultry + layer hen)	NR	NR	NR	NR
Livestock water intake for milk (L/day)	M	3	D	60	D&D Default NUREG/CR5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Livestock soil intake (kg/day)	M	3	D	0.7	D&D Default NUREG/CR5512, Vol. 3 Table 6.87 [11] Discussion and calculation provided Section A.6.1.8	NR	NR	NR	NR
Mass loading for foliar deposition (g/m^3)	P	3	D	4.00E-04	NUREG/CR-5512, Vol. 3 Table 6.87, gardening [11]	NR	NR	NR	NR
Depth of soil mixing layer (m)	P	2	S	Triangular	NUREG/CR-6697, Att. C [18]	0	0.15	0.6	
Depth of roots (m)	P	1	S	Uniform	NUREG/CR-6697, Att. C [18]	0.3	4.0		
Drinking water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR
Household water fraction from ground water (if used)	B,P	3		NA	Radon inhalation pathway inactive				
Livestock water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR
Irrigation fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR
Wet weight crop yield for Non-Leafy (kg/m^2)	P	2	S	Truncated Lognormal – N	NUREG/CR-6697, Att. C [18]	0.56	0.48	0.001	0.999
Wet weight crop yield for Leafy (kg/m^2)	P	3	D	2.89	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Wet weight crop yield for Fodder (kg/m^2)	P	3	D	1.89	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Growing Season for Non-Leafy (y)	P	3	D	0.246	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Growing Season for Leafy (y)	P	3	D	0.123	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Growing Season for Fodder (y)	P	3	D	0.082	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Translocation Factor for Non-Leafy	P	3	D	0.1	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Translocation Factor for Leafy	P	3	D	1	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR
Translocation Factor for Fodder	P	3	D	1	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Weathering Removal Constant for Vegetation (1/y)	P	2	S	Triangular	NUREG/CR-6697, Att. C [18]	5.1	18	84	
Wet Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Wet Foliar Interception Fraction for Leafy	P	2	S	Triangular	NUREG/CR-6697, Att. C [18]	0.06	0.67	0.95	
Wet Foliar Interception Fraction for Fodder	P	3	D	0.35	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Dry Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Dry Foliar Interception Fraction for Leafy	P	3	D	0.35	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Dry Foliar Interception Fraction for Fodder	P	3	D	0.35	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Storage times of contaminated foodstuffs (days):									
Fruits, non-leafy vegetables, and grain	B	3	D	14	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Leafy vegetables	B	3	D	1	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Milk	B	3	D	1	D&D Default NUREG/CR-5512, Vol. 3 Table 6.87 [11]	NR	NR	NR	NR
Meat	B	3	D	20	D&D Default NUREG/CR-5512, Vol. 3	NR	NR	NR	NR
Fish	B	3	D	NA	Aquatic pathway inactive	NR	NR	NR	NR
Crustacea and mollusks	B	3	D	NA	Aquatic pathway inactive	NR	NR	NR	NR
Well water	B	3	D	1	RESRAD User's Manual Table D.6 [17]	NR	NR	NR	NR
Surface water	B	3	D	1	RESRAD User's Manual Table D.6 [17]	NR	NR	NR	NR
Livestock fodder	B	3	D	45	RESRAD User's Manual Table D.6 [17]	NR	NR	NR	NR
Special Radionuclides (C-14)									
C-12 concentration in water (g/cm^3)	P	3	D	2E-05	RESRAD User's Manual [17] Napier, et al [19]	NR	NR	NR	NR
C-12 concentration in contaminated soil (g/g)	P	3	D	0.03	Value applies to site-specific soil which is sand or silt loam with minimal organic content [17] and NUREG-5512, Vol 3, Table 6.87 [11]	NR	NR	NR	NR
Fraction of vegetation carbon from soil	P	3	D	0.02	2% of carbon absorbed through roots- Equation L.31 from [17]	NR	NR	NR	NR
Fraction of vegetation carbon from air	P	3	D	0.98	98% of carbon absorbed through air – Equation L.31 from [17]	NR	NR	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
C-14 evasion layer thickness in soil (m)	P	2	D	0.3	The maximum soil thickness layer through which C-14 can escape to the air by conversion to CO2. C-14 below this depth is assumed to be trapped [17]	NR	NR	NR	NR
C-14 evasion flux rate from soil (1/sec)	P	3	D	7E-07	Evasion rate for sandy soil from Table L.2 of [17] which is appropriate for FCS 22/y converted to 7E-07/s	NR	NR	NR	NR
C-12 evasion flux rate from soil (1/sec)	P	3	D	1E-10	Evasion rate from Equation L.27 of [17] 0.032/y converted to 1E-10/s	NR	NR	NR	NR
Fraction of grain in beef cattle feed	B	3	D	0.09	NUREG-5512, Vol 3, table 6.87 [11] Calculation provided in Section A.6.1.8	NR	NR	NR	NR
Fraction of grain in milk cow feed	B	3	D	0.03	NUREG-5512, Vol 3, table 6.87 [11] Calculation provided in Section A.6.1.9	NR	NR	NR	NR
Dose Conversion Factors (DCF) (Inhalation mrem/pCi)									
All Radionuclides	M	3	D	FGR11 dose factors – RESRAD Default	RESRAD default is the most conservative FGR11 inhalation DCF	NR	NR	NR	NR
Dose Conversion Factors (DCF) (Ingestion mrem/pCi)									
All Radionuclides	M	3	D	FGR11 dose factors – RESRAD Default	RESRAD default is the most conservative FGR11 ingestion DCF	NR	NR	NR	NR
Plant Transfer Factors (pCi/g plant)/(pCi/g soil)									
Am-241	P	1	D	1.83E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
C-14	P	1	D	1.28E+00	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Ce-144	P	1	D	3.93E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Cm-243/ 244	P	1	D	1.83E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Co-58	P	1	D	1.46E-01	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Co-60	P	1	D	1.46E-01	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Cs-134	P	1	D	7.81E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Cs-137	P	1	D	7.81E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Eu-152	P	1	D	4.22E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Eu-154	P	1	D	4.22E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Eu-155	P	1	D	4.22E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Fe-55	P	1	D	1.83E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
H-3	P	1	D	1.01E+01	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Ni-59	P	1	D	9.24E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Ni-63	P	1	D	9.24E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Np-237	P	1	D	3.72E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Pu-238	P	1	D	1.83E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Pu-239/ 240	P	1	D	1.83E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Pu-241	P	1	D	1.83E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Sb-125	P	1	D	1.95E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Sr-90	P	1	D	5.89E-01	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Tc-99	P	1	D	9.28E+00	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Meat Transfer Factors (pCi/kg)/(pCi/d)									
Am-241	P	1	D	5.74E-05	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
C-14	P	1	D	6.11e-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Ce-144	P	1	D	3.71E-05	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Cm-243/ 244	P	1	D	4.00E-05	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Co-58	P	1	D	6.00E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Co-60	P	1	D	6.00E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Cs-134	P	1	D	6.55E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Cs-137	P	1	D	6.55E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Eu-152	P	1	D	4.02E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Eu-154	P	1	D	4.02E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Eu-155	P	1	D	4.02E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Fe-55	P	1	D	3.93E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
H-3	P	1	D	2.36E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Ni-59	P	1	D	9.26E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Ni-63	P	1	D	9.26E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Np-237	P	1	D	1.59E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Pu-238	P	1	D	1.14E-04	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Pu-239/ 240	P	1	D	1.14E-04	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Pu-241	P	1	D	1.14E-04	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Sb-125	P	1	D	1.85E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Sr-90	P	1	D	1.31E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Tc-99	P	1	D	1.60E-04	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Milk Transfer Factors (pCi/L)/(pCi/d)									

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Am-241	P	1	D	3.20E-06	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
C-14	P	1	D	2.25E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Ce-144	P	1	D	4.81E-05	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Cm-243/ 244	P	1	D	3.72E-06	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Co-58	P	1	D	3.22E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Co-60	P	1	D	3.22E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Cs-134	P	1	D	1.37E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Cs-137	P	1	D	1.37E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Eu-152	P	1	D	1.11E-04	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Eu-154	P	1	D	1.11E-04	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Eu-155	P	1	D	1.11E-04	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Fe-55	P	1	D	4.82E-04	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
H-3	P	1	D	1.85E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Ni-59	P	1	D	3.20E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Ni-63	P	1	D	3.20E-02	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Np-237	P	1	D	1.61E-05	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Pu-238	P	1	D	1.39E-06	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Pu-239/ 240	P	1	D	1.39E-06	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Pu-241	P	1	D	1.39E-06	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Sb-125	P	1	D	1.11E-04	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Sr-90	P	1	D	2.76E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]			
						1	2	3	4
Tc-99	P	1	D	1.59E-03	75th percentile of RESRAD Default PDF	NR	NR	NR	NR
Bioaccumulation Factors for Fish ((pCi/kg)/(pCi/L))									
NA	P	3	NA	Inactive Pathway	NA	NA	NA	NA	NA
Bioaccumulation Factors for Crustacea/ Mollusks ((pCi/kg)/(pCi/L))									
NA	P	3	NA	Inactive Pathway	NA	NA	NA	NA	NA
Graphics Parameters									
Number of points				32	RESRAD Default	NR	NR	NR	NR
Spacing				log	RESRAD Default	NR	NR	NR	NR
Time Integration Parameters									
Maximum number of points for dose				17	RESRAD Default	NR	NR	NR	NR

a P = physical, B = behavioral, M = metabolic; (NUREG/CR-6697, Attachment B, Table 4 [18])
b 1 = high-priority parameter, 2 = medium-priority parameter, 3 = low-priority parameter (NUREG/CR-6697, Attachment B, Table 4.1 [18])
c D = deterministic, S = stochastic
d Distributions Statistical Parameters:
- Lognormal-n: 1= mean, 2 = standard deviation
- Bounded lognormal-n: 1= mean, 2 = standard deviation, 3 = minimum, 4 = maximum
- Truncated lognormal-n: 1= mean, 2 = standard deviation, 3 = lower quantile, 4 = upper quantile
- Bounded normal: 1 = mean, 2 = standard deviation, 3 = minimum, 4 = maximum
- Beta: 1 = minimum, 2 = maximum, 3 = P-value, 4 = Q-value
- Triangular: 1 = minimum, 2 = mode, 3 = maximum
- Uniform: 1 = minimum, 2 = maximum
- Continuous linear distribution, number of points, value of points, and cdf of points
- Continuous logarithmic distribution, number of points, value of points, and cdf of points.

E Distribution is not provided in DCH 2015, Table 2.13.2 [10]. The distribution in DCH 2015, Table 2.13.10 [10] is used
f Distribution is not provided in DCH 2015, Table 2.13.1 [10]. The distribution in DCH 2015, Table 2.13.10 [10] is used

Calculations and Additional Information Supporting Parameter Selection

A.6.1.1 Evapotranspiration Coefficient

The default PDF for the Evapotranspiration Coefficient (C_e) in RESRAD ONSITE is a uniform distribution with a minimum and maximum of 0.5 and 0.75, respectively. The C_e is unitless and must be less than one. To provide additional assurance that this range is applicable to the FCS site, a regional C_e was estimated using the method described in the Argonne National Laboratory "Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil and Building Structures," 2015 [10]. This includes two steps: 1) deriving a regional evapotranspiration rate (ET_r) of 75 cm/yr from Figure 3.3.1 of [10] and 2) using the regional ET_r to estimate a regional C_e of 0.93 (see calculation below). The 0.94 value exceeds the PDF maximum of 0.75. Therefore, the PDF maximum was increased to 0.99 to include the regional estimate. The minimum PDF value of 0.5 was not changed.

$$\boldsymbol{C_e = \frac{ET_c}{(1 - C_r)P_r + I_r} = 0.94}$$

Where:

C_e = Regional Evapotranspiration Coefficient
ET_r = Regional Evapotranspiration rate of crop (RESRAD uses the term ET_r) = 0.75 m/yr
C_r = Runoff Coefficient = 0.2 (unitless)
P_r = Precipitation = 0.76 m/yr
I_r = Irrigation Rate = 0.19 m/yr

A.6.1.2 The m/s average annual wind speed is calculated as follows:

$$W_{m/s} = \frac{W_{mph} * 1609}{3600}$$

Where:

$W_{m/s}$ = average annual wind speed (m/s)
W_{mph} = site-specific annual average wind speed in units of mph

1609 = conversion factor m/mi
3600 = conversion factor s/hr

A.6.1.3 The 0.19 m/y irrigation rate is calculated as follows:

$$I_r = \frac{V_i}{A_i} * 0.3048 = 0.19 \text{ m/yr}$$

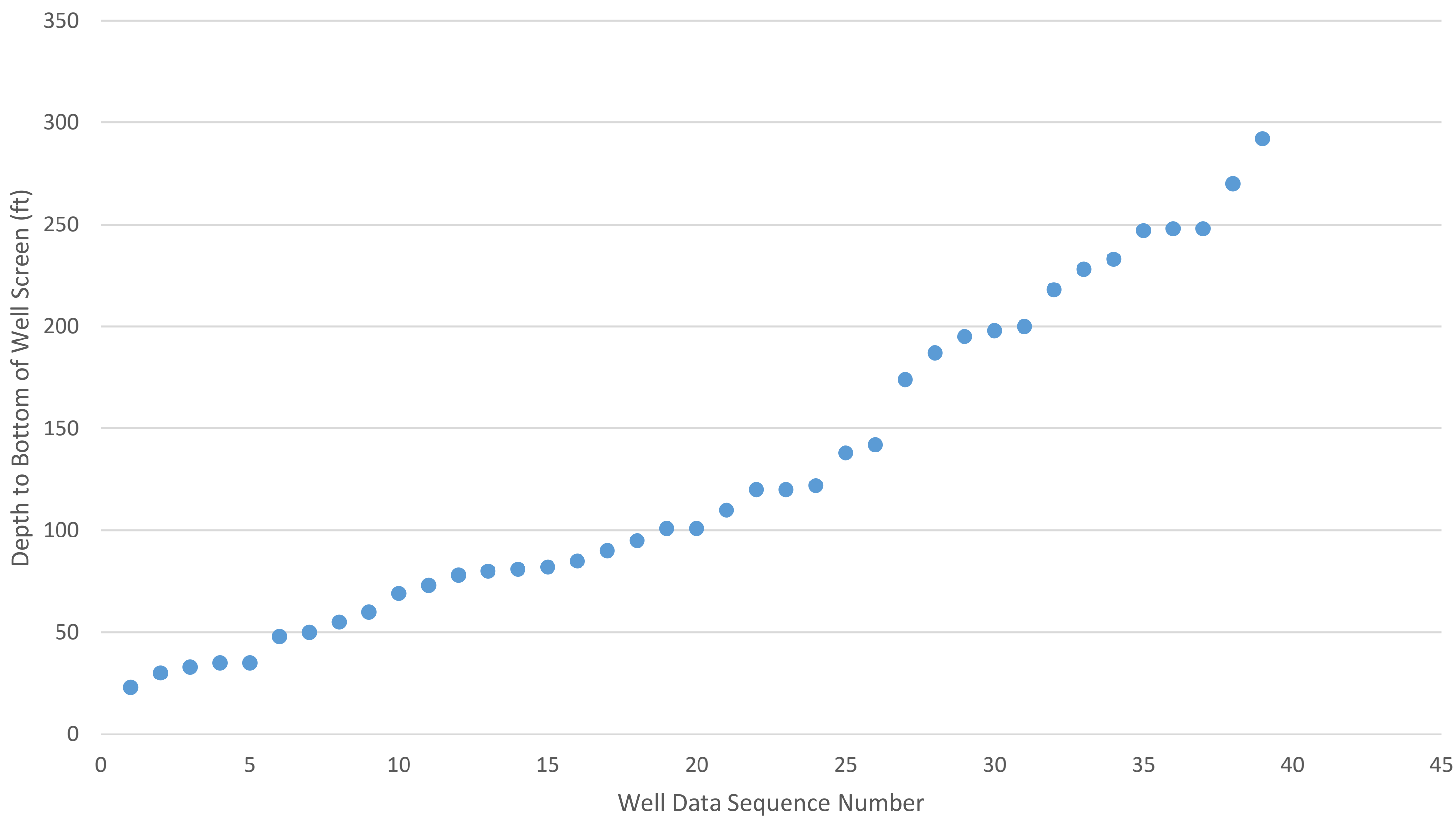
Where:

I_r = irrigation rate (m/yr)
V_i = annual irrigation volume in Nebraska = 4,856,075 acre-feet/yr
A_i = total irrigated acres in Nebraska = 7,665,880 acres
0.3048 = conversion factor m/ft

A.6.1.4 Well Pump Intake Depth

The depth to the bottom of the well screen for 36 local irrigation and domestic wells are provided in [1]. The data is plotted in Figure A.6.1. The data is used to generate a continuous linear PDF for well pump intake depth. The well depths from 23 ft to 78 ft are plotted in Figure A.6.2 to select the Y intercept value to use as the well depth at 0.0 cumulative probability. Finally, the well data are converted from ft to m and fit to a continuous linear PDF as shown in Figure A.6.3 and listed in the Table A.6.1.

Figure A.6.1 - Depth to Bottom of Well Screen (Local Wells)



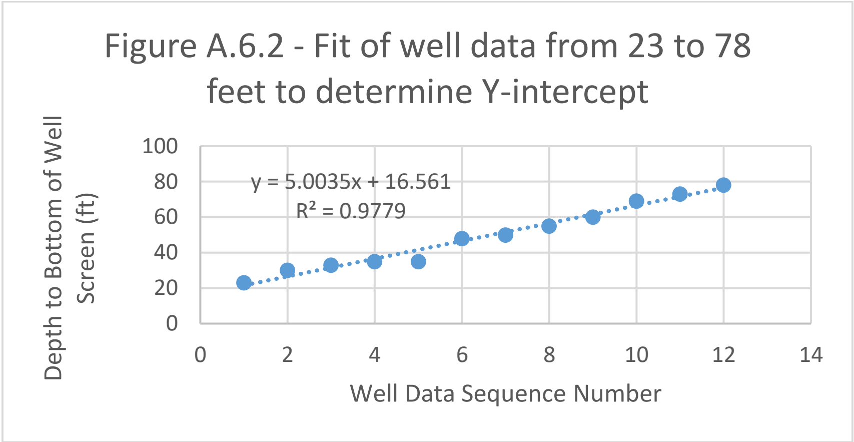
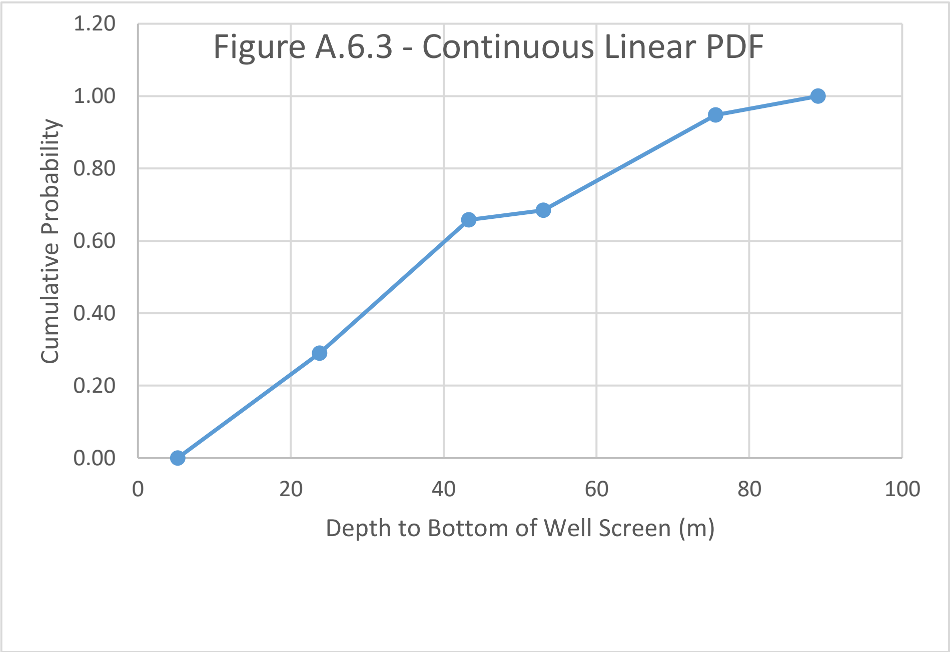


Table A.6.1 – Continuous Linear PDF for Well Pump Intake Depth

Well Pump Intake Depth (m)	Cumulative Probability
5	0.00
24	0.29
43	0.66
53	0.68
76	0.95
89	1.00



A.6.1.5 Well Pumping Rate

Calculated using method in NUREG/CR-6697, Att. C, Section 3.10 with Nebraska-specific irrigation rate, NUREG/CR-5512 vol. 3, Table 6.87 human drinking water intake and livestock water intake. Three scenarios calculated: 1) no garden or livestock (i.e., no irrigation), 2) irrigated land area of 22000 m^2 (2,000 m^2 garden and 20,000 m^2 for livestock), and 3) irrigated land area of 40,000 m^2.

Equations:

$$I_{r,l} = 110 * \frac{365.25}{1000}$$

$$I_{r,f} = 0.19 * A_i$$

$$I_{dw} = 478 * \frac{4}{1000}$$

Where:

$I_{r,l}$ = annual livestock annual water ingestion rate (m^3/yr)
110 = livestock daily water ingestion rate (L/d)
365.25 = d/yr
1000 = conversion factor L/m
$I_{r,f}$ = agriculture filed irrigation rate (m^3/yr)
0.19 = site-specific irrigation rate (0.19 m^3/yr)
A_i = area of irrigated fields (m^2)
478 = annual human drinking water ingestion rate (L/yr)
4 = 4 persons in household

Well Pumping Rate Calculation [13]

Inputs to Calculation

Irrigation rate	0.19	m^3/y
Household water use family of 4	328.7	m^3/y
meat and dairy cow water ingestion (1 dairy, 1 meat)	110	L/d
Drinking water per person	478	L/yr

Well Pumping Rate

	No Irrigation, No Garden, No Livestock	**Irrigation of 22,000 m^2**	**Irrigation of 40,000 m^2**
Household Water Use (m^3/yr)	328.7	328.7	328.7
Meat and Dairy Cow (m^3/yr)	0.0	40.2	40.2
Irrigation (m^3/yr)	0.0	4180.0	7600.0
Drinking Water (family of 4) (m^3/yr)	1.9	1.9	1.9
Well Pumping Rate (m^3/yr)	330.6	4550.8	7970.8

A.6.1.6 Unsaturated Zone Thickness

Onsite water levels collected from 17 wells in June 2020 show that the groundwater elevations ranged from 993.5 to 995.5 ft [1]. Given the site surface elevation of 1004 feet, this corresponds to 10.5 to 8.5 feet bgs. The DSAR states that the water table ranges from 2 to 17 ft bgs [3]. The 2-foot value is expected to correspond to a relatively high stage of the Missouri River and therefore occur infrequently. Under typical conditions, the water table is estimated to be 15-20 ft bgs [1] [3] [2]. The thickness of the soil contaminated zone is assumed to be 1 m which is a conservative value. Minimal soil contamination was identified during site characterization (see LTP Chapter 2) and no subsurface soil contamination was identified. The standard 0.15 m soil contaminated zone thickness is justified and would result in higher DCGLs, but the 1 m thickness was selected to ensure that the DCGLs apply if thicker soil sources (> 0.15 m) are identified during remediation or FSS operations. Because the water table could periodically be closer to the ground surface than the long-term average of 15 – 20 ft bgs, the unsaturated zone is set to a nominal thin value of 0.1 m which corresponds to a water table depth of 1.1 m (3.6 ft) bgs. A thin unsaturated zone is conservative because it reduces retardation time and minimizes the effect of decay in the calculation of maximum groundwater concentrations.

A.6.1.7 Inhalation Rate

From NUREG 5512, Vol 3, Table 6.87, occupancy times, converted to a fraction of year for entry to RESRAD ONSITE, are 0.66, 0.11, and 0.01, for indoor, outdoor and gardening, respectively. The occupancy fractions were normalized to values of 0.846, 0.141, and 0.013 for indoor, outdoor and gardening, respectively, were calculated. The corresponding inhalation rates are 0.9 m^3/hr, 1.4 m^3/hr. and 1.7 m^3/hr. The occupancy time weighted inhalation rate, $I_{r,w}$ is calculated as follows assuming 8766 hr/yr (365.25*24):

$$I_{r,w} = (0.846 * 8766 * 0.9) + (0.141 * 8766 * 1.4) + (0.013 * 8766 * 1.7) = 8600\, m^3/yr$$

A.6.1.78 Livestock Intake of Soil

The D&D default value for soil intake of beef cattle and milk cow is 2% of the dry matter forage intake which is 8.13 kg/d and 35.17 kg/d, respectively (Table 6.87 of [11]). The RESRAD ONSITE parameter for livestock intake of soil does not differentiate between beef cattle and milk cow. Therefore, the highest forage intake of 35.17 kg/d for milk cow is used to calculate the livestock intake of soil. Two percent of 35.17 kg/d is 0.70 kg/d.

A.6.1.9 Fraction of Grain in Beef Cattle and Milk Cow Feed

The total feed intake rate for beef cattle and dairy cow is derived by summing the D&D default feed contributions from forage, hay, and grain (Table 6.87 of [11]). The total feed intakes are 27.1 kg/d and 63.21 kg/d for beef cattle and milk cow, respectively. The corresponding contribution from grain are 2.42 kg/d and 1.95 kg/d. The fraction of grain in beef cattle and milk cow feed is therefore 2.42/27.1 = 0.09 and 1.95/63.21 = 0.03, respectively.

A.6.1.10 25th, 50th, and 75th Percentile Values of Stochastic Parameter Distributions

Non-Radionuclide Specific	**25th**	**50th**	**75th**
Contaminated Zone Erosion Rate	7.59E-04	1.49E-03	2.92E-03
Contaminated Zone b parameter	2.87	3.60	4.50
Evapotranspiration	0.62	0.75	0.87
Wind Speed	3.27	3.75	4.22
Runoff Coefficient	0.28	0.45	0.63
Well pump intake depth	21.40	34.80	59.00
b parameter of unsaturated zone	2.87	3.60	4.50
mass loading for inhalation	1.84E-05	2.35E-05	2.87E-05
Indoor Dust filtration factor	0.35	0.55	0.75
Depth of soil mixing layer	0.15	0.23	0.34
Depth of roots	1.23	2.15	3.08
Wet weight crop yield of fruit grain and non-leafy vegetables	1.27	1.75	2.42
Weathering removal constant of all vegetation	21.50	33.00	47.90
Wet foliar interception fraction of leafy vegetables	0.43	0.58	0.70
Humidity in air	5.79	7.24	9.07

Radionuclide Specific - Kd
DCH 2015 Table 2.13.2, Loam Soil Type

	log mean	std dev	25th	50th	75th
Am-241	8.34	1.79	1.25E+03	4.19E+03	1.40E+04
C-14	2.4	3.22	1.26E+00	1.10E+01	9.67E+01
Ce-144	8.01	1.1	1.43E+03	3.01E+03	6.32E+03
Cm-243/ 244	9.85	0.69	1.19E+04	1.90E+04	3.02E+04
Co-58	6.7	2.71	1.31E+02	8.12E+02	5.05E+03

Co-60	6.7	2.71	1.31E+02	8.12E+02	5.05E+03
Cs-134	8.16	1.39	1.37E+03	3.50E+03	8.93E+03
Cs-137	8.16	1.39	1.37E+03	3.50E+03	8.93E+03
Eu-152	6.72	3.22	9.45E+01	8.29E+02	7.27E+03
Eu-154	6.72	3.22	9.45E+01	8.29E+02	7.27E+03
Eu-155	6.72	3.22	9.45E+01	8.29E+02	7.27E+03
Fe-55	6.79	0.69	5.58E+02	8.89E+02	1.42E+03
H-3	-2.81	0.5	4.30E-02	6.02E-02	8.44E-02
Ni-59	5.19	1.61	6.06E+01	1.79E+02	5.32E+02
Ni-63	5.19	1.61	6.06E+01	1.79E+02	5.32E+02
Np-237	3.14	1.39	9.05E+00	2.31E+01	5.90E+01
Pu-238	6.86	1.39	3.73E+02	9.53E+02	2.43E+03
Pu-239/ 240	6.86	1.39	3.73E+02	9.53E+02	2.43E+03
Pu-241	6.86	1.39	3.73E+02	9.53E+02	2.43E+03
Sb-125	4.11	1.1	2.90E+01	6.09E+01	1.28E+02
Sr-90	4.04	1.61	1.92E+01	5.68E+01	1.68E+02
Tc-99	-2.66	1.1	3.33E-02	6.99E-02	1.47E-01

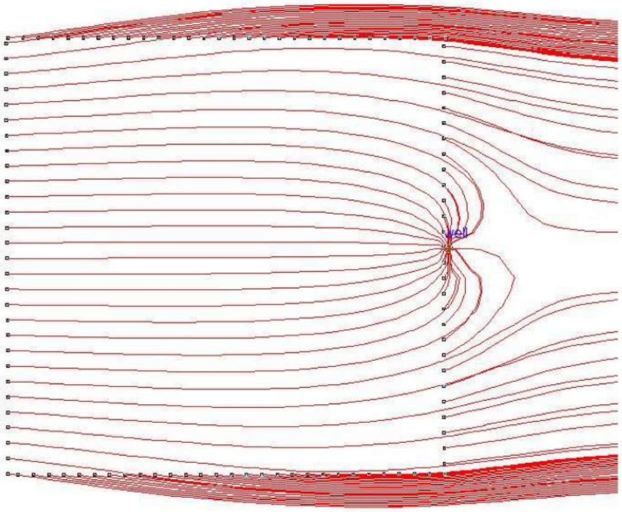
DCH 2015 2.13.1 Sand Soil Type

	log mean	std dev	25th	50th	75th
Am-241	6.91	1.95	2.69E+02	1.00E+03	3.73E+03
C-14	2.4	3.22	1.26E+00	1.10E+01	9.67E+01
Ce-144	5.99	1.00E-04	3.99E+02	3.99E+02	3.99E+02
Cm-243/ 244	8.13	2.64	5.72E+02	3.39E+03	2.01E+04
Co-58	5.56	2.89	3.70E+01	2.60E+02	1.82E+03
Co-60	5.56	2.89	3.70E+01	2.60E+02	1.82E+03
Cs-134	6.27	1.79	1.58E+02	5.28E+02	1.77E+03
Cs-137	6.27	1.79	1.58E+02	5.28E+02	1.77E+03
Eu-152	6.72	3.22	9.45E+01	8.29E+02	7.27E+03

Eu-154	6.72	3.22	9.45E+01	8.29E+02	7.27E+03
Eu-155	6.72	3.22	9.45E+01	8.29E+02	7.27E+03
Fe-55	5.77	1.00E-04	3.21E+02	3.21E+02	3.21E+02
H-3	-2.81	0.5	4.30E-02	6.02E-02	8.44E-02
Ni-59	4.87	2.3	2.76E+01	1.30E+02	6.15E+02
Ni-63	4.87	2.3	2.76E+01	1.30E+02	6.15E+02
Np-237	2.64	1.39	5.49E+00	1.40E+01	3.58E+01
Pu-238	5.99	1.39	1.56E+02	3.99E+02	1.02E+03
Pu-239/ 240	5.99	1.39	1.56E+02	3.99E+02	1.02E+03
Pu-241	5.99	1.39	1.56E+02	3.99E+02	1.02E+03
Sb-125	2.83	1.79	5.07E+00	1.69E+01	5.67E+01
Sr-90	3.09	1.79	6.57E+00	2.20E+01	7.35E+01
Tc-99	-3.22	1.1	1.90E-02	4.00E-02	8.39E-02

Attachment 6-2 MT3D Basement Particle Flow and Well Water Concentration Modeling Results

Fort Calhoun Station
Blair, Nebraska
0127960-006
MT3D: 1 Meter Source Thickness, Downstream Center (Outside)



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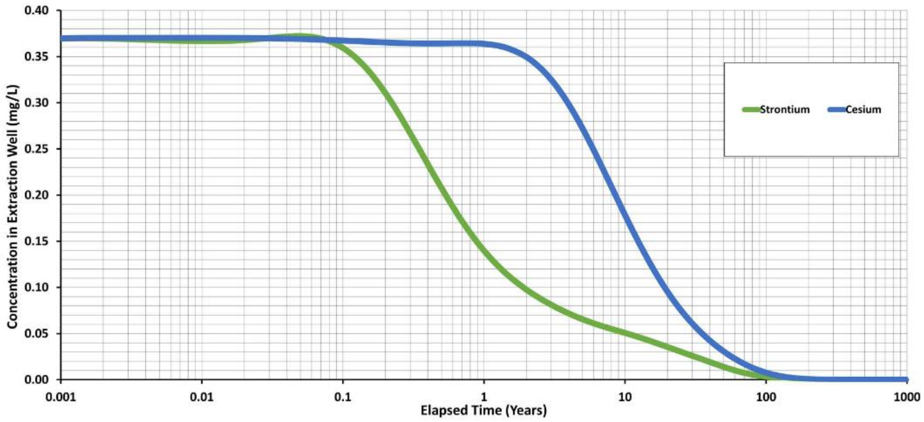
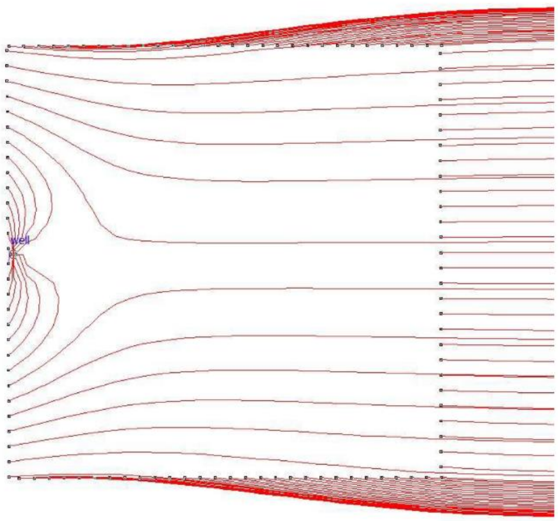


Cesium and Strontium Well Concentration

Haley & Aldrich Inc.
MT3D: 1 Meter Source Thickness, Downstream Center (Outside)

Fort Calhoun Station
Blair, Nebraska
0127960-006
MT3D: 1 Meter Source Thickness, Upstream Center



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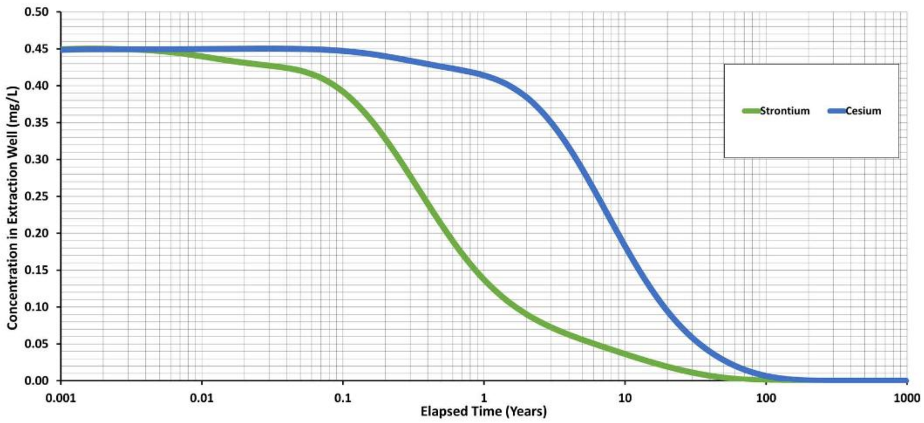
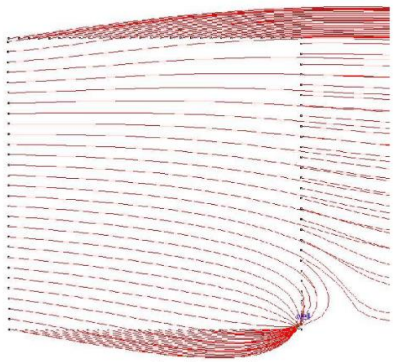


Cesium and Strontium Well Concentration

Haley & Aldrich Inc.
MT3D: 1 Meter Source Thickness, Upstream Center

Fort Calhoun Station
Blair, Nebraska
0127960-006
MT3D: 1 Meter Source Thickness, Downstream Corner



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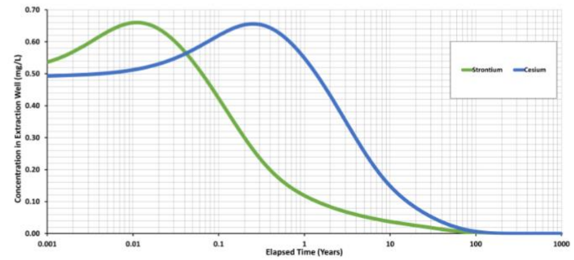
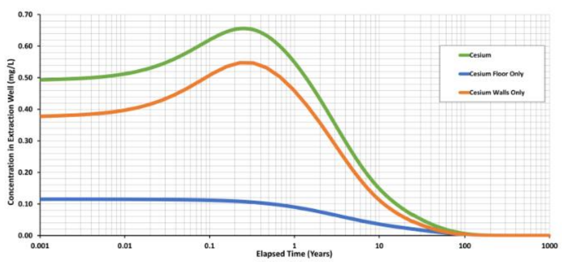
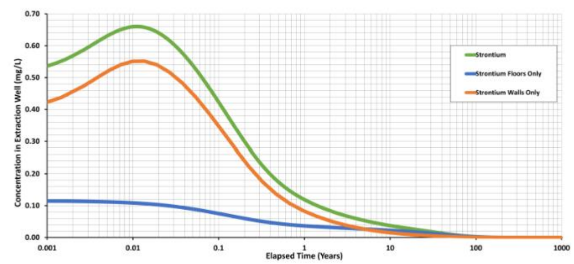


Cesium and Strontium Well Concentration



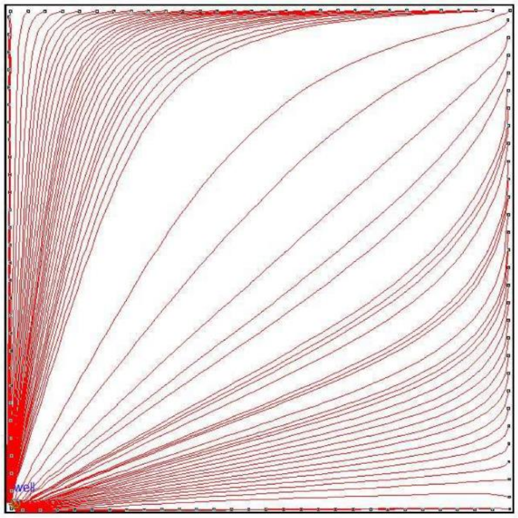
Cesium Well Concentration, Floor and Wall Contribution



Strontium Well Concentration, Floor and Wall Contribution

Haley & Aldrich Inc.
MT3D: 1 Meter Source Thickness, Downstream Corner

Fort Calhoun Station
Blair, Nebraska
0127960-006
MT3D: 0.05 Meter Source Thickness, Upstream Corner, Bathtub



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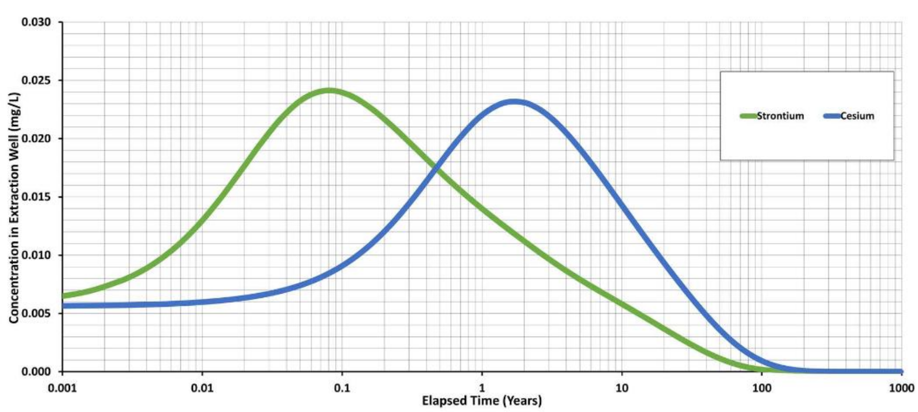
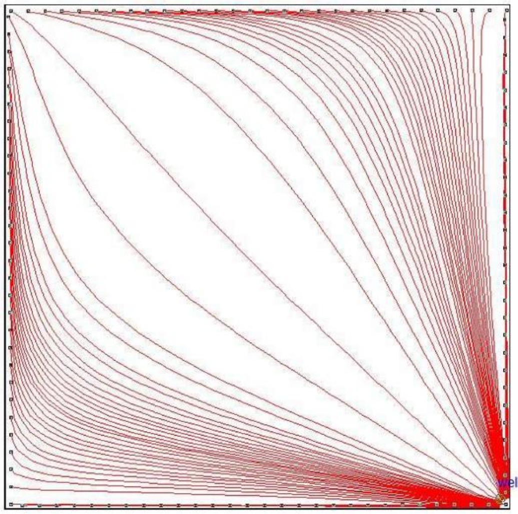


Cesium and Strontium Well Concentration

Haley & Aldrich Inc.
MT3D: 0.05 Meter Source Thickness, Upstream Corner, Bathtub

Fort Calhoun Station
Blair, Nebraska
0127960-006
MT3D: 1 Meter Source Thickness, Downstream Corner, Bathtub



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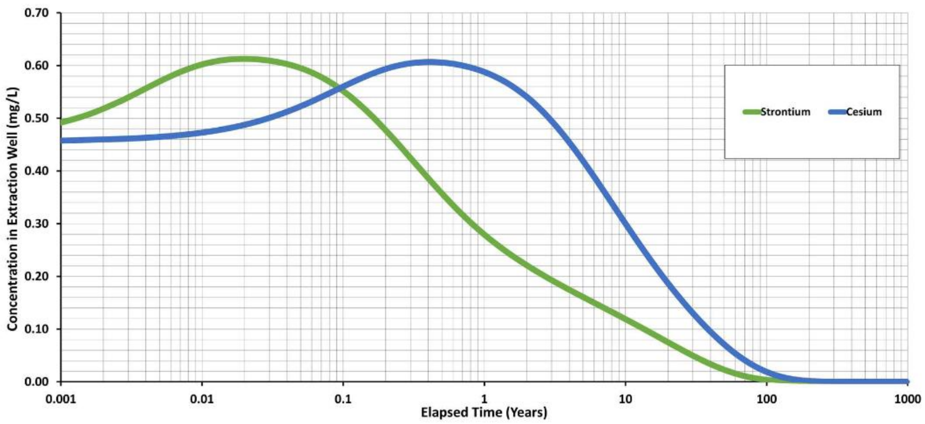

Cesium and Strontium Well Concentration

Haley & Aldrich Inc.
MT3D: 1 Meter Source Thickness, Downstream Corner, Bathtub

Attachment 6-3 List of RESRAD Reports Generated For DCGL Development

Soil Uncertainty Analysis

- Am-241 Soil Regression and Correlation Report
- Am-241 Soil Uncertainty Report
- C-14 Soil Regression and Correlation Report
- C-14 Soil Uncertainty Report
- Ce-144 Soil Regression and Correlation Report
- Ce-144 Soil Uncertainty Report
- Cm-243 Soil Regression and Correlation Report
- Cm-243 Soil Uncertainty Report
- Cm-244 Soil Regression and Correlation Report
- Cm-244 Soil Uncertainty Report
- Co-58 Soil Regression and Correlation Report
- Co-58 Soil Uncertainty Report
- Co-60 Soil Regression and Correlation Report
- Co-60 Soil Uncertainty Report
- Cs-134 Soil Regression and Correlation Report
- Cs-134 Soil Uncertainty Report
- Cs-137 Soil Regression and Correlation Report
- Cs-137 Soil Uncertainty Report
- Eu-152 Soil Regression and Correlation Report
- Eu-152 Soil Uncertainty Report
- Eu-154 Soil Regression and Correlation Report
- Eu-154 Soil Uncertainty Report
- Eu-155 Soil Regression and Correlation Report
- Eu-155 Soil Uncertainty Report
- Fe-55 Soil Regression and Correlation Report
- Fe-55 Soil Uncertainty Report
- H-3 Soil Regression and Correlation Report
- H-3 Soil Uncertainty Report
- Ni-59 Soil Regression and Correlation Report
- Ni-59 Soil Uncertainty Report
- Ni-63 Soil Regression and Correlation Report
- Ni-63 Soil Uncertainty Report
- Np-237 Soil Regression and Correlation Report
- Np-237 Soil Uncertainty Report

- Pu-238 Soil Regression and Correlation Report
- Pu-238 Soil Uncertainty Report
- Pu-239 Soil Regression and Correlation Report
- Pu-239 Soil Uncertainty Report
- Pu-240 Soil Regression and Correlation Report
- Pu-240 Soil Uncertainty Report
- Pu-241 Soil Regression and Correlation Report
- Pu-241 Soil Uncertainty Report
- Sb-125 Soil Regression and Correlation Report
- Sb-125 Soil Uncertainty Report
- Sr-90 Soil Regression and Correlation Report
- Sr-90 Soil Uncertainty Report
- Tc-99 Soil Regression and Correlation Report
- Tc-99 Soil Uncertainty Report

Soil DCGL Reports

- FCS Soil DCGL 0.15 m
- FCS Soil DCGL 1 m

BFM Insitu Uncertainty Analysis

- FCS BFM insitu Regression and Correlation Report Am-241
- FCS BFM insitu Regression and Correlation Report C-14
- FCS BFM insitu Regression and Correlation Report Ce-144
- FCS BFM insitu Regression and Correlation Report Cm-243
- FCS BFM insitu Regression and Correlation Report Cm-244
- FCS BFM insitu Regression and Correlation Report Co-58
- FCS BFM insitu Regression and Correlation Report Co-60
- FCS BFM insitu Regression and Correlation Report Cs-134
- FCS BFM insitu Regression and Correlation Report Cs-137
- FCS BFM insitu Regression and Correlation Report Eu-152
- FCS BFM insitu Regression and Correlation Report Eu-154
- FCS BFM insitu Regression and Correlation Report Eu-155
- FCS BFM insitu Regression and Correlation Report Fe-55
- FCS BFM insitu Regression and Correlation Report H-3
- FCS BFM insitu Regression and Correlation Report Ni-59

- FCS BFM insitu Regression and Correlation Report Ni-63
- FCS BFM insitu Regression and Correlation Report Np-237
- FCS BFM insitu Regression and Correlation Report Pu-238
- FCS BFM insitu Regression and Correlation Report Pu-239
- FCS BFM insitu Regression and Correlation Report Pu-241
- FCS BFM insitu Regression and Correlation Report Sb-125
- FCS BFM insitu Regression and Correlation Report Sr-90
- FCS BFM insitu Regression and Correlation Report Tc-99

- FCS BFM insitu Uncertainty Report Am-241
- FCS BFM insitu Uncertainty Report C-14
- FCS BFM insitu Uncertainty Report Ce-144
- FCS BFM insitu Uncertainty Report Cm-243
- FCS BFM insitu Uncertainty Report Cm-244
- FCS BFM insitu Uncertainty Report Co-58
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- FCS BFM insitu Uncertainty Report Pu-239
- FCS BFM insitu Uncertainty Report Pu-241
- FCS BFM insitu Uncertainty Report Sb-125
- FCS BFM insitu Uncertainty Report Sr-90
- FCS BFM insitu Uncertainty Report Tc-99

BFM *insitu* DSR

- FCS BFM insitu Scenario DSR

BFM Drilling Spoils DSR

FCS Drilling Spoils DSR

Embedded Pipe DSR

FCS Embedded Pipe DSR Concentration Report

FCS Embedded Pipe DSR

Buried Pipe Uncertainty Analysis

FCS Buried Pipe insitu Regression and Correlation Report Am-241
FCS Buried Pipe insitu Regression and Correlation Report C-14
FCS Buried Pipe insitu Regression and Correlation Report Ce-144
FCS Buried Pipe insitu Regression and Correlation Report Cm-243
FCS Buried Pipe insitu Regression and Correlation Report Cm-244
FCS Buried Pipe insitu Regression and Correlation Report Co-58
FCS Buried Pipe insitu Regression and Correlation Report Co-60
FCS Buried Pipe insitu Regression and Correlation Report Cs-134
FCS Buried Pipe insitu Regression and Correlation Report Cs-137
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FCS Buried Pipe insitu Regression and Correlation Report H-3
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FCS Buried Pipe insitu Regression and Correlation Report Ni-63
FCS Buried Pipe insitu Regression and Correlation Report Np-237
FCS Buried Pipe insitu Regression and Correlation Report Pu-238
FCS Buried Pipe insitu Regression and Correlation Report Pu-239
FCS Buried Pipe insitu Regression and Correlation Report Pu-240
FCS Buried Pipe insitu Regression and Correlation Report Pu-241
FCS Buried Pipe insitu Regression and Correlation Report Sb-125
FCS Buried Pipe insitu Regression and Correlation Report Sr-90
FCS Buried Pipe insitu Regression and Correlation Report Tc-99

FCS Buried Pipe insitu Uncertainty Report Am-241
FCS Buried Pipe insitu Uncertainty Report C-14
FCS Buried Pipe insitu Uncertainty Report Ce-144
FCS Buried Pipe insitu Uncertainty Report Cm-243
FCS Buried Pipe insitu Uncertainty Report Cm-244
FCS Buried Pipe insitu Uncertainty Report Co-58
FCS Buried Pipe insitu Uncertainty Report Co-60

- FCS Buried Pipe insitu Uncertainty Report Cs-134
- FCS Buried Pipe insitu Uncertainty Report Cs-137
- FCS Buried Pipe insitu Uncertainty Report Eu-152
- FCS Buried Pipe insitu Uncertainty Report Eu-154
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- FCS Buried Pipe insitu Uncertainty Report Fe-55
- FCS Buried Pipe insitu Uncertainty Report H-3
- FCS Buried Pipe insitu Uncertainty Report Ni-59
- FCS Buried Pipe insitu Uncertainty Report Ni-63
- FCS Buried Pipe insitu Uncertainty Report Np-237
- FCS Buried Pipe insitu Uncertainty Report Pu-238
- FCS Buried Pipe insitu Uncertainty Report Pu-239
- FCS Buried Pipe insitu Uncertainty Report Pu-240
- FCS Buried Pipe insitu Uncertainty Report Pu-241
- FCS Buried Pipe insitu Uncertainty Report Sb-125
- FCS Buried Pipe insitu Uncertainty Report Sr-90
- FCS Buried Pipe insitu Uncertainty Report Tc-99

Buried Pipe DSR

- FCS Buried Pipe Excavation DSR 0.15 m
- FCS Buried Pipe Excavation DSR 1.0 m

Less Likely But Plausible Embedded Pipe Drilling Spoils

- FCS LLBP EP Drilling Spoils

Soil Area Factor

- FCS Soil AF 1 DCGL 0.15 m
- FCS Soil AF 2 DCGL 0.15 m
- FCS Soil AF 5 DCGL 0.15 m
- FCS Soil AF 10 DCGL 0.15 m
- FCS Soil AF 100 DCGL 0.15 m
- FCS Soil AF 143 DCGL 0.15 m
- FCS Soil AF 1 DCGL 1 m
- FCS Soil AF 2 DCGL 1 m
- FCS Soil AF 5 DCGL 1 m
- FCS Soil AF 10 DCGL 1 m
- FCS Soil AF 100 DCGL 1 m
- FCS Soil AF 143 DCGL 1 m

Fill DCGL

- FCS Fill DCGL

FORT CALHOUN STATION DECOMMISSIONING PROJECT
LICENSE TERMINATION PLAN

CHAPTER 7
UPDATE OF THE SITE-SPECIFIC DECOMMISSIONING COSTS

REVISION 0

TABLE OF CONTENTS

LIST OF TABLES

ABBREVIATIONS

AIF	Atomic Industrial Forum
CFR	Code of Federal Regulations
DCE	decommissioning cost estimate
DOE	U.S. Department of Energy
FCS	Fort Calhoun Station
GTCC	greater than Class C
ISFSI	Independent Spent Fuel Storage Installation
LLRW	low-level radioactive waste
LTP	License Termination Plan
NESP	National Environmental Studies Project
NRC	U.S. Nuclear Regulatory Commission
OPPD	Omaha Public Power District
PSDAR	Post-Shutdown Decommissioning Activities Report
PWR	pressurized water reactor
SFM	spent fuel management
WCS	Waste Control Specialists LLC
UCF	unit cost factor

7 UPDATE OF THE SITE-SPECIFIC DECOMMISSIONING COSTS

7.1 Introduction

In accordance with 10 CFR 50.82(a)(9)(ii)(F) and Regulatory Guide 1.179, "Standard Format and Contents for License Termination Plans for Nuclear Power Reactors" [1], the updated site-specific cost estimates and funding plans for the Fort Calhoun Station (FCS) decommissioning project are provided. Regulatory Guide 1.179 provides guidance on the details of the information to be presented in the license termination plan (LTP).

The LTP must provide an estimate of the remaining decommissioning costs at the time of LTP submittal and also compare these estimated costs with the present funds set aside for decommissioning. If it is determined that there is a deficit in the present funding, the LTP must indicate the means for ensuring that adequate funds are available to complete the decommissioning.

The decommissioning cost estimate (DCE), at a minimum, needs to include an evaluation of the following cost elements:

- cost assumptions used, including contingency factor
- major decommissioning activities and tasks
- unit cost factors (UCF)
- estimated costs of decontamination and removal of equipment and structures
- estimated costs of waste disposal, including disposal site surcharges
- estimated final status survey (FSS) costs
- estimated total costs

The cost estimate should focus on the remaining work, detailed activity by activity, including costs of labor, materials, equipment, energy, and services. The cost estimate should include the cost of the planned remediation activities as well as the cost of the transportation and disposal of the waste generated by the remedial work conducted.

Historical Perspective

FCS was a 1500 Mwt Combustion Engineering pressurized water reactor (PWR) situated along the Missouri River approximately 20 miles north of the City of Omaha in the vicinity of Fort Calhoun, Nebraska.

On June 16, 2016, the Omaha Public Power District (OPPD) board voted unanimously to shut down FCS, and the plant permanently ceased operation on October 24, 2016. The permanent core offload was completed on November 13, 2016.

The station initially selected the SAFSTOR decommissioning option. In October 2019, the OPPD board authorized FCS to transition from the SAFSTOR decommissioning option into the DECON option, which completes the nuclear decommissioning activities within a significantly shorter time period.

In February 2017, OPPD entered into a five-year agreement for support services with Energy*Solutions*, an international nuclear services company with expertise in all aspects of decommissioning. OPPD retained the license and management responsibility for the facility, while benefitting from the advisory services provided by Energy*Solutions*. In April 2019, a contact providing services using the DECON strategy was awarded to Energy*Solutions*, thereby terminating the prior agreement. The new contract is defined as a collaborative, teamwork approach, blending the decommissioning expertise of Energy*Solutions* with the site-specific knowledge of OPPD staff.

7.2 Decommissioning Cost Estimate

The DCE presented herein represents the cost to complete the remaining decommissioning work as of the end of the 4th quarter 2020. This estimate was prepared based upon the schedule of the remaining work, incorporating the experience that has been gained while performing similar decommissioning tasks over the past several years. To a large extent, this DCE is based upon an existing and operating decommissioning organization, in which actual contracts for services are already in place. As such, there is a high degree of certainty regarding expected work productivity, the cost of labor and the cost of services required to support the remainder of the project. The DCE also includes application of contingency, as a specific provision for unforeseeable elements of cost within the defined project scope. Contingencies are particularly important where previous experience has shown that unforeseeable events, which may increase costs, are likely to occur. The contingency, as used in this estimate, does not account for price escalation and inflation in the costs of decommissioning over the remaining project duration.

The cost estimate was prepared to include all costs associated with the decommissioning and unrestricted release of the FCS site other than the area bounded by the Independent Spent Fuel Storage Installation (ISFSI), and includes radiological decommissioning (i.e., those costs required to accomplish such unrestricted release), spent fuel management (SFM), and site restoration (i.e., non-radiological remediation aimed at leaving the site in a safe and stable condition).

The following subsections present a description of how the cost estimate was prepared and a summary and breakdown of the estimated costs.

7.2.1 Cost Estimate Description and Methodology

Energy*Solutions* maintains a proprietary decommissioning cost model based upon the fundamental technical approach established in AIF/NESP-036, "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates," [2], dated May 1986. The cost model has been updated frequently in accordance with regulatory requirements and industry experience. The cost model includes elements for estimating distributed and undistributed costs. Distributed costs are activity-specific and include planning and preparation costs as well as costs for decontamination, packaging, disposal, and removal of major components and systems. For example, costs for the segmentation, packaging, and disposal of the reactor internals are distributed costs. Undistributed costs, sometimes referred to as collateral costs, are typically time-dependent costs such as utility (Licensee) and decommissioning general contractor (DGC) staff, property taxes, insurance, regulatory fees and permits, energy costs, and security staff.

The methodology for preparing cost estimates for a selected decommissioning alternative requires development of a site-specific detailed work activity sequence based upon the plant inventory. The activity sequence is used to define the labor, material, equipment, energy resources, and duration required for each activity. In the case of major components, individual work sequence activity analyses are performed based on the physical and radiological characteristics of the component, and the packaging, transportation, and disposal options available.

In the case of structures and small components and equipment such as piping, pumps, and tanks, the work durations and costs are calculated based on UCFs. UCFs are economic parameters developed to express costs per unit of work output, piece of equipment, or time. They are developed using decommissioning experience, information on the latest technology applicable to decommissioning, and engineering judgment.

7.2.2 Summary of the Site-Specific Decommissioning Cost Estimate

The overall remaining decommissioning cost (including scope risk contingency) was estimated to be $940 million (in year of expenditure dollars).

The cost estimate includes the costs for license termination costs (corresponding to 10 CFR 50.75(c) requirements), SFM costs (corresponding to 10 CFR 50.54(bb) requirements), and site restoration costs (corresponding to activities such as clean building demolition and site grading). A breakout of the cost for each part of the decommissioning program is provided in Table 7-1.

Table 7-1 Cost for Radiological Decommissioning, Spent Fuel Management, and Site Restoration

License Termination	Spent Fuel Management	Site Restoration	Total
$590,961,242	$306,530,358	$42,252,239	**$939,743,837**

Detailed breakdowns of the estimated costs for license termination, SFM, and site restoration programs are provided in Sections 7.2.3, 7.2.4, and 7.2.5, respectively. Section 7.2.6 presents the estimated contingency costs for each of these programs.

7.2.3 License Termination Costs

Consistent with the NRC definition of decommissioning under 10 CFR 50.75(c), the decommissioning costs under this category consider only those costs associated with normal decommissioning activities necessary for release of the site (other than the ISFSI) for unrestricted use. It does not include costs associated with the disposal of non-radiological materials or structures beyond those necessary to terminate the Part 50 license or the costs associated with construction or operation of an ISFSI.

As summarized in Section 7.2.2 above, the total estimated cost for license termination decommissioning, including contingency, is $591 million.

The remaining decommissioning scope of work included in this estimate is described in detail in other chapters of this LTP. Overall, that work scope includes completion of the removal, transportation and disposal of the major components; completion of the removal, transportation and disposal of the remaining equipment; decontamination and/or bulk demolition of radiological impacted structures and transportation and disposal of the resulting radioactive wastes; performance of the FSS and associated license termination activities. The estimated costs include the labor, equipment, materials, services and fees needed to conduct the work. The estimated cost also includes all of the program support activities and services necessary to manage and safely carry out a large scale dismantlement and demolition project. These program support activities include project management, work controls and site administration; technical support services, such as radiation protection, safety, engineering, security, QA/QC, environmental monitoring, waste management and decommissioning subject matter experts needed to support the project.

A high-level breakdown of the estimated cost by phase is provided in Table 7-2.

Table 7-2 DECON Cost Summary by Phase

Period	Item	Description	Estimate Cost	Contingency	Estimated Total
Period 1 Total – Transition through SNF Pool to Pad			**$1,768**	**$177**	**$1,945**
1a Total – SAFSTOR & Transition			0	0	0
1b Total – Spent Fuel Pool to Pad			$1,768	$177	$1,945
Period 2 Total – Decommissioning & License Termination			**$564,631**	**$64,824**	**$629,454**
2a Total – Decommissioning Planning & Transition			$4,562	$456	$5,018
2b Total – Rad Decommissioning & License Termination			$304,348	$38,795	$343,143
2c – Site Restoration			$29,652	$2,965	$32,618
2d – Decomissioning Undistributed Costs			$226,068	$22,607	$248,675
Period 3 Total – SNF/GTCC Dry Storage & Disposition			**$277,279**	**$31,065**	**$308,3545**
3a – Dry Fuel Storage			$260,668	$28,573	$289,241
3b – Fuel Transfer & ISFSI Decommissioning			$16,612	$2,492	$19,104
Grand Total			$843,678	$96,066	$939,744

Note: A more detailed breakdown of the costs by resource and by decommissioning activity are provided in Table 7-7. Values presented in the table are in thousands of 2020 dollars.

The total estimated cost for radioactive waste disposition (containers, transportation and disposal) is $84.1 million. As presented in Table 7-7, these waste management costs are comprised of four distinct categories: Class A Large Components, Class B/C Waste, Class A Containerized Wastes, and Class A Bulk Materials. Costs for on-site handling of greater than Class C (GTCC) waste (i.e., reactor vessel internals) are included in the "Major Component Removal" category shown on Table 7-7. However, no costs for disposal of this waste is included in the estimate, as it is assumed that disposal of this waste will be included as a part of spent fuel disposition.

Waste management costs comprise a significant portion of the decommissioning cost estimate. Additionally, limited future access to disposal sites licensed for receipt of Class B and C wastes introduces a significant level of uncertainty with respect to the appropriateness of using existing rate structures to estimate disposal costs of these wastes. Energy*Solutions*' approach to estimating waste disposal costs is discussed in the following paragraphs.

7.2.3.1 Waste Classification

Regulations governing disposal of radioactive waste are stringent in order to ensure control of the waste and preclude adverse impact on public health and safety. At present, low-level radioactive waste (LLRW) disposal is controlled by 10 CFR 61, which went into effect in December 1983. This regulation stipulates the criteria for the establishment and operation of shallow-land LLRW burial facilities. Embodied within this regulation are criteria and classifications for packaging LLRW such that it is acceptable for burial at licensed LLRW disposal sites.

For each waste classification, 10 CFR 61 stipulates specific criteria for physical and chemical properties that the LLRW must meet in order to be accepted at a licensed disposal site. The LLRW disposal criteria of 10 CFR 61 require that LLRW generators determine the proportional amount of a number of specific radioactive isotopes present in each container of disposable LLRW. This requirement for isotopic analysis of each container of disposable LLRW is met by employing a combination of analytical techniques such as computerized analyses based upon scaling factors, sample laboratory analyses, and direct assay methods. Having performed an isotopic analysis of each container of disposable LLRW, the waste must then be classified according to one of the classifications (Class A, B, C, or GTCC waste) as defined in 10 CFR 61.

Energy*Solutions*' classification of LLRW resulting from decommissioning activities is based on AIF/NESP-036, NUREG/CR-0130, "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station" [3], plant-specific information, and recent industry experience. The estimated curie content of the reactor vessel and internals at shutdown is typically derived from NUREG/CR-0130 for PWRs and NUREG/CR-0672, "Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station" [4], for boiling-water reactors, and adjusted for the different mass of components and period of decay.

A summary of waste disposal volumes by waste classification is provided in Table 7-3.

Table 7-3 Waste Disposal Volumes by Waste Classification

Waste Class	Volume (cf)
Class A Waste	2,497,082
Class B & C Waste	1,666
GTCC Waste	1,180

Table 7-6 provides a waste disposal summary itemized by volumes, waste form, waste class, weight, volume and costs for packaging, transportation, and disposal.

7.2.3.2 Packaging

Selection of the type and quantity of containers required for Class B and C wastes is based on the most restrictive of either Curie-content, dose-rate, container weight limit, or container volume limit. Residual spent fuel and GTCC wastes from segmentation of the reactor vessel internals and other accident related waste is normally packaged in modified spent fuel canisters and this packaging is assumed for the DCE. The selection of container type for Class A waste is based on the transportation mode (i.e., rail or truck) and waste form. The quantity of Class A waste containers is determined by the most restrictive of either container weight limit or container volume limit. Large components, such as steam generators, pressurizers, and reactor recirculation pumps, are shipped as their own containers with additional shielding as required.

Container costs are obtained from manufacturers specializing in the design and fabrication of storage containers for nuclear materials. Shielded transport cask and liner costs are obtained from the cask owners and operators.

7.2.3.3 Transportation

Transportation routes to processing and disposal facilities are determined based on available transportation modes (truck, rail, or combinations). Transportation costs for the selected routes and modes are obtained from vendor quotes or published tariffs whenever possible.

7.2.3.4 Class A Disposal Options and Rates

Class A waste that meets the waste acceptance criteria are to be disposed of at Energy*Solutions*' LLRW disposal facility in Clive, Utah. All reported waste disposal costs include packaging, transportation, and any applicable surcharges.

7.2.3.5 Class B and C Disposal Options and Rates

Currently, within the United States, there are only three operational commercial near-surface disposal facilities licensed to accept Class B and C LLRW: the Barnwell facility, operated by Energy*Solutions* in Barnwell, South Carolina; the U.S. Ecology facility in Richland, Washington; and the facility in Andrews County, Texas, operated by Waste Control Specialists

(WCS). Barnwell only accepts waste from states within the Atlantic Compact, and U.S. Ecology only accepts waste from states within the Northwest and Rocky Mountain Compacts. However, the WCS facility will accept waste from the Texas Compact (comprised of Texas and Vermont) and from non-compact generators. The Texas Compact Commission on March 23, 2012, approved amendments to rules allowing the import of non-compact generator LLRW for disposal at the WCS Andrews County facility.

Transportation costs in this estimate for the Class B and C waste are based on a distance of 900 miles one way from FCS to the WCS facility.

7.2.3.6 GTCC

Wastes identified as 10 CFR 61 Class A, B, and C may be disposed of at near-surface disposal facilities. Certain components are highly activated and may exceed the radionuclide concentration limitations for 10 CFR 61 Class C waste. In accordance with 10 CFR 61, these components, which are referred to as GTCC wastes, cannot be disposed of in a near-surface LLRW disposal facility and must be transferred to a geologic repository or a similar site approved by the NRC. Highly activated sections of the reactor vessel internals and certain decommissioning processes will result in GTCC waste. Presently, a facility does not exist for the disposal of wastes exceeding 10 CFR 61 Class C limitations. Energy*Solutions* assumes that the U.S. Department of Energy (DOE) will accept this waste along with dry fuel. Although courts have held that the DOE is obligated to accept and dispose of GTCC waste, issues regarding potential costs remain unsettled. Therefore, Energy*Solutions* conservatively estimates a GTCC waste disposal cost.

7.2.3.7 LLRW Volume Reduction

Because current Class A LLRW disposal rates are significantly lower than LLRW volume reduction rates, Energy*Solutions* does not assume on-site volume reduction techniques such as waste compaction or an aggressive decontamination, survey, and release effort.

7.2.3.8 Non-Radioactive Non-Hazardous Waste Disposal

Energy*Solutions* assumes that recyclable, non-radioactive scrap metal resulting from the decommissioning program will be sold to a scrap metal dealer.

7.2.3.9 Hazardous and Industrial Waste Disposal

Uncontaminated lead shielding remaining after shutdown was assumed to be removed from its installed locations and shipped off-site by entities having a need for the material. The entities will receive the lead at no charge in return for providing the removal and shipping services. Non-radioactive contaminated surfaces coated with tightly adhering and undamaged lead-based paint will be removed as non-hazardous building demolition debris. All other chemicals and hazardous materials present at shutdown will be removed and properly disposed of during decommissioning.

7.2.4 Spent Fuel Management Costs

OPPD acknowledges that the costs to construct and operate an ISFSI and other SFM costs are not considered by the NRC staff as part of decommissioning costs. Nevertheless, as there is significant interest by many stakeholders in these costs, they are presented herein. As presented in Section 7.2.2 above, the estimated cost for the anticipated work scope is $306 million.

Overall, the SFM work scope includes the design and procurement of GTCC casks, fuel pool operations, ISFSI, and operation of the ISFSI until termination of the reactor license.

Construction of the ISFSI was completed in October 2018, and spent fuel transfer operations were complete May 2020.

The estimated costs include the labor, equipment, materials, services, fees, and program support activities necessary to safely manage the spent nuclear fuel. The cost summary by cost type is presented in Table 7-4.

Table 7-4 DECON Cost Summary by Cost Type

Item	Cost
Labor Cost	$355,986
Material & Equipment Cost	$51,487
Waste Packaging & Transportation	$28,910
Waste Disposal	$138,268
Subcontracts	$136,613
Other Direct Costs	$132,414
Contingency	$96,066
Total Cost	**$939,744**

Note: Cost values are in thousands of 2020 dollars.

A more detailed breakdown of the cost by resource requirements (e.g., labor, materials, and services) is provided in Table 7-8.

7.2.5 Site Restoration Costs

The estimated cost for the anticipated work scope for site restoration is $42.2 million. Overall, that work scope includes clean building demolition, backfilling of any open excavations or void spaces, non-radiological environmental remediation, and final grading and stabilization against erosion.

The estimated costs include the labor, equipment, materials, services, and fees needed to conduct the work. In general, most of this work is anticipated to be performed by contractors; however, the estimated cost also includes all of the program support activities and services necessary to manage and safely carry out project.

A breakdown of the estimated site restoration cost is provided in Table 7-5.

7.2.6 Contingency

Contingencies are applied to cost estimates primarily to allow for unknown or unplanned occurrences during the actual program (e.g., increased radioactive waste materials volumes over that expected, equipment breakdowns, weather delays, and labor strikes). This is consistent with the definition provided in the DOE Cost Estimating Guide, DOE G 430.1-1, 3-28-97 (DOE G) [5]. Contingency "covers costs that may result from incomplete design, unforeseen and unpredictable conditions, or uncertainties within the defined project scope. The amount of contingency will depend on the status of design, procurement, construction, and the complexity and uncertainties of the component parts of the project. Contingency is not to be used to avoid making an accurate assessment of expected costs." Energy*Solutions* determines site-specific contingency factors to be applied to each estimate based on industry practices.

The DOE has established a recommended range of contingencies as a function of completeness of program design, DOE G. The ranges are:

Type of Estimate	Contingency Range as a % of Total Estimate
Planning Phase Estimate	20-30
Budget Estimate	15-25
Title I (Preliminary Design Estimate)	10-20
Title II (Definitive Design Estimate)	5-15

The FCS DECON scenario has had substantial development and the Fuel Pool to Pad program has been completed. For the purposes of this estimate, we have considered this to be a budget estimate designed to provide sufficient information to OPPD to assess its financial obligations for decommissioning FCS. It is not a detailed budget, but rather a financial analysis prepared in advance of the detailed execution planning and budgeting work required to carry out the decommissioning.

A reactor decommissioning program will be conducted under an NRC-approved Quality Assurance Program which meets the requirements of 10 CFR 50, Appendix B. However, the development of the quality assurance program, the performance of work under that program, and the effort required to ensure compliance with the program, is already included in the detailed cost estimate. Therefore, Energy*Solutions* does not include quality assurance as an element of the contingency allowance. The same is true for contamination. Where radioactive contamination or activated materials are dealt with, the cost factors and associated calculations fully reflect the cost impact of that material, and a separate contingency is not required specifically due to working with contamination.

7.3 <u>Decommissioning Funding Plan</u>

As required by the NRC, OPPD maintains an external trust fund to accumulate moneys for the decommissioning of FCS. OPPD began its decommissioning accrual and funding in July 1983, which moved to an NRC required fund in 1990 (“1990 Plan”). The market value of the 1990 Plan’s decommissioning fund was $337.8 million as of December 31, 2020.

In 1992, OPPD began accumulating funds in a separate decommissioning fund based on the difference between the site-specific study's estimated cost to fully decommission FCS and the NRC's regulated formula based cost to decommission the radiated portions of FCS ("1992 Plan"). OPPD began to add additional funds to the 1992 Plan in 2017 and expects to continue until 2024. OPPD contributed an additional $112.6 million to the 1992 Plan in 2020, and paid expenses of $149.2 million. The market value of the 1992 Plan's decommissioning fund was $204.3 million as of December 31, 2020.

7.4 References

[1] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.179, Revision 2, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," 2019.

[2] "Atomic Energy Forum/National Environmental Studies Project, AIF/NESP-036, "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates" 1986".

[3] U.S. Nuclear Regulatory Commission, "NUREG/CR-0130, Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," 1978.

[4] U.S. Nuclear Regulatory Commission, "NUREG/CR-0672, Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station," 1980.

[5] U.S. Department of Energy, "DOE G 430.1-1, Cost Estimating Guide," 1997.

Table 7-5 Estimated Site Restoration Cost

Period	Item Number	Item Description	Labor Cost	Materials & Equipment	Waste Packaging & Transportation	Waste Disposal	Sub-contracts	Other Direct Cost (ODC)	Contingency	Total Cost (Estimated)	License Termination	Spent Fuel Management	Site Restoration	Man-Hours
2c	**2c.1.1**	Asbestos Abatement					1,427		143	1,570	0	-	1,570	
2c	**2c.1.2**	Turbine Building					4,714		471	5,185	-	-	5,185	
2c	**2c.1.3**	Intake Structure					3,687		369	4,056	-	-	4,056	
2c	**2c.1.4**	Admin Building (not removed)					750		75	825	-	-	825	
2c	**2c.1.5**	Flex Bldg. (not removed)					-		-	-	-	-	-	
2c	**2c.1.6**	Mausoleum					115		12	127	-	-	127	
2c	**2c.1.7**	Training Center (not removed)					-		-	-	-	-	-	
2c	**2c.1.8**	New & Old Warehouse Demo					365		36	401	-	-	401	
2c	**2c.1.9**	Misc Clean Building Demo					1,151		115	1,266	-	-	1,266	
2c	**2c.1.10**	Underground Piping & Utilities					4,836		484	5,320	-	-	5,320	
2c	**2c.1.11**	Yard/Parking lot pavement & concrete					2,038		204	2,242	-	-	2,242	
2c	**2c.1.12**	MET Tower					-		-	-	0	-	-	
2c	**2c.1.13**	D&D Contractor Staff (incl with work)						-	-	-	-	-	-	
2c	**2c.1.14**	D&D Contractor Tools & Equipment (incl with work)						-	-	-	-	-	-	
	2c.1	**Clean Building Demolition**	**-**	**-**	**-**	**-**	**19,084**	**-**	**1,908**	**20,992**	**0**	**-**	**20,992**	**0**
2c	**2c.2.1**	Process / Reuse On-Site Fill Materials					6,342		634	6,976	-	-	6,976	
2c	**2c.2.2**	Imported Fill Materials		1,730					173	1,903	-	-	1,903	
	2c.2	**Fill & Backfill Materials**	**-**	**1,730**	**-**	**-**	**6,342**	**-**	**807**	**8,879**	**-**	**-**	**8,879**	**0**
2c	**2c.3.1**	Backfill & Grade Buildings					715		72	787	-	-	787	

	2c.3	**Backfill & Grade Buildings**	**-**	**-**	**-**	**-**	**715**	**-**	**72**	**787**	**-**	**-**	**787**	**0**
2c	**2c.4.1**	Non-Rad Environmental Remediation					1,205		121	1,326	-	-	1,326	
	2c.4	**Non-Rad Environmental Remediation**	**-**	**-**	**-**	**-**	**1,205**	**-**	**121**	**1,326**	**-**	**-**	**1,326**	**0**
2c	**2c.5.1**	Final Site Grading					577		58	634	-	-	634	
	2c.5	**Final Site Restoration**	**-**	**-**	**-**	**-**	**577**	**-**	**58**	**634**	**-**	**-**	**634**	**0**
2c - Site Restoration			**-**	**1,730**	**-**	**-**	**27,923**	**-**	**2,965**	**32,618**	**0**	**-**	**32,618**	**0**

Note: Cost values are in thousands of 2020 dollars.

Table 7-6 Waste Disposal Summary by Volume

Facility	Waste Form	Waste Class	Waste Weight (LBs)	Waste Volume (CF)	Burial Volume (CF)	Packaging & Transporation	Burial Cost	Total Waste Cost
Rad Waste								
WCS	Cask	B/C	161,178	1,791	1,666	$1,599	$20,819	$22,418
Clive	CWF	A	422,708	9,682	11,613	$2,524	$3,360	$5,884
Clive	Debris	A	139,648,601	2,432,698	2,441,460	$13,611	$102,892	$116,503
Clive	Large Component	A	2,604,487	26,650	28,132	$7,898	$6,427	$14,325
Clive	Legacy Components	A			15,878	$3,279	$3,121	$6,400
			142,836,973	**2,470,821**	**2,498,748**	**$28,910**	**$136,619**	**$165,529**
Other								
Local Land	Clean/Exempt	F	1,982,766	72,184	72,200		$3,070	$3,485
Local Recycled	Recycled Metals	F	24,497,545	1,224,877	1,225,692		$(1,802)	$(1,802)
On Site	Clean/Exempt	F	182,542,639	2,028,252	2,028,400		-	-
			209,022,949	**3,325,313**	**3,326,292**		**$1,268**	**$1,684**
Grand Total			**351,859,923**	**5,796,134**	**5,825,040**	**$28,910**	**$137,887**	**$167,213**

Note: Cost values are in thousands of 2020 dollars.

Table 7-7 Estimated Cost to Complete Summary

Period	Item Number	Item Description	Labor Cost	Materials & Equipment	Waste Packaging & Transportation	Waste Disposal	Sub-contracts	Other Direct Cost (ODC)	Contingency	Total Cost (Estimated)	License Termination	Spent Fuel Management	Site Restoration	Man-Hours
1a										-				
	1a.1	**Planning & Procedures (completed)**	-	-	-	-	-	-	-	-	-	-	-	**0**
1a	**1a.2.1**	ISFSI Design & Engineering						-	-	-	-	-	-	
1a	**1a.2.2**	ISFSI Construction						-	-	-	-	-	-	
	1a.2	**ISFSI Design & Construction**	-	-	-	-	-	-	-	-	-	-	-	**0**
1a	**1a.3.1**	ISFSI Security Modifications								-				
										-				
	1a.3	**ISFSI Security Modifications (completed)**	-	-	-	-	-	-	-	-	-	-	-	**0**
1a	**1a.4.1**	Perform Fuel Sampling & Analysis								-				
1a	**1a.4.2**	Fuel Pool Operations & Maintenance								-				
	1a.4	**Fuel Pool Operations (completed)**	-	-	-	-	-	-	-	-	-	-	-	**0**
1a Total - SAFSTOR & Transition			-	-	-	-	-	-	-	-	-	-	-	**0**
1b	**1b.1.1**	ISFSI & Plant Security Staff	-						-	-	-	-	-	0
	1b.1	**Security through Fuel Transfer**	-	-	-	-	-	-	-	-	-	-	-	**0**
1b	**1b.2.1**	Design & Procure Fuel Casks		-					-	-	-	-	-	
1b	**1b.2.2**	Design & Procure GTCC Casks		560					56	616	-	616	-	
1b	**1b.2.3**	Cask Vendor Tech Support (incl with procurement)						-	-	-	-	-	-	
	1b.2	**Design & Procure Fuel & GTCC Casks**	-	**560**	-	-	-	-	**56**	**616**	-	**616**	-	**0**

1b	**1b.3.1**	Spent Fuel Operations Support						-	-	-	-	-	-	
1b	**1b.3.2**	Fuel Pool Water Processing						735	74	809	-	809	-	
1b	**1b.3.3**	Fuel Pool Clean-up						473	47	520	-	520	-	
	1b.3	**Fuel Pool Operations**	**-**	**-**	**-**	**-**	**-**	**1,208**	**121**	**1,329**	**-**	**1,329**	**-**	**0**
1b	**1b.4.1**	Procure Transport/Transfer Equipment						-	-	-	-	-	-	
1b	**1b.4.2**	Spent Fuel Transfer (includes equipment)					-		-	-	-	-	-	
	1b.4	**Fuel Transfer Operations**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**0**
1b	**1b.5.1**	Utility Project Staff	-						-	-	-	-	-	0
1b	**1b.5.2**	ISFSI Operations & Maintenance (incl with Utility staff)						-	-	-	-	-	-	
	1b.5	**Utility Staff through Fuel Transfer**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**0**
1b Total - Spent Fuel Pool to Pad			**-**	**560**	**-**	**-**	**-**	**1,208**	**177**	**1,945**	**-**	**1,945**	**-**	**0**
Period 1 Total - Transition through SNF Pool to Pad			**-**	**560**	**-**	**-**	**-**	**1,208**	**177**	**1,945**	**-**	**1,945**	**-**	**0**
2a	**2a.1.1**	Historic Site Assessment (HSA)						-	-	-	-	-	-	
2a	**2a.1.2**	Environmental Assessments						-	-	-	-	-	-	
	2a.1	**Site Assessments (completed)**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**0**
2a	**2a.2.1**	LTP Preparation						536	54	590	590	-	-	
	2a.2	**LTP Planning & Preparation**	**-**	**-**	**-**	**-**	**-**	**536**	**54**	**590**	**590**	**-**	**-**	**0**
2a	**2a.3.1**	General Planning Services (incl with Utility Staff)						-	-	-	-	-	-	
2a	**2a.3.2**	Site Surveys & Characterization	138	35					17	190	190	-	-	2,129
2a	**2a.3.3**	Procurement Planning (incl with Utility Staff)						-	-	-	-	-	-	
2a	**2a.3.4**	Baseline Development	-					-	-	-	-	-	-	15,260
	2a.3	**Planning Services**	**138**	**35**	**-**	**-**	**-**	**-**	**17**	**190**	**190**	**-**	**-**	**17,389**
2a	**2a.4.1**	Road Upgrades						-	-	-	-	-	-	

2a	**2a.4.2**	Rail Upgrades					3,853		385	4,238	4,238	-	-	
	2a.4	**Infrastructure Upgrades**	**-**	**-**	**-**	**-**	**3,853**	**-**	**385**	**4,238**	**4,238**	**-**	**-**	**0**
2a	**2a.5.1**	Cold & Dark Operations					-		-	-	-	-	-	
2a	**2a.5.2**	Temporary Power Upgrades (incl with cold & dark)						-	-	-	-	-	-	
	2a.5	**Decommissioning Electrical**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**-**	**0**
2a Total - Decommissioning Planning & Transition			**138**	**35**	**-**	**-**	**3,853**	**536**	**456**	**5,018**	**5,018**	**-**	**-**	**17,389**
2b	**2b.1.1**	Develop RVI/RV Plans and Procedures	194					104	30	328	328	-	-	2,000
2b	**2b.1.2**	Procure RVI Equipment		16,780					1,678	18,458	18,458	-	-	
2b	**2b.1.3**	RVI Equipment Tech Support					2,468		247	2,714	2,714	-	-	
2b	**2b.1.4**	RVI Segmentation & Removal					8,611		861	9,472	9,472	-	-	
2b	**2b.1.5**	RVI GTCC Operations & Loading (incl with RVI Removal)						-	-	-	-	-	-	
2b	**2b.1.6**	Procure RV Equipment		16,410					1,641	18,051	18,051	-	-	
2b	**2b.1.7**	RV Equipment Tech Support					432		43	475	475	-	-	
2b	**2b.1.8**	RV Segmentation & Removal					3,974		397	4,372	4,372	-	-	
	2b.1	**Reactor Vessel**	**194**	**33,190**	**-**	**-**	**15,484**	**104**	**4,897**	**53,870**	**53,870**	**-**	**-**	**2,000**
2b	**2b.2.1**	Heavy Lift / Transfer Equipment (incl with steam generator)					-	-	-	-	-	-		
2b	**2b.2.2**	Remaining Legacy Large Component Removal					-		-	-	-	-	-	
2b	**2b.2.3**	Steam Generator Removal					4,289		429	4,718	4,718	-	-	
2b	**2b.2.4**	Pressurizer Removal					72		7	79	79	-	-	
2b	**2b.2.5**	RCP Removal					287		29	315	315	-	-	
	2b.2	**Large Component Removal**	**-**	**-**	**-**	**-**	**4,647**	**-**	**465**	**5,112**	**5,112**	**-**	**-**	**0**
2b	**2b.3.1**	Asbestos Abatement					2,349		235	2,584	2,584	-	-	
2b	**2b.3.2**	Chemical Decon (incl with work)					-		-	-	-	-	-	
2b	**2b.3.3**	Reactor Bldg. Interior / Systems Demo					3,624		362	3,987	3,987	-	-	
2b	**2b.3.4**	Aux Bldg. Interior / Systems Demo					3,506		351	3,856	3,856	-	-	

2b	**2b.3.5**	Rad Waste Bldg. Interior Demo					1,812		181	1,993	1,993	-	-	
2b	**2b.3.6**	Misc Rad Systems Removal					7,149		715	7,864	7,864	-	-	
2b	**2b.3.7**	D&D Contractor Staff					1,812		181	1,993	1,993	-	-	
2b	**2b.3.8**	D&D Contractor Tools & Equipment (incl with work)					-		-	-	-	-	-	
2b	**2b.3.9**	D&D Contractor Scaffolding (incl with work)					-		-	-	-	-	-	
	2b.3	**Rad Bldg. Interior / Systems D&D**	**-**	**-**	**-**	**-**	**20,252**	**-**	**2,025**	**22,278**	**22,278**	**-**	**-**	**0**
2b	**2b.4.1**	Reactor Building Demo					8,604		860	9,464	9,464	-	-	
2b	**2b.4.2**	Aux Building Demo					10,755		1,076	11,831	11,831	-	-	
2b	**2b.4.3**	Rad Waste Bldg. Interior Demo					2,151		215	2,366	2,366	-	-	
2b	**2b.4.4**	Misc Rad Building & Open Air demo					6,663		666	7,329	7,329	-	-	
2b	**2b.4.5**	Firing Range (included)					-		-	-	-	-	-	
2b	**2b.4.6**	Drainage Lagoons					458		46	504	504	-	-	
2b	**2b.4.7**	D&D Contractor Staff					-		-	-	-	-	-	
2b	**2b.4.8**	D&D Contractor Tools & Equipment (incl with work)					-		-	-	-	-	-	
2b	**2b.4.9**	D&D Contractor Scaffolding					-		-	-	-	-	-	
	2b.4	**Rad Building Open Air Demolition**	**-**	**-**	**-**	**-**	**28,631**	**-**	**2,863**	**31,494**	**31,494**	**-**	**-**	**0**
2b	**2b.5.1**	Waste Handling & Loadout Equipment (incl with waste handling)					-	-	-	-	-	-		
2b	**2b.5.2**	Waste Handling & Loadout					10,162		1,016	11,179	11,179	-	-	
2b	**2b.5.3**	Waste Rad Techs (incl with waste handling)						-	-	-	-	-	-	
	2b.5	**Field Waste Operations**	**-**	**-**	**-**	**-**	**10,162**	**-**	**1,016**	**11,179**	**11,179**	**-**	**-**	**0**
2b	**2b.6.1**	Waste Packaging & Transportation			28,910				4,337	33,247	33,247	-	-	
2b	**2b.6.2**	Class A Waste Disposal				115,800			17,370	133,170	133,170	-	-	
2b	**2b.6.3**	Class B & C Waste Disposal				20,819			3,123	23,942	23,942	-	-	
2b	**2b.6.4**	Remaining Legacy Large			-	-			-	-	-	-	-	

		Component Disposal												
	2b.6	**Rad Waste Transportation & Disposal**	**-**	**-**	**28,910**	**136,619**	**-**	**-**	**24,829**	**190,358**	**190,358**	**-**	**-**	**0**
2b	**2b.7.1**	Recycle Materials						(1,802)	(270)	(2,072)	(2,072)	-	-	
2b	**2b.7.2**	Non- Rad Local Landfill						3,485	523	4,008	4,008	-	-	
	2b.7	**Non-Rad & Hazardous Waste Disposal**	**-**	**-**	**-**	**-**	**-**	**1,684**	**253**	**1,936**	**1,936**	**-**	**-**	**0**
2b	**2b.8.1**	Rad Protection Techs					7,984		798	8,782	8,782	-	-	
2b	**2b.8.2**	Rad Protection Techs	6,261						626	6,887	6,887	-	-	96,316
	2b.8	**Field Radiation Protection**	**6,261**	**-**	**-**	**-**	**7,984**	**-**	**1,424**	**15,669**	**15,669**	**-**	**-**	**96,316**
2b	**2b.9.1**	Final Site Surveys	6,135	2,045				2,045	1,022	11,247	11,247	-	-	94,377
	2b.9	**Field Rad Surveys & Final Site Surveys**	**6,135**	**2,045**	**-**	**-**	**-**	**2,045**	**1,022**	**11,247**	**11,247**	**-**	**-**	**94,377**
2b Total - Rad Decommissioning & License Term			**12,589**	**35,235**	**28,910**	**136,619**	**87,162**	**3,833**	**38,795**	**343,143**	**343,143**	**-**	**-**	**192,693**
2c	**2c.1.1**	Asbestos Abatement					1,427		143	1,570	0	-	1,570	
2c	**2c.1.2**	Turbine Building					4,714		471	5,185	-	-	5,185	
2c	**2c.1.3**	Intake Structure					3,687		369	4,056	-	-	4,056	
2c	**2c.1.4**	Admin Building (not removed)					750		75	825	-	-	825	
2c	**2c.1.5**	Flex Bldg. (not removed)					-		-	-	-	-	-	
2c	**2c.1.6**	Mausoleum					115		12	127	-	-	127	
2c	**2c.1.7**	Training Center (not removed)					-		-	-	-	-	-	
2c	**2c.1.8**	New & Old Warehouse Demo					365		36	401	-	-	401	
2c	**2c.1.9**	Misc Clean Building Demo					1,151		115	1,266	-	-	1,266	
2c	**2c.1.10**	Underground Piping & Utilities					4,836		484	5,320	-	-	5,320	
2c	**2c.1.11**	Yard/Parking lot pavement & concrete					2,038		204	2,242	-	-	2,242	
2c	**2c.1.12**	MET Tower					-		-	-	0	-	-	
2c	**2c.1.13**	D&D Contractor Staff (incl with work)						-	-	-	-	-	-	
2c	**2c.1.14**	D&D Contractor Tools & Equipment (incl with work)						-	-	-	-	-	-	
	2c.1	**Clean Building Demolition**	**-**	**-**	**-**	**-**	**19,084**	**-**	**1,908**	**20,992**	**0**	**-**	**20,992**	**0**
2c	**2c.2.1**	Process / Reuse On-Site Fill Materials					6,342		634	6,976	-	-	6,976	

2c	**2c.2.2**	Imported Fill Materials		1,730					173	1,903	-	-	1,903	
	2c.2	**Fill & Backfill Materials**	**-**	**1,730**	**-**	**-**	**6,342**	**-**	**807**	**8,879**	**-**	**-**	**8,879**	**0**
2c	**2c.3.1**	Backfill & Grade Buildings					715		72	787	-	-	787	
	2c.3	**Backfill & Grade Buildings**	**-**	**-**	**-**	**-**	**715**	**-**	**72**	**787**	**-**	**-**	**787**	**0**
2c	**2c.4.1**	Non-Rad Environmental Remediation					1,205		121	1,326	-	-	1,326	
	2c.4	**Non-Rad Environmental Remediation**	**-**	**-**	**-**	**-**	**1,205**	**-**	**121**	**1,326**	**-**	**-**	**1,326**	**0**
2c	**2c.5.1**	Final Site Grading					577		58	634	-	-	634	
	2c.5	**Final Site Restoration**	**-**	**-**	**-**	**-**	**577**	**-**	**58**	**634**	**-**	**-**	**634**	**0**
2c - Site Restoration			**-**	**1,730**	**-**	**-**	**27,923**	**-**	**2,965**	**32,618**	**0**	**-**	**32,618**	**0**
2d	**2d.1.1**	Utility Management Staff	68,854						6,885	75,739	72,710	-	3,030	774,280
2d	**2d.1.2**	Engineering (incl with utility staff)						-	-	-	-	-	-	
2d	**2d.1.3**	Work Control / Document Control (incl with utility staff)						-	-	-	-	-	-	
2d	**2d.1.4**	Office Equipment, Misc Expenses						-	-	-	-	-	-	
2d	**2d.1.5**	Temporary Facilities						2,632	263	2,895	2,780	-	116	
2d	**2d.1.6**	Environmental Program						1,051	105	1,156	1,109	-	46	
2d	**2d.1.7**	Decommissioning NRC Fees & Inspections						5,253	525	5,778	5,778	-	-	
2d	**2d.1.8**	Regulatory Affairs Program (incl with utility staff)						-	-	-	-	-	-	
2d	**2d.1.9**	Characterization Program (incl with utility & contractor staff)					-	-	-	-	-	-		
2d	**2d.1.10**	Waste Management (incl with utility & contractor staff)						-	-	-	-	-	-	
2d	**2d.1.11**	Quality Assurance Program						-	-	-	-	-	-	
2d	**2d.1.12**	Safety Program & Supplies		2,627					263	2,889	2,774	-	116	
2d	**2d.1.13**	Training Program *(In processing Time)*					3,690		369	4,059	3,897	-	162	
	2d.1	**Utility Staff during Decommissioning**	**68,854**	**2,627**	**-**	**-**	**3,690**	**8,936**	**8,411**	**92,517**	**89,047**	**-**	**3,470**	**774,280**

2d	**2d.2.1**	Plant Operations & Maintenance *(non-labor)*		4,000					400	4,400	4,224	-	176	
2d	**2d.2.2**	Water Processing						714	71	786	754	-	31	
2d	**2d.2.3**	Utility Costs						1,103	110	1,213	1,165	-	49	
	2d.2	**Plant Operations & Maintenance**	**-**	**4,000**	**-**	**-**	**-**	**1,817**	**582**	**6,399**	**6,143**	**-**	**256**	**0**
2d	**2d.3.1**	Plant Security	16,668						1,667	18,335	17,601	-	733	127,400
	2d.3	**Plant Security during Decommissioning**	**16,668**	**-**	**-**	**-**	**-**	**-**	**1,667**	**18,335**	**17,601**	**-**	**733**	**127,400**
2d	**2d.4.1**	DGC Contractor Mobilization						525	53	578	555	-	23	
2d	**2d.4.2**	Decommissioning General Contractor Staff	47,199					31,466	7,867	86,532	83,071	-	3,461	306,600
2d	**2d.4.3**	Regulatory Affairs Program *(incl with DGC staff)*						-	-	-	-	-	-	
2d	**2d.4.4**	Contractor Rad Protection Program		4,675					467	5,142	4,936	-	206	
2d	**2d.4.5**	Characterization Program *(incl with DGC staff)*						-	-	-	-	-	-	
2d	**2d.4.6**	Waste Management *(incl with DGC staff)*						-	-	-	-	-	-	
2d	**2d.4.7**	DGC Contractor Demobilization						525	53	578	555	-	23	
	2d.4	**Contractor Project Management**	**47,199**	**4,675**	**-**	**-**	**-**	**32,517**	**8,439**	**92,830**	**89,117**	**-**	**3,713**	**306,600**
2d	**2d.5.1**	ANI Insurance Costs (post shutdown)						-	-	-	-	-	-	
2d	**2d.5.2**	ANI Insurance Costs (post fuel transfer)						3,168	317	3,485	3,345	-	139	
2d	**2d.5.3**	NEIL Insurance Costs (post shutdown)						-	-	-	-	-	-	
2d	**2d.5.4**	NEIL Insurance Costs (post fuel transfer)						1,848	185	2,033	2,033	-	-	
2d	**2d.5.5**	Iowa Property Tax						120	12	132	127	-	5	
2d	**2d.5.6**	Nebraska Use Tax (5.5%)						2,000	200	2,200	2,112	-	88	
	2d.5	**Insurance and Taxes**	**-**	**-**	**-**	**-**	**-**	**7,136**	**714**	**7,850**	**7,617**	**-**	**233**	**0**
2d	**2d.6.1**	Corporate A&G Allocations						11,926	1,193	13,119	12,594	-	525	
2d	**2d.6.2**	Communications (Telecom) Allocations						1,224	122	1,346	1,293	-	54	

2d	**2d.6.3**	Information Technology Allocations						14,400	1,440	15,840	15,206	-	634	
2d	**2d.6.4**	Stores Expense Adder						400	40	440	422	-	18	
	2d.6	**Corporate Allocations**	**-**	**-**	**-**	**-**	**-**	**27,950**	**2,795**	**30,745**	**29,515**	**-**	**1,230**	**0**
2d - Decommissioning Undistributed Costs			**132,721**	**11,301**	**-**	**-**	**3,690**	**78,356**	**22,607**	**248,675**	**239,041**	**-**	**9,635**	**1,208,280**
Period 2 Total - Decommissioning & License Termination			**145,448**	**48,301**	**28,910**	**136,619**	**122,628**	**82,725**	**64,824**	**629,454**	**587,202**	**-**	**42,252**	**1,418,362**
3a	**3a.1.1**	ISFSI Security during Decommissioning	16,952						1,695	18,647	-	18,647	-	288,600
3a	**3a.1.2**	ISFSI Security during DFS	77,851						7,785	85,636	-	85,636	-	1,530,880
3a	**3a.1.3**	ISFSI Operations & Maintenance						17,546	2,632	20,177	-	20,177	-	
3a	**3a.1.4**	ISFSI NRC Fees						9,658	1,449	11,107	-	11,107	-	
3a	**3a.1.5**	ISFSI DAW Waste				1,650			247	1,897	-	1,897	-	
	3a.1	**ISFSI Security & Operations**	**94,802**	**-**	**-**	**1,650**	**-**	**27,204**	**13,808**	**137,464**	**-**	**137,464**	**-**	**1,819,480**
3a	**3a.2.1**	ISFSI Property Taxes (not required)						-	-	-	-	-	-	
3a	**3a.2.2**	ISFSI Insurance						17,904	2,686	20,589	-	20,589	-	
	3a.2	**ISFSI Insurance & Taxes**	**-**	**-**	**-**	**-**	**-**	**17,904**	**2,686**	**20,589**	**-**	**20,589**	**-**	**0**
3a	**3a.3.1**	ISFSI Utility Staff during Decommissioning	18,903						1,890	20,793	-	20,793	-	240,760
3a	**3a.3.2**	ISFSI Utility Staff during DFS	96,833						9,683	106,516	-	106,516	-	1,364,480
	3a.3	**ISFSI Utility Staff**	**115,735**	**-**	**-**	**-**	**-**	**-**	**11,574**	**127,309**	**-**	**127,309**	**-**	**1,605,240**
3a	**3a.4.1**	Corporate A&G Allocations						3,373	506	3,879	-	3,879	-	
	3a.4	**ISFSI Corporate Allocations**	**-**	**-**	**-**	**-**	**-**	**3,373**	**506**	**3,879**	**-**	**3,879**	**-**	**0**
3a - Dry Fuel Storage			**210,538**	**-**	**-**	**1,650**	**-**	**48,481**	**28,573**	**289,241**	**-**	**289,241**	**-**	**3,424,720**
3b	**3b.1.1**	Procure/Rent Transport/Transfer Equipment		2,627					394	3,021	-	3,021	-	
3b	**3b.1.2**	GTCC Transfer Operations						-	-	-	-	-	-	
3b	**3b.1.3**	Spent Fuel Transfer Operations					10,716		1,607	12,324	-	12,324	-	
	3b.1	**Spent Fuel & GTCC Transfer to DOE**	**-**	**2,627**	**-**	**-**	**10,716**	**-**	**2,001**	**15,344**	**-**	**15,344**	**-**	**0**
3b	**3b.2.1**	ISFSI Decommissioning & Demolition					3,269		490	3,759	3,759	-	-	

3b	**3b.2.2**	ISFSI Site Restoration						-	-	-	-	-	-	
	3b.2	**ISFSI Decommissioning & Demolition**	**-**	**-**	**-**	**-**	**3,269**	**-**	**490**	**3,759**	**3,759**	**-**	**-**	**0**
3b - Fuel Transfer & ISFSI Decommissioning			**-**	**2,627**	**-**	**-**	**13,985**	**-**	**2,492**	**19,104**	**3,759**	**15,344**	**-**	**0**
Period 3 Total - SNF/GTCC Dry Storage & Disposition			**210,538**	**2,627**	**-**	**1,650**	**13,985**	**48,481**	**31,065**	**308,345**	**3,759**	**304,585**	**-**	**3,424,720**
Grand Total			**355,986**	**51,487**	**28,910**	**138,268**	**136,613**	**132,414**	**96,066**	**939,744**	**590,961**	**306,530**	**42,252**	**4,843,082**

Note: Cost values are in thousands of 2020 dollars

Table 7-8 Estimated Staffing Numbers

Staff	2021	2022	2023	2024	2025	2026	2027-2058	2059
1b.1.1 OPPD Security Staff through Fuel Transfer								
1b.5.1 OPPD Utility Staff through Fuel Transfer								
2d.1.1 OPPD Utility Staff during Decommissioning	114	102	83	52	20	2		
2d.3.1 OPPD Plant Security during Decommissioning	18	18	16	6	2	0		
3a.1.1 OPPD ISFSI Security during Decommissioning	23	23	23	23	23	23		2
3a.1.2 OPPD ISFSI Security during Spent Fuel Storage							23	
3a.3.1 OPPD ISFSI Staff during Decommissioning	18	18	18	18	19	20		5
3a.3.2 OPPD ISFSl Staff during Spent Fuel Storage							21	
Average OPPD Personnel Count by Year	**173**	**161**	**140**	**100**	**64**	**45**	**44**	**7**
2a.3.4 DGC Baseline Development Staff								
2d.4.2 Decommissioning General Contractor Staff	22	22	22	22	20	4		
Average DGC Personnel Count by Year	**22**	**22**	**22**	**22**	**20**	**4**		
Total Average Staff & Securtiy Personnel	**195**	**183**	**162**	**122**	**84**	**49**	**44**	**7**

FORT CALHOUN STATION DECOMMISSIONING PROJECT
LICENSE TERMINATION PLAN

CHAPTER 8
SUPPLEMENT TO THE ENVIRONMENTAL REPORT

REVISION 0

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ABBREVIATIONS

AEC	Atomic Energy Commission
ALARA	as low as is reasonably achievable
AMSL	above mean sea level
BMP	best management practices
CE	Combustion Engineering
CNW	Chicago and Northwestern
DSAR	Defueled Safety Analysis Report
FCS	Fort Calhoun Station
FSS	final status survey
GEIS	general environmental impact statement
GTCC	greater than Class C
HASP	health and safety plan
IOF	ISFSI Operations Facility
ISFSI	Independent Spent Fuel Storage Installation
LTP	license termination plan
LLRW	low-level radioactive waste
MW	megawatts
NDEQ	Nebraska Department of Environment Quality
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
NSHPO	Nebraska State Historic Preservation Office
OCA	Owner Controlled Area
ODCM	Off-Site Dose Calculation Manual
OPPD	Omaha Public Power District
PCB	polychlorinated biphenyls
PSDAR	Post Shutdown Decommissioning Activity Report
PWR	pressurized water reactor
REMP	Radiological Environmental Monitoring Program
RGPP	Radiological Groundwater Protection Program

RP	radiation protection
SEIS	supplemental environmental impact statement
SWPPP	Storm Water Pollution Prevention Plan
EPA	U.S. Environmental Protection Agency

8 SUPPLEMENT TO THE ENVIRONMENTAL REPORT

8.1 Introduction

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(G) and the guidance of Regulatory Guide 1.179, "Standard Format and Contents for License Termination Plans for Nuclear Power Reactors" [1], this chapter provides a supplement to the environmental report describing any new information or significant environmental change associated with the site-specific decommissioning and site closure activities performed at the Fort Calhoun Station (FCS) site.

8.1.1 Purpose

This chapter supplements the Omaha Public Power District (OPPD) "Updated Environmental Report – Fort Calhoun Station" as supplemented [2], describing any new information or significant environmental changes associated with the site-specific decommissioning and license termination activities presented in this License Termination Plan (LTP). The supplement includes a detailed description of the remaining decommissioning and site closure activities, the interaction between those activities and the environment, and the likely environmental impact of those activities. The supplement discusses whether the activities and their impacts are bounded by the impacts predicted by the U.S. Atomic Energy Commission (AEC) "Final Environmental Statement related to operation of Fort Calhoun Station Unit 1" (AEC Environmental Statement) [3]; NUREG-1437, Supplement 12, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Regarding Fort Calhoun Station, Unit 1 [4]; and the FCS "Post Shutdown Decommissioning Activity Report" (PSDAR) [5]. This chapter discusses decommissioning activities, with a focus on those activities to be performed from time of submittal of this LTP until the licensee's completion of decommissioning activities.

8.1.2 Background

FCS is comprised of one 533.7 MWe pressurized water reactor (PWR) unit, with supporting facilities, which was operated by OPPD from 1973 to 2016.

The station was granted a construction permit by the AEC in 1968, and first commercial operation was achieved in 1973. OPPD made the decision to permanently shut down FCS due to a variety of factors, including economic analyses associated with production cost, economies of scale issues, and excess system supply. Permanent cessation of operations at FCS occurred on October 24, 2016. Certification of Permanent Defueled Status was achieved on November 13, 2016.

In accordance with the requirements of 10 CFR 50.82, OPPD submitted the initial revision of the PSDAR to the U.S. Nuclear Regulatory Commission (NRC) on March 30, 2017. The reactor at FCS remained in a SAFSTOR condition until December 16, 2019. At this point, the PSDAR was amended to address the acceleration of decommissioning activities, change the decommissioning schedule and cost milestones, and commit to notifying the NRC of the final disposition of the reactor closure vessel head. The amended PSDAR established the DECON method as the current decommissioning approach and described the accelerated decommissioning schedule with a lower revised cost estimate to reflect current knowledge and waste disposal options.

The environmental impacts of decommissioning operations at FCS were assessed as part of the PSDAR submittal. The assessments included the evaluation of impacts against those noted in the AEC Environment Statement and NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" (GEIS) [6]. The reference facility used from NUREG-0586 is an approximately 931 MWe PWR, operated by Maine Yankee Atomic Power Company and designed by Combustion Engineering (CE). The FCS PWR, designed by CE, had a much lower power output. With the power output of FCS lower than that of the reference plant, the use of the reference site will provide a bounded comparison of the generic environmental assessment.

The amended PSDAR concluded that the decommissioning of the FCS would be accomplished with no significant adverse environmental impacts and that the environmental impacts associated with the site-specific decommissioning activities for FCS would be bounded by previously-issued environmental impact statements.

8.2 Site Location and Description

The FCS site is located on the southwest bank of the Missouri River at river mile 646.0, approximately 19.4 miles north of Omaha, Nebraska. LTP Chapter 1, Figure 1-1, shows the geographical location of the site relative to nearby towns, cities, and the river. OPPD has a perpetual easement on approximately 117 acres of land primarily on the northeast bank of the river directly opposite the plant buildings.

FCS is situated within parts of Sections 16, 17, 20, and 21, Township 18 North, Range 12 East of Washington County, Nebraska in the Modale, IA quadrangle. The site is part of the Missouri River bottomland, which is a nearly level plain about 15 miles wide at Blair, Nebraska, 8 miles wide at the site, and narrowing to 3 miles wide in the vicinity of Omaha-Council Bluffs. On the southwestern part of the site, the ground rises sharply about 60 feet to a higher-level area, which is bounded on the south by U.S. Highway 75, formerly U.S. Highway 73.

The elevation of this plain averages about 1,000 feet above mean sea level (AMSL) at the site. The surface of the land, starting from the Missouri River at about 997 feet AMSL, falls to an old channel of the river before rising again to approximately 1,004 feet AMSL. Beyond this point, the land then gradually falls off to about 1,000 feet AMSL, rises again to approximately 1,020 feet AMSL, and then rises approximately 60 feet to a higher plateau at elevation 1,080 feet AMSL. LTP Chapter 1, Figure 1-4, provides a depiction of the FCS site topography.

The site is comprised of approximately 660 acres, which is owned or controlled by OPPD. A portion of the property has been previously removed from the FCS License but is maintained under OPPD ownership. LTP Chapter 1, Figure 1-2, provides a depiction of the FCS Owner Controlled Area (OCA).

A rail spur from the Chicago and Northwestern (CNW) Railway was constructed to serve the construction of the FCS. The original CNW tracks and rail spur have since been removed. In 1994, a permanent easement was granted to allow the construction of a new rail spur in the approximate location of the old CNW railway to allow trains to serve the Cargill industrial facility located north of FCS. In late 2020, OPPD added three lines (Lines A, B, and C, approximately 1,360, 1,595, and 2,230 feet in length, respectively) to the rail spur to allow for the direct loading of waste into rail cars. The added rail lines will remain in place after license termination with the exception of the portion of the line that resides within the waste processing structure.

8.2.1 Site Description after Unrestricted Release

This section provides a summary of the final condition of the site at the conclusion of decommissioning and site closure activities. The “End State” is defined as the configuration of the remaining below-ground basements, above-ground structures, piping, and open land areas at the time of license termination. Chapters 3 and 6 of the LTP provide more detailed discussions of the FCS end state.

The following above-ground buildings will remain at the time of license termination: Training Center, FLEX Building, OCA Entrance Building, 3451 Old Building, 3451 New Building, and the 1251 Control and Switchgear Building. With the exception of the removal of the 345kv and 161kv lines to the station, the Switchyard will remain after decommissioning. All other structures will be removed to a minimum of three feet below grade (approximately 1,001 feet AMSL). Because there are no dose model implications, if the number and identification of above-ground buildings to remain and be subject to final status survey (FSS) changes, the NRC will be notified.

For the Containment Building basement, all concrete will be removed from the inside of the steel liner, leaving only the remaining exposed liner below 1,001 feet AMSL and the structural concrete outside of the liner. In the Auxiliary Building basement, all internal walls and floors below 1,001 feet AMSL will be removed, leaving only the reinforced concrete floors and outer walls of the building structures. In the Turbine Building, all interior walls and floors below 1,001 feet AMSL will be removed with the exception of the turbine pedestals. For the Spent Fuel Pool, the only portion of the structure that will remain is the lower pool below the 1,001 feet AMSL and the concrete structure of the Fuel Transfer Canal after the steel liner has been removed. There are additional below-ground structures that will remain, including the lower concrete portions of the Circulating Water inlet piping and Circulating Water discharge tunnels.

An evaluation was performed regarding the disposition of the Intake Structure. The option to leave the structure’s basement in place below 1,001 feet AMSL and backfill was determined to be the least disruptive to the environment. The impact of leaving the Intake Structure and intake and discharge piping in place is discussed in a correspondence developed by Olsson Engineering with the Army Corps of Engineers [7].

For all excavations created by the removal of buried piping, components, or slab-on-grade structures, a radiological assessment will be performed prior to backfill. For all below-ground basements that will remain, an FSS will be performed, and contingent upon the completion of confirmatory surveys and regulatory approval, the basements will be backfilled. If a major below-ground building has been removed in its entirety, an FSS will be performed on the resultant excavation (using the most-restrictive classification of the building prior to demolition) prior to backfill of the excavation. All void spaces will be backfilled using clean fill material to grade (1,004 feet AMSL). The end state will also include a range of buried piping, embedded piping, and penetrations. All buried piping that is abandoned in place will be capped and/or filled with grout. The restored areas on the site will be backfilled, graded, and returned to natural contours. The demolition debris will be segregated for recycling or disposal.

The ISFSI, ISFSI Operations Facility (IOF), and support structures will remain in the reduced 10 CFR Part 50 license and are not within the scope of the proposed partial site release.

Several structures will remain at the end state as requested by OPPD. These structures are as follows:

- OPPD Electrical Switchyard and supporting structures (note: the Switchyard will remain in active use after decommissioning in support of the existing OPPD off-site electrical transmission and distribution system)
- FLEX Building
- Training Center
- Training Center - Sanitary Sewage System
- OCA Entrance Building
- Paved roadways and rail lines, including the lines and rail spur constructed in 2011 and 2020, allowing for rail service at the site via connection to the nearby Union Pacific railway

After all demolition and remediation activities are compete, OPPD will use the FSS process described in Chapter 5 of this LTP to demonstrate that the FCS and surrounding open land areas, with the exception of the ISFSI, comply with radiological criteria for unrestricted use specified in 10 CFR 20.1402. As part of the decommissioning process, all reactor fuel was loaded into casks and transferred to the ISFSI. The greater than Class C (GTCC) waste will be segregated and transported for storage in the ISFSI. It is expected that the fuel will remain on-site in dry storage within the ISFSI until it is transferred as determined by the U.S. Department of Energy. The ISFSI, which occupies approximately 6.5 acres, was constructed in the northwest corner of the previous FCS protected area, northeast of the OPPD switchyard.

Following the conclusion of radiological remediation activities and prior to initiating final survey, isolation and control measures will be implemented. The control measures will be implemented to ensure the final radiological condition is not compromised by the potential for recontamination as a result of access by personnel or equipment. Open land areas, access roads, and boundaries will be posted with signs restricting access. Isolation and control measures will be implemented through approved plant procedures and will remain in force throughout FSS activities, until there is no risk of recontamination from decommissioning or the survey area has been released from the license.

Several services, such as the Blair City water service and OPPD electrical service, will remain in operation to support the ISFSI monitoring and security operations. The station monitoring wells will remain in service for continued periodic sampling. There are no potable wells in service on site. Water service is provided through the City of Blair municipal water supply which draws water from the Missouri River via a water intake about 5 miles north of the site.

The Missouri River, which flows generally north to south, forms the northeast to southeast site boundary. This part of the river is referred to by the Army Corps of Engineers as the Blair Bend. The river limits are under control of the Army Corps of Engineers who have established a structure azimuth line, which acts as another site boundary.

8.3 Remaining Dismantlement and Decommissioning Activities

Key dismantlement and decommissioning activities that have been completed include activities associated with the removal of system piping and components, the segmentation and packaging of the internals from the FCS reactor, and the demolition and disposal of several ancillary structures.

Chapter 3 of this LTP provides details on the dismantlement, demolition, and remediation activities currently performed and remaining activities to be executed to achieve the end state condition.

8.4 Impacts to the PSDAR

The PSDAR, amended in December 2019, describes the planned decommissioning operations at the site and concludes that the potential environmental impacts associated with decommissioning the site have already been postulated in, and will be bounded by, the previously issued environmental impact statements, specifically:

- Final Environmental Statement, as amended and supplemented
- NUREG-0586
- NUREG-1496, “Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities” [8]

Chapter 3 of this LTP identified the dismantlement and decontamination activities that are scheduled to be completed prior to unrestricted release of the site (excluding the ISFSI footprint). These identified activities are not significantly different from what was proposed in the PSDAR. Although additional details regarding major decommissioning activities will be defined during ongoing decommissioning planning efforts, no significant impacts beyond those identified in the PSDAR have been identified. Subsequent sections in this chapter provide additional evaluation and information regarding the environmental effects of decommissioning.

8.5 Fort Calhoun Station Environmental Description

8.5.1 Geography and Demography

8.5.1.1 Site Location and Description

The site location and description were previously discussed in Section 8.2. FCS is situated within parts of Sections 16, 17, 20, and 21, Township 18 North, Range 12 East of Washington County, Nebraska in the Modale. IA quadrangle. The site is part of the Missouri River bottomland, which is a nearly level plain about 15 miles wide at Blair, Nebraska, 8 miles wide at the site, and narrowing to 3 miles wide in the vicinity of Omaha-Council Bluffs. The elevation of this flood plain averages about 1,000 feet AMSL at the site. On the southwestern part of the site, the ground rises sharply about 60 feet to a higher, level area, which is bounded on the south by U.S. Highway 75, formerly U.S. Highway 73. The topography at the site ranges from approximately 997 feet to 1,080 feet AMSL and represents a river bottom deposit along the Missouri River subsequent to the most recent period of glaciations.

8.5.1.2 Population

According to the U.S. Census Bureau American Community Survey report from 2010 [9], the population of Washington County is 20,234. The nearest municipality is the city of Blair, Nebraska, about three miles northwest, with a population of 7,990. Fort Calhoun, Nebraska, is about five miles southeast with a population of 908. Kennard Village, Nebraska, about seven miles from the plant site has a population of 167. Omaha, Nebraska, and Council Bluffs, Iowa, metropolitan area is 10 to 25 miles southeast of the site with a majority of the population beyond the 15-mile radius from the site with a population of 789,342. Missouri Valley, Iowa, about 11 miles east, has a population of 2,838.

8.5.1.3 Site Access, Land, and Water Use

The remaining site encompasses approximately 540 acres and is located on the alluvial plain of the Missouri River in a predominantly agricultural region roughly ten miles north of the Omaha metropolitan area. The site is relatively isolated and is bordered to the northwest by an industrial facility and farm fields, to the southeast by farm fields, to the south by a state highway with scattered residence, and to the north by the Missouri River, farm fields, and a wild life refuge area. The Missouri River is also used for commercial barge and recreational boat traffic.

According to the “Historical Site Assessment for Fort Calhoun Station” [10], there are no residences within 0.5 miles of FCS, as measured from the center of the Auxiliary Building stack. The seven nearest residences are from 3,000 to 4,000 feet distant. There are no schools, hospitals, prisons, motels, or hotels in the immediate vicinity of the site.

8.5.1.4 Climate

Nebraska is located midway between two distinctive climate zones, the humid east and the dry west. Cyclic weather conditions representative of either zone, or combinations of both, occur. Changes in weather result from the invasion of large masses of air with dissimilar properties. These air masses tend to get their characteristics from either the warm and humid south-southeast, the warm and dry southwest, the cool and dry north-northwest, or the cold continental polar air of the north. The region is also affected by many storms or cyclones (areas of low pressure) which travel across the country, generally from west to east. Periodic and rapid changes in the weather are normal, especially in the winter. The mean annual temperature for the region is 51.1 degrees Fahrenheit. The January monthly mean is 20.2 degrees Fahrenheit, while that for July is 77.7 degrees Fahrenheit.

Annual average precipitation for the region is about 28.5 inches, but annual amounts vary widely from year to year. About 75 percent of the precipitation occurs during showers and thunderstorms from April through September. Snowfall amounts to about 30 inches of snow as the annual average, but total annual amounts vary widely from year to year.

The surface wind direction and speed are quite varied during all seasons of the year. The prevailing wind direction from May through December is from south-southeast; north-northwesterly winds prevail during the remainder of the year. The mean annual wind speed is 10.6 miles per hour.

Tornado events in the counties surrounding FCS since 1950 include: Washington County, NE, recorded 19 events, Douglas County, NE, recorded 20 events, Pottawattamie County, IA, recorded 42 events, and Harrison County, IA, recorded 23 events. On June 7, 1953, a category F2 tornado, 5.7 miles away from the FCS, injured one person and caused an estimated $25,000 in damages. On May 6, 1975, a category F4 tornado, 16.9 miles away from FCS, killed three people, injured 133 people, and caused approximately $250,000,000 in damages. There has been no reported tornado causing damage within five miles of the Station (Source: National Centers for Environmental Information, NOAA website [11]).

8.5.2 Geology and Seismology

The soils below FCS include thick beds of limestone, dolomite, shale, sandstone, and thin layers of coal beds. The deeper formations were deposited in marine depositional environments with the shallow soils from the lateral migration of the paleo river channel. The major tectonic features of the mid-continent region began to develop late in the Paleozoic Era, and probably most of the important structural features of the Nebraska-Iowa Missouri River Valley area had already developed or were developing by the end of Permian period. However, there is no record of movement of the fault in historic times, or any indication of activity in recent geologic time.

At the beginning of the Pleistocene period, the Missouri River Valley and its main tributaries were established in their approximate present positions. Subsequently, under successive glacial movements, the valleys were filled and re-opened several times. During this period, the Peorian loess was deposited on the terraces and adjacent uplands. It is probable that only the upper part of the alluvium in the Missouri River Valley is actually of recent age and that deeper deposits are mostly of Pleistocene age.

According to the Defueled Safety Analysis Report (DSAR), unconsolidated sediments at the plant site generally range from 65 to 75 feet in thickness. The soils are typically interstratified and cross-bedded. These soils may be grouped generally into two units:

- an upper fine-grained sandy clay with silt approximately 20 to 50 feet thick
- an underlying carbonate bedrock surface at a depth of approximately 65 to 75 feet below ground surface

The upper units were representing former river deposits and are not likely continuous, but rather have preferential channels formed by paleo-oxbow deposits.

Pennsylvanian-aged limestone and shale (bedrock) of the Kansas City Formation are encountered below the overburden soils. The bedrock below the site consists of various types of limestone formations.

The site lies in a region of infrequent seismic activity. Since the middle of the 19th century, from the first historical record of earthquake occurrence in the area, only 13 shocks with epicentral Modified Mercalli Intensities of V or greater have occurred within 200 miles of the plant site.

Only one earthquake of Modified Mercalli Intensity VI has been reported within 200 miles of the site. It occurred in March 1935, near Tecumseh, Nebraska.

8.5.3 Hydrology and Hydrogeology

Water levels recorded at the site show that the groundwater gradients are nearly flat, with only a gentle slope toward the river, about 10 feet below the ground surface. Water levels at the site varied from elevations 993.7 to 992.4 feet, while the river levels recorded during this same period ranged from elevations 993.2 to 992.4 feet. Groundwater levels vary with changes in the river level. The rate of groundwater flow in the alluvial soils varies with the permeability; however, the groundwater flow rate, or velocity, is very slow due to the low gradients. The coefficient of permeability varied from about 0.5 to 3 feet per day in the upper sandy silt and silty sand. In the lower fine-to-coarse sands and gravels, coefficients of permeability as high as 20 feet per day were measured.

According to the DSAR, groundwater flow directions have been reported to be both toward the Missouri River (northeasterly) and away from the Missouri River (south-southwesterly). Flow directions towards the river appear to represent times when Missouri River levels are relatively low (e.g., during the spring, summer, and early fall, when most precipitation occurs and the river flow is relatively high). Flow directions away from the river appear to represent times when Missouri River levels are relatively high, causing bank storage effects (e.g., during late fall and winter when the river recedes).

The FCS site is bounded on the northeast and southeast by a portion of the Blair Bend of the Missouri River. The Corps of Engineers maintains river structures to prevent further meandering of the channel within the alluvial flood plain; the structures take the form of pile dikes and revetments.

Fish Creek is an intermittent drainage stream that runs along the northwest boundary of the ISFSI Protected Area, which is on the northwest boundary of the deconstruction area. This stream discharges into a larger wetland, before flowing in the Missouri River.

8.6 Environmental Effects of Decommissioning

8.6.1 Summary

The evaluation and methodology of the environmental effects (or impacts) of the decommissioning of FCS follows the approach outlined in the GEIS. This approach includes identification of environmental issues as either generic or site-specific. If the issue is considered generic, it is assigned a significance level of either “Small,” “Moderate,” or “Large.” If identified as generic, the environmental impact is considered to be bounded by the evaluation in the GEIS, which concludes that the impact significance is “Small.” In this event, site-specific evaluation by licensees is generally not required.

For those environmental issues or decommissioning activities that require site-specific evaluation, a standard approach is followed. It is summarized as follows:

1) The issue or activity is summarized including a summary of the impacts as reported in the original Environmental Statement (ES) and PSDAR. Note that many decommissioning activities are not identified in these documents.

2) Applicable regulations, permits, limits or other regulatory requirements are identified.

3) Potential impacts from decommissioning activities relating to the environmental issue are described.

4) An evaluation is performed. This includes analysis and professional judgment to estimate or determine whether the activity is likely to make a noticeable impact on the environment considering the available information. If an impact is likely, existing and additional mitigation measures that can be taken are evaluated. If an impact cannot be avoided, a determination is made as to whether the impact is likely to seriously damage the resource or attribute.

5) A conclusion is reached. A conclusion is derived from the evaluation steps summarized above. The conclusion identifies the level of significance of the impacts. Site-specific issues are not bounded by the GEIS evaluation.

Table 8-1 was used as the basis for the site-specific environmental impact assessment for FCS. It is excerpted from Table 6.1 of NUREG-0586, Supplement 1. The first step in this process is to screen the issues to identify those that are site-specific. Decommissioning activities specific to FCS are then reviewed, and the activities that may require site-specific evaluation are identified. The screening identified the following:

- off-site land use activities: changes in demographics and zoning that have occurred in the past 40 years
- aquatic ecology affected by activities beyond the operational area: changes in designation of sensitive areas (local wetlands and Missouri River bank)
- terrestrial ecology affected by activities beyond the operational area: changes in designation of sensitive areas (local wetlands and Missouri River bank)
- threatened and endangered species: changes in local flora and fauna and designation of threatened and endangered species that have occurred in the past 40 years
- environmental justice: changes in demographics and socioeconomic status in the past 40 years
- cultural and historic resource impacts beyond the operational areas: changes in local historic landmark designations and other cultural resources

The following decommissioning activities, which required evaluation of impacts across several environmental attributes or issues, were identified:

- rail line upgrade and extension (on-site)
- Circulating Water inlet and outlet piping disposition impact on aquatic ecology (within and beyond the operational area)

8.6.2 Radiological Effects of Decommissioning

8.6.2.1 Occupational Radiation Exposure

During decommissioning, OPPD has implemented, and will continue to implement, a Radiation Protection (RP) Program in accordance with the license specifications and the requirements of 10 CFR Part 20. The objectives of the RP Program are to control radiation hazards, avoid accidental radiation exposures, maintain occupational worker exposures to less than the administrative limit of less than 2,000 mrem/year total effective dose equivalent, and to maintain doses to workers and the public as low as is reasonably achievable (ALARA).

On October 24, 2016, OPPD placed FCS in a SAFSTOR condition (a period of safe storage of the stabilized and defueled facility). The reactor at FCS remained in a SAFSTOR condition until December of 2019, when FCS changed the decommissioning method to DECON. This period allowed for the decay of most short-lived radionuclides, which subsequently, reduced radiation levels at the facility. This, combined with the effective implementation of the RP Program and ALARA measures, minimizes the projected and actual occupational radiation dose exposure during the decommissioning of FCS. It is anticipated that the most significant contributors to occupational dose from remaining dismantlement activities is the segmenting, packaging, and shipping of the reactor vessel internals and the reactor vessel.

The GEIS estimates that 560 to 1000 person-rem would be needed to decommission a PWR. The current occupational dose expended and dose expected to complete decommissioning for FCS is estimated to be approximately 285 person-rem. This is below the GEIS estimate.

As the occupational dose for the decommissioning will meet the regulatory standards of 10 CFR 20, it is therefore bounded by the criteria in the GEIS, and the impact is considered "Small."

8.6.2.2 Off-Site Radiation Exposure and Monitoring

OPPD implements a regulatory compliant Radiological Environmental Monitoring Program (REMP) at FCS, which provides annual reports with an accurate assessment of the radiological environment in and around the environs of the site. The REMP provides assurance that the radioactive gaseous and liquid effluent releases during plant operations do not exceed the concentration limits of 10 CFR 20, the dose limits of 10 CFR 50, Appendix I, or the fuel cycle dose limits of 40 CFR 190. OPPD and Energy*Solutions* will continue to adhere to these limits throughout the course of the decommissioning.

At FCS, the Circulating Water Discharge Tunnel is the main authorized effluent release pathway for the discharge of treated and filtered radioactive liquid waste to the Missouri River. Liquid effluents are monitored and sampled prior to release from on-site storage tanks.

The gaseous pathway analysis is subject to the meteorological conditions during the time of the release. Due to plant shutdown and cessation of noble gas and other radionuclide generation, gaseous effluents do not present a significant release or exposure pathway. Routine air sampling is performed to determine the dose due to radioactive gaseous releases.

The direct radiation exposure is measured continuously with the use of passive monitoring devices. The dose is integrated over three months to accumulate a statistically significant exposure.

The design basis for the ISFSI precludes airborne radioactive releases during spent fuel storage and provides adequate shielding to minimize exposure. Radiation monitoring for the ISFSI is performed in accordance with the RP Program implemented at FCS. In accordance with the worst-case scenario in the design basis, the projected doses at the site boundary are substantially below the limits established in 10 CFR 72.106(b) where there is total loss of the confinement barrier. Exposure from the ISFSI to the nearest permanent resident will not exceed 25 mrem/year as specified in 10 CFR 72.104 and 40 CFR Part 190.

Consequently, the public dose from decommissioning is bounded by the criteria in the GEIS, and the impact is considered "Small."

8.6.2.3 Environmental Effects of Accidents and Decommissioning Events

Decommissioning accident analysis is integral to the licensing design basis for FCS. While decommissioning radioactively contaminated structures, systems, and components at FCS, it is necessary to assure the safety of the public in the surrounding area and workers. Worker safety is addressed in the RP and Safety programs for the FCS decommissioning project, which rely on ALARA principles and the FCS-SAF-103, "FCS Deconstruction Health and Safety Plan" (HASP) [12]. The safety of the public is principally related to potential hazards associated with an airborne release of radioactive materials during decommissioning operations.

During decommissioning, FCS will perform decontamination and dismantlement of structures, systems, and components in addition to maintenance, waste management, and surveillance. The accidents discussed in NUREG-0586, Supplement 1 associated with immediate dismantling would also be applicable during the decommissioning of FCS. However, the potential consequences associated with those accidents would be less at FCS because of the reduction of the total radionuclide inventory at FCS due to:

- decontamination efforts made before decommissioning,
- prior radioactive waste shipments, and
- radioactive decay.

Consequently, the potential decommissioning accidents at FCS are bounded by the accident evaluation presented in NUREG-0586, Supplement 1.

Operational accidents during decommissioning could result from equipment failure, human error, and service conditions. With the spent nuclear fuel located in the ISFSI, operational accidents during decommissioning can be categorized as follows:

- radioactive waste transportation accidents
- explosions and/or fires associated with explosive and/or combustible materials
- loss of contamination control
- natural phenomena

- human caused events external to FCS

These potential operational accidents during decommissioning are addressed in NUREG-0586, Supplement 1 for immediate dismantlement and, consequently, are bounding for the decommissioning of FCS.

8.6.2.4 Storage and Disposal of Low-Level Radioactive Waste

The decommissioning of FCS has required, and will continue to require, the disposal of large volumes of low-level radioactive waste (LLRW), including contaminated equipment, tools, clothing, and bulk debris materials such as concrete, metal, and asphalt. Materials that cannot be free released are, and will continue to be, dispositioned as LLRW. Through the proper implementation of the Waste Management Program, Process Control Program, and associated procedures, OPPD ensures the appropriate segregation, classification, processing, packaging, shipment, and control of solid, liquid, and gaseous radioactive wastes.

The majority of the Class A LLRW from FCS will be shipped to the Energy*Solutions* disposal site in Clive, Utah. No significant impacts are expected from the disposal of LLRW. The total volume of LLRW for disposal was estimated in the Decommissioning Plan to be approximately 6,000,000 cubic feet. Actual waste volumes and classifications may vary. The vast majority of waste will be loaded into 8-120A or 3-60B casks, gondola rail cars, or articulating bulk cars and shipped to Clive, Utah. Oversized or overweight components, such as the reactor vessel head, are shipped using multiple axle tractor/trailer rigs or special rail cars. Rail and truck shipments are made in accordance with U.S. Department of Transportation regulations.

OPPD completed the construction of the ISFSI in 2005 with initial loading of 10 dry storage casks (DSC) commencing in August 2006 and completed in August 2009. The final spent nuclear fuel cask loading campaign started in October 2019 and concluded in May 2020. OPPD completed the transfer of all its spent nuclear fuel contained in 40 DSCs to the ISFSI, in May 2020. The multi-purpose fuel canisters within the casks are seal-welded and leak tight; therefore, no leakage is expected during normal operation, off-normal conditions, or design basis events. The storage of the fuel at the ISFSI does not generate any gaseous, liquid, or solid radioactive waste. The spent nuclear fuel will remain in storage at the ISFSI under the Part 50 license until the fuel is transferred to a permanent repository. GTCC waste will be stored in two seal-welded leak tight canisters within storage casks co-located at the ISFSI with the spent fuel.

8.6.2.5 Radiological Criteria for License Termination

Following the completion of decontamination, dismantlement, and remediation activities, radiological surveys will be performed to demonstrate that the dose from any residual radioactivity remaining in as-left structure basements and soils at FCS (excluding the ISFSI) meets the unrestricted release criteria specified in 10 CFR 20.1402. Once the balance of the site is remediated and the as-left radiological conditions are demonstrated to be below the unrestricted release criteria, the 10 CFR Part 50 license will be reduced to the area around the ISFSI. LTP Chapter 5 and Chapter 6 provide the methodology for demonstrating compliance with the unrestricted release criteria.

8.6.3 Non-radiological Effects of Decommissioning

8.6.3.1 On-Site Land Use

The environmental impact associated with on-site land uses have been determined by the NRC within Section 4.3.1 of NUREG-0586, Supplement 1 to be generically considered as a "Small" impact.

The decommissioning project is located and executed within the remaining boundary of the FCS property previously used for power generation. Some on-site roads have been refurbished, and a reinforced heavy haul path was constructed to support the transfer of radioactive waste to the rail tent. No barge slips are being utilized. The rail was originally installed during the construction of the station and was modified to support decommissioning activities. On-site land activities such as vehicle parking and equipment/container laydown, storage, staging, and waste loading occur, and continue to occur, in a manner similar to when the facility was operational. Several structures such as the Switchyard, ISFSI, IOF, FLEX Building, Training Center, as well as all roadways and a majority of rail lines, will remain at license termination.

Uncontaminated concrete and other demolition debris, where radiological surveys demonstrate that the concrete is free of plant-derived radionuclides above background will be disposed of as clean waste. Demolition debris found to be contaminated or potentially contaminated based on process knowledge will be disposed of as LLRW. Consequently, the burial of demolition debris contaminated with residual radioactivity will not have the potential to affect land use and ground or surface water quality.

As during the operation of the facility, decommissioning activities have not been conducted in wetlands. The wetlands around the plant have been protected in accordance with environmental regulations and permits.

There is no information pertaining to any significant environmental changes associated with the site-specific decommissioning activities. Site closure will comply with applicable U.S. Environmental Protection Agency (EPA) regulatory requirements.

In accordance with the guidance presented in the GEIS, the potential impacts to land use onsite are considered "Small."

8.6.3.2 Off-Site Land Use (in the Vicinity)

Only areas within the existing site boundary will be used to support decommissioning and license termination activities (such as temporary storage and staging areas). Appropriate isolation and control measures will be instituted to prevent the spread of contamination. These measures will also be monitored to ensure their effectiveness. Thus, no environmental impacts associated with the use of off-site lands are anticipated from the decommissioning activities at FCS.

Of the 660-acre site, approximately 119 acres is designated as part of the area to be decommissioned. The remaining land belonging to OPPD contains structures, is part of the open area, or has already been released.

It is assumed that construction activities will disturb one acre or greater of soil and requires a National Pollution Discharge Elimination System (NPDES) Construction Storm Water General

Permit from the Nebraska Department of Environment Quality (NDEQ) and possible NRC notification, prior to proceeding with the activity. OPPD has a Storm Water Pollution Prevention Plan (SWPPP) for decommissioning and construction activities at the FCS site. The SWPPP contains best management practices (BMPs) to avoid and/or minimize sediment and erosion discharges to watercourses and wetlands. All BMPs will be in place prior to initiating decommissioning/construction activities.

Once decommissioning is complete, decommissioned areas at FCS will be either be left as a gravel/paved area with appropriate permanent storm water controls left in place or restored back to grassland; therefore, there will be a net gain in undisturbed area at FCS and no significant adverse impacts to land use as a result of decommissioning.

Decommissioning activities are not being performed in areas defined as “environmentally sensitive” within the site boundary, nor in land which adjoins similar off-site land areas. Consequently, the off-site land areas are not affected by the decommissioning activities and the potential impacts to land use off-site are considered “Small.”

8.6.3.3 Water Use

In accordance with Section 4.3.2 of NUREG-0586, Supplement 1, the environmental impact associated with water use has been determined to be generally applicable with a “Small” impact.

FCS is located on the bank of the Missouri River, and the Intake Structure affects approximately 80 feet of riverbank. The predominant water usage during the operation of FCS was the use of water from the Missouri River as secondary cooling water for the reactor systems. With the plant shutdown and fuel removed from the reactor and located in the ISFSI, the remaining use of the river is for dilution of released liquid waste. The use of water from the Missouri River during decommissioning activities is significantly less than the usage during operations.

Water from the Missouri river is used extensively for municipal and domestic water supplies. There are multiple potable water intakes located on the Missouri river near FCS. The nearest substantial upstream intake is located about 1 mile northwest of the FCS site and downstream intake approximately 20 miles from the site. The City of Blair provides potable water services to support FCS. Potable water use during decommissioning operations is not expected to be greater than the potable water use experience during operations. The IOF sewage system is self-contained and does not release to the environment at the site. The Training Center will utilize a new on-site sewage waste system. Released wastewater will continue to be processed in accordance with the site’s NPDES permits.

Consequently, in accordance with the GEIS, the potential impacts to water use are considered “Small.”

8.6.3.4 Water Quality

The environmental impact evaluation associated with surface and groundwater quality in section 4.3.3 of NUREG-0586, Supplement 1 has been determined to be generally applicable to FCS with a “Small” impact.

This section evaluates potential project effects on those portions of the natural environment related to surface water and groundwater. Surface water generally refers to streams, rivers, ponds, reservoirs, and lakes. At FCS, the nearby bodies of water are the Missouri River, surface streams near the site, including Fish Creek (immediately northwest of the IOF), Long Creek (0.7 miles southeast), DeSoto Lake (1.8 miles east), and the surrounding wetlands.

At FCS, all non-radiological water discharges to Fish Creek and the Missouri River are controlled under NPDES permits, including Storm water Discharges from Industrial Activity, Industrial, and General NPDES Permit Authorizing Dewatering Discharges Activity. These permits were issued by the NDEQ. In addition, impacts are greatly reduced through implementation of appropriate BMP for soil erosion and sedimentation control. There is no impact to the nearby lake.

Radiological impacts are minimized through adherence to Off-Site Dose Calculation Manual (ODCM) [13] limits and assessed through the REMP and the Radiological Groundwater Protection Program (RGPP). Potential groundwater impacts are monitored by the routine sampling of 23 permanent on-site RGPP wells and other ground water locations at FCS.

As the water from the Missouri River is no longer used to cool operating reactor systems at FCS, the thermal impact to the Missouri River has been reduced.

No adverse impacts on surface water and groundwater are expected from the implementation of decommissioning activities. Consequently, the potential impacts to surface and groundwater quality are bounded by the GEIS and considered “Small.”

8.6.3.5 Air Quality

The environmental impact evaluation associated with air quality in section 4.3.4 of NUREG-0586, Supplement 1 has been determined to be generally applicable to FCS with a “Small” impact.

There are five non-radiological types of decommissioning activities listed in Section 4.3.4.3 of the GEIS that have the potential to affect air quality:

- worker transportation to and from the site
- dismantling of systems and removal of equipment
- movement of open storage of materials on-site
- demolition of buildings and structures
- shipment of material and debris to off-site locations

8.6.3.5.1 Worker Transportation

The work force at FCS has decreased significantly from the time the plant ceased operation in 2016 to a work force of approximately 300 people. The work force will temporarily increase during decommissioning by approximately 150 people. There will also be occasional increases during specific D&D activities until completion. The work force during decommissioning will be smaller than the work force needed during plant construction and routine refueling/maintenance operations. Therefore, there will be no significant adverse changes in air quality associated with changes in worker transportation since these changes in worker transportation will generally not be detectable or destabilizing.

8.6.3.5.2 Dismantling Systems and Removal of Equipment

There are potential sources of particulate matter that could impact air quality during the dismantlement of systems and the associated release of gases from systems during removal. Several mitigation efforts can be used to minimize fugitive dust such as wet suppression and chemical stabilization agents. In addition, airborne contamination can be minimized by isolating certain contaminated areas and implementing the use of air filtration systems when activities are located in areas that are not ventilated to the plant stack and are likely to generate airborne radioactivity or other hazardous pollutants. Other sources of air pollutants such as refrigerants will be disposed of according to applicable local, state, and federal regulations. FCS complies with all applicable federal and state air quality regulations, including the requirements of the NDEQ, and will implement the BMPs to minimize particulate matter generated during decommissioning and the released to the environment detectable off-site.

8.6.3.5.3 Movement and Open Storage of Materials On-Site

Movement of equipment and open storage of materials such as construction debris and soil stockpiling during decommissioning may result in fugitive dust. However, BMPs such as temporarily stabilizing stockpiled soil with seed and mulch and spraying of the debris containing particulates and dust will minimize fugitive dust during stockpiling. Similar BMPs will be established to mitigate effects while moving material within the site. Therefore, no significant adverse impacts to air quality from the particulate matter generated as a result of movement or storage of material onsite are anticipated.

8.6.3.5.4 Demolition of Buildings or Structures

It is anticipated that the demolition of buildings and structures will temporarily increase fugitive dust at FCS during decommissioning. Demolition activities will be conducted in an organized and methodical manner to avoid and minimize significant amounts of particulate and fugitive dust generation at one time during decommissioning. As demolition and loading of material occurs, the area exposed will be sprayed to minimize airborne dust and particulates. It is therefore anticipated that the demolition of buildings and structures will potentially create temporary impacts to air quality, but none that would be considered significant or adverse.

8.6.3.5.5 Shipments of Material to an Off-Site Location

It is anticipated that truck traffic will be required to remove construction materials, debris, and equipment from the FCS. The removal of materials will take place during active decommissioning, and will be include varying periods of heavier and lighter activity. An average of eight trucks per day is estimated over the course of active decommissioning. This increase in total truck traffic when added to the Nebraska Department of Transportation average of 585 heavy vehicles per day from 2018, is lower than the truck traffic recorded while FCS was in service of 640 heavy vehicles per day [14].

Fugitive dust and small particulates generated from truck traffic will be the primary contributor of potential air quality impacts during this phase of decommissioning. All appropriate BMPs will be implemented and maintained throughout decommissioning to ensure that there is a minimal amount of impacts to air quality.

In addition to the current air monitoring program at FCS, all air emissions will be monitored during decommissioning activities (fugitive dust, equipment exhaust, etc.) and will continue to be monitored in accordance with the ODCM, which sets limits on doses caused by effluents, based upon the ALARA objectives of 10 CFR 50.34a, 10 CFR 50.36a, and Section IV.B.1 of Appendix I to 10 CFR 50. Effluents are reported annually to the NRC.

FCS complies with all applicable federal and state air quality regulations, including the requirements of the NDEQ, and will implement BMPs to minimize fugitive dust during demolition and decommissioning activities.

No adverse impacts on air quality are expected from the implementation of decommissioning activities. Consequently, the potential impacts to air quality are bounded by the GEIS and considered "Small."

8.6.3.6 Aquatic Ecology

The environmental impact evaluation associated with aquatic ecology in section 4.3.5 of NUREG-0586, Supplement 1 has been determined to be generally applicable to FCS with a "Small" impact.

The aquatic habitat at FCS includes the area in front of the intake structure integral to the Missouri River bank. Habitats associated with this area were previously disturbed during the initial construction of the facility. However, the implementation of decommissioning activities is not expected to disturb existing aquatic habitats, the flora and fauna of nearby streams and wetlands. The maintenance of the riverbank will allow the river and local aquatic habitats to be maintained.

Various fresh water fish species, macro-invertebrate populations, and vegetation exist within these aquatic environments and were identified during a study contracted by OPPD. OPPD performed a project decision model to assist with the decision for removing or leaving the Intake Structure and Circulating Water Discharge Tunnel at FCS. The project decision model concluded that the removal of the above ground structures and filling of the Lower Intake Structure and Circulating Water Discharge Tunnel should be completed. This decision resulted in the least impact to the environment, including aquatic ecology considerations.

Plans for the demolition of structures at FCS do not include the removal of waste or equipment by barge. Consequently, there is no impact to the riverbank from this type of activity.

OPPD will continue to maintain its NPDES permits and decommissioning operations will be performed within applicable NPDES limits. Furthermore, protection of the onsite and adjacent wetlands is, and will continue to be, a priority when planning any onsite dismantlement or waste management operation. In addition, the BMPs are implemented to prevent impacts to the aquatic systems.

Exotic species can threaten native species and ecosystems due to aggressive growth, reproduction or survival rate, and diseases or parasites they may transmit to native species. The decommissioning of FCS will not introduce any exotic plants or animals into the environment.

The potential impacts to the aquatic ecology within the site boundary are bounded by the GEIS and considered to be "Small." The potential impacts to the aquatic ecology beyond the site boundary have also been evaluated and considered to be "Small."

8.6.3.7 Terrestrial Ecology

The environmental impact evaluation associated with terrestrial ecology in section 4.3.6 of NUREG-0586, Supplement 1 has been determined to be generally applicable to FCS with a "Small" impact.

Direct impacts can result from activities such as clearing native vegetation or filling a wetland. OPPD anticipates minimal disturbance of habitat beyond the operational areas of the plant. All dismantlement, demolition, and waste staging activities are envisioned to be conducted within the operational area of the site. In addition, the NDEQ controls significant impacts to the environment through regulation of construction activities.

Indirect impacts may result from effects such as erosional runoff, dust, or noise. Any construction activities that would disturb one acre or greater of soil requires a storm water permit from the NDEQ prior to proceeding with the activity. The storm water permit contains BMPs to control sediment and the effects of erosion associated with the construction activity. Fugitive dust emissions will be controlled through the judicial use of water spraying.

Section 4.3.6 of the GEIS concludes that if BMPs are used to control indirect disturbances and habitat disturbance is limited to operational areas, the potential impacts to terrestrial ecology are "Small." As discussed above, there are no unique disturbances to the terrestrial ecology anticipated during the decommissioning of FCS. Currently, FCS will be able to conduct all of these decommissioning activities on previously disturbed land. As required, the environmental impact will be reevaluated when activity is planned on previously undisturbed land. Therefore, based on the updated environmental report, OPPD concludes that the impacts of FCS decommissioning on terrestrial ecology are bounded by the GEIS.

8.6.3.8 Threatened or Endangered Species

The threatened and endangered species are identified in the 2002 Environmental Report. In October 2018, a review for plant and wildlife species protected by the Nebraska Nongame and Endangered Species Act was requested by the Nebraska Game and Parks Commission. The conclusion of the reports indicate the terrestrial species that may be at or near the FCS include the following:

- three bird species: the federal and state threatened bald eagle (Haliaeetus leucocephalus) and piping plover (Charadnus melodus), the federal and state endangered least tern (Sterna antillarum)
- one reptile species: the state threatened massasauga rattle snake (Sistrurus catenatus)
- three plant species: the federal and state threatened western prairie fringed orchid (Plantanthera praeclara), the state threatened small white lady's-slipper (Cypripedium candidum) and American ginseng (Panax quinquefolium)

- three fish species: the federal and state endangered pallid sturgeon (Scaphirhynhcus albus), the state threatened lake sturgeon (Acipenser fulvescens) and sturgeon chub (Macrhybopsis gelida)
- The only new species that may be at or near FCS not listed in 2002 was identified in 2018. It was a mammal species: the federal and state threatened northern long-eared bat (Myotis septentrionalis).

Least terns and piping plovers nest on riverine sandbars within the central United States, including those present along the Missouri River. The loss of sandbar nesting habitat due to river channelization and changes in flow from the construction and operation of main-stem dams have resulted in population declines for both the least tern and the piping plover along the Missouri River.

The amount of cooling water withdrawn from the Missouri River will significantly decrease, thus reducing the potential impacts of impingement, entrainment, and thermal discharges on aquatic species. One potential adverse impact from the decrease in cooling water withdrawn may be the elimination of the thermal refuge for aquatic species in the discharge area, which are preyed upon by the bald eagle in the winter months. Removal of the intake and discharge facilities as well as other shoreline structures will be conducted in accordance with BMPs outlined in permits issued by the NDEQ and the U.S. Army Corps of Engineers.

The historic range of the massasauga included eastern Nebraska and Washington County, but there are no recent records within 50 miles of FCS. Extant populations of the massasauga have been documented only in Colfax and Pawnee counties. This small rattlesnake prefers wet prairie habitat.

Two plant species are listed by the State of Nebraska, but not by the Federal Government. These include small white lady's-slipper (Nebraska-listed as threatened; occurs in wet meadows) and American ginseng (Nebraska-listed as threatened; occurs in high quality upland forest). The listed species are not known to occur on FCS.

The western prairie fringed orchid (federally listed as threatened) is found most often on unplowed, calcareous prairies and sedge meadows. It potentially occurs in Washington County, based on historic observations, but no populations are known to occur in the county, and the potential for occurrence on or near FCS is low given the lack of prairie habitat in these areas.

No designated critical habitat exists for any of the listed aquatic species on or in the vicinity of FCS. No aquatic species in the area is proposed for listing or is a candidate for listing. The FCS site is located within a reach of the Missouri River that has been channelized, with a relatively uniform width and swift current. This channel degradation results in a reduction of sediment and organic matter, flow modifications, and channel narrowing. As a result, it is believed that the cues for spawning are no longer present.

OPPD environmental personnel conduct monthly site environmental inspections to monitor for adverse environmental impacts and general environmental conditions including protected species. OPPD also has a plan in place for protection of the bats, as well as other mammals and birds that may be encountered in their service territory.

The environmental impacts during decommissioning are expected to be minimal on threatened and endangered terrestrial species. Additionally, FCS has administrative controls in place which require that significant project activities undergo an environmental review prior to the activity occurring, which ensures that impacts are minimized through implementation of BMPs and the OPPD Avian Protection Plan [15]. State permits are required, which include evaluation of the impacts to the environment and considerations of threatened and endangered species that may occur from the specific activity. Any necessary mitigation activities, should any be identified due to environmental consequences at the time, would receive the appropriate state permitting. This process also includes federal triggers to check for federal threatened and endangered species. No demolition at FCS would be allowed to proceed without the appropriate approval. Based on the above, the planned decommissioning of FCS will not result in a direct mortality or otherwise jeopardize the local population of any threatened or endangered species.

The potential impacts to “Threatened” or “Endangered” species are bounded by the GEIS and considered “Small.”

8.6.3.9 Occupational Issues/Safety

The environmental impact evaluation associated with occupational issues in Section 4.3.10 of NUREG-0586, Supplement 1 has been determined to be generally applicable to FCS with a “Small” impact. While decommissioning involves increased industrial activities and safety focus, similar programs addressing worker safety were implemented during the operation of the facility and also during repair and refueling outages. The occupational issues and safety impacts assessed are those related to human health and safety, including impacts from physical, chemical, ergonomic, and biological hazards. Radiological impacts were previously discussed in Section 8.6.2.1.

OPPD and Energy*Solutions* are committed to decommissioning FCS safely, and OPPD has established the HASP to effectively control hazards in the work environment and prevent occupational injuries and illnesses. The HASP and OPPD/Energy*Solutions* comply with federal and state regulations including Nebraska Department of Labor and the U.S. Occupational Health and Safety Administration requirements. The HASP applies to all OPPD and Energy*Solutions* employees as well as visitors and contract personnel working under direct OPPD or Energy*Solutions* supervision.

Numerous safety practices and communications are conducted at the site and include, but are not limited to:

- Safety is emphasized as the first topic of discussion at meetings.
- All workers are provided a Health and Safety booklet.
- Worker training and required certifications are reviewed prior to assignment to tasks requiring specific worker qualifications. Certain specialty subcontractors are mobilized, as necessary.
- Safety Data Sheets are obtained and reviewed for chemicals brought on-site.

- Health and Safety staff are involved in reviewing and approving decommissioning work packages and participating in pre-job walk downs, work condition assessments, and reviews.
- Daily and weekly safety messages are issued as well as Safety Bulletins to communicate awareness of significant safety issues and lessons learned.
- Safety stand-downs are held whenever serious safety events occur to communicate and reinforce safety events and lessons learned site-wide.

Therefore, occupational issues/safety is evaluated to be bounded by the GEIS, and the impact is considered “Small.”

8.6.3.10 Cost

A detailed discussion of the site decommissioning project costs is presented in Chapter 7 of this LTP.

8.6.3.11 Socioeconomic Impacts

OPPD’s original decision to permanently cease plant operations was not subject to NRC review or approval. On June 16, 2016, the OPPD Board of Directors voted to permanently cease operations of FCS for financial reasons.

As FCS transitions from shutdown and into the different phases of decommissioning, an overall decrease in plant staff will occur. The lost wages of these plant staff may result in decreases in revenues available to support the local economy.

Although FCS may have some effect on the region as a whole, the vast majority of FCS employees have resided in Washington, Douglas, and Sarpy Counties. FCS employees may be expected to impact the economy the most in terms of real estate and consumer goods within the counties where they live. Therefore, any effects of FCS’s closure are expected to be focused within these counties. Although effects outside of the counties are possible, if the effects within these counties are negligible, it is expected that effects in the surrounding areas are also negligible. Therefore, FCS closure should not have a significant adverse impact on the local economy in the years following closure.

Therefore, socioeconomic impacts are evaluated to be bounded by the GEIS, and the impact is considered “Small.”

8.6.3.12 Environmental Justice

While low-income and minority populations are present in the vicinity of the former FCS, the percentages of low-income and minorities within the FCS census tract are lower than those in other surrounding county census tracts. No disproportionate impact to the greater population, including special groups, is expected.

An existing rail spur was modified to transport large components and other waste from FCS. The refurbished rail spur is used to transport waste over an existing route. Decommissioning activities will cause increases in truck traffic to and from FCS to transport equipment and debris. The truck traffic will use existing highway and main street routes. Since a majority of the waste will be removed by rail, the increase in truck traffic will be temporary. There will be no environmental justice impact relative to rail and truck transportation as a result of decommissioning.

Based on the radiological environmental monitoring program data from FCS, the Supplemental Environmental Impact Statement (SEIS) determined that the radiation and radioactivity in the environmental media monitored around the plant have been well within applicable regulatory limits. As a result, the SEIS found that no disproportionately high and adverse human health impacts would be expected in special pathway receptor populations (i.e., minority and/or low-income populations) in the region as a result of subsistence consumption of water, local food, fish, and wildlife.

There is no reason to believe that low-income and minority populations will be adversely impacted by the decommissioning project. Per the GEIS and this evaluation, the potential site-specific impact is considered “Small.”

8.6.3.13 Cultural, Historic, and Archaeological Resources

Based on a review of the FCS property through the Nebraska State Historic Preservation Office (NSHPO) files and information provided by the applicant, the NRC concluded in Section 4.4.5 of the SEIS that the potential impacts from decommissioning of FCS on historic and archaeological resources would be “Small.”

FCS has an existing rail spur that was modified to ship waste and large components off-site. Land disturbance for the removal of large components is minimized, because removal is primarily conducted via the site rail system.

A section of the plant site that lies north of the rail spur and is bounded on the southwest by U.S. Highway 75 was determined as having Moderate to High Potential. It contains remnants of the former town of Desoto, Nebraska. Based on the impacts of past construction activities, the plant site being situated on floodplain alluvium, and having been developed since 1850, and the section of the site that lies south of the current Union Pacific rail spur should be categorized as having no potential for cultural resources, either prehistoric or historic.

Environmental review procedures have been put in place at FCS regarding undertakings that involve land disturbing activities in undisturbed surface and subsurface areas. These environmental protection procedures include contacting the NSHPO to establish the actions necessary to protect known or as of yet undiscovered cultural resources before an action is allowed to occur. The cultural, historic, and archaeological impact evaluation conducted in the GEIS focused on similar attributes as the SEIS. The GEIS evaluated direct effects such as land clearing and indirect effects such as erosion and siltation.

The conclusion for the license renewal evaluation is also applicable to the decommissioning period, because:

- decommissioning activities will be primarily contained to disturbed areas located away from areas of existing or high potential for archaeological sites,
- construction activities that disturb one acre or greater of soil are permitted by NDEQ approval, and BMPs are required to control sediment and the effects of erosion, and

- environmental protection procedures pertaining to archaeological and cultural resources will remain in effect during decommissioning.

Therefore, based on the updated environmental report, OPPD concludes that the impacts of FCS decommissioning on cultural, historic, and archaeological resources are “Small” and are bounded by the GEIS. Based on the historical information in the AEC Environmental Statement, the results of the reviews of historic, cultural, and archaeological resources performed in 2013 and 2014, current transportation methods for large components, and soil erosion control work practices, the decommissioning will have no significant impact on cultural and historic resources. Archaeological resources would be “Small.”

8.6.3.14 Aesthetics

The environmental impact evaluation associated with aesthetics in Section 4.3.15 of NUREG-0586, Supplement 1 has been determined to be generally applicable to FCS with a “Small” impact.

The impact of decommissioning on site aesthetics (e.g., visual skyline) is limited in terms of land disturbance and duration. These impacts are temporary and will cease when decommissioning is completed.

The location of the ISFSI is set back several hundred yards from the river frontage and located near the existing switchyard. Once all of the major plant structures and buildings on the riverbank are removed, aesthetics will improve by providing a more open view of the Missouri River. Restoration of the site to a more natural grade will result in a contiguous open view of the Missouri River.

Aesthetics will improve once the site is returned to open space. Therefore, the environmental impact associated with aesthetics is evaluated to be bounded by the GEIS, and the impact is considered “Small.”

8.6.3.15 Noise

The environmental impact evaluation associated with noise in Section 4.3.16 of NUREG-0586, Supplement 1 has been determined to be generally applicable to FCS with a “Small” impact.

FCS is located on the riverbank of the Missouri River. There are no residences within one half mile of the station, and no schools, hospitals, prison, motels, or hotels are in the immediate vicinity of the site. The center of the nearest community, Blair, Nebraska, is located approximately three miles to the northwest of the site.

Noise generation will primarily result from demolition activities involving heavy construction equipment. The noise from the shipment of waste will be minimal since the primary transportation method for shipment of LLRW will be by rail. Noise associated with decommissioning and shipment of waste is intermittent and temporary.

The ISFSI construction was completed in 2005. The ISFSI is a passive facility and there will be minimal noise generated from its operation. Once the decommissioning is complete, noise levels in the vicinity of the FCS site will be reduced to levels below those experienced during the operation of the facility.

Due to the distance of the station from sensitive receptors, there will be limited temporary impacts on noise levels during decommissioning and demolition activities. Therefore, the environmental impact associated with noise is evaluated to be bounded by the GEIS, and the impact is considered "Small."

8.6.3.16 Irretrievable Resources

During the demolition and structural dismantlement of the station, recycling and asset recovery efforts will be made. Some metals (e.g., from turbine, transformer components, etc.) have been released as clean scrap.

Uranium is a natural resource that is irretrievably consumed during power operation. After the plant is shutdown, uranium is no longer consumed. The use of the environment (air, water, land) is not considered to represent a significant irreversible or irretrievable resource commitment, but rather a relatively short-term investment. Since the FCS site will be decommissioned to meet the unrestricted release criteria found in 10 CFR 20.1402, the land is not considered an irreversible resource. LLRW has been, and will continue to be, shipped to the Energy*Solutions* disposal site in Clive, Utah. This facility has sufficient space for the disposal of this waste. In addition, Class B/C waste that is generated may also be shipped to the Waste Control Specialists facility in Andrews, Texas.

As stated in the GEIS, irretrievable resources that would occur during the decommissioning process are the materials used to decontaminate the facility (e.g., rags, solvents, gases, and tools) and fuel used for construction machinery and for transportation of materials to and from the site. These resource commitments are considered to be minor and are neither detectable nor destabilizing. Therefore, the environmental impact associated with irretrievable resources is evaluated to be bounded by the GEIS, and the impact is considered "Small."

8.6.3.17 Traffic and Transportation

The number of shipments and the volume of waste shipped are greater during decommissioning than during the operation of the facility. Non-radiological impacts of transportation include increased traffic and wear and tear on roadways. Because the majority of the waste will be transported by rail, the average number of daily shipments from the site will be relatively small. Consequently, it is anticipated that there will be no significant effect on traffic flow or road wear. The impacts of a transportation accident would be neither detectable nor destabilizing. Therefore, the environmental impact associated with traffic, transportation is evaluated to be bounded by the GEIS, and the impact is considered as "Small."

8.6.3.18 Placement of Clean Concrete Demolition Debris and Sand Mix in Major Building Basements: Terrestrial Ecology and Transportation

OPPD evaluated the use of clean concrete demolition debris for basement fill end state, and subsequently determined only clean fill is to be used.

8.7 Overview of Regulatory Governing Decommissioning Activities and Site Release

8.7.1 Federal Requirements

Decommissioning activities that are subject to federal regulations include:

- spent fuel and GTCC storage at the ISFSI
- handling, packaging, and shipment of radioactive waste
- worker radiation protection
- license termination and final site release
- worker health and safety
- liquid effluent releases
- hazardous waste generation/disposition
- handling and removal of asbestos
- characterization and removal of polychlorinated biphenyls (PCBs)
- handling and removal of lead paint

8.7.1.1 NRC

The majority of radiological activities falls under Title 10 of the Code of Federal Regulations and are administered by the NRC. Applicable Title 10 regulations include:

- Part 20 – Radiation protection
- Part 50 – Decommissioning activities
- Part 51 – Environmental protection
- Part 61 – Disposal of radioactive waste
- Part 71 – Packaging and transportation of radioactive waste (regulations in 49 CFR Parts 171 through 174 also apply)
- Part 72 – Licensing requirements for the independent storage of spent nuclear fuel, high-level radioactive waste, and reactor-related GTCC waste
- Part 73 – Physical protection of plants and materials

8.7.1.2 EPA

The EPA regulations outlined in Title 40 of the Code of Federal Regulations apply as follows:

- Part 61 – Asbestos handling and removal
- Parts 122 to 125 – NPDES
- Part 141 – Safe drinking water standards

- Part 190 – Radiation protection standards for nuclear power operations
- Parts 260 to 272 – Resource Conservation and Recovery Act
- Part 280 – Underground storage tanks
- Part 761 – Toxic Substance Control Act for PCBs
- Part 129-132 – Clean Water Act

8.7.2 State and Local Requirements

Permits and approvals from or notifications to state and local agencies are required for safety and environmental protection purposes. Decommissioning activities and related site operations that fall under State and local jurisdiction include, but are not limited to, the following:

- Nebraska Department of Environmental Quality
- Nebraska Department Health and Human Services
- Nebraska State Historical Society

This information provided above is a general overview of the applicable regulations and not intended to be all-inclusive.

8.8 Conclusion

As previously evaluated in the FCS PSDAR, the non-radiological environmental impacts from decommissioning FCS are temporary and not significant. The potential issues identified as “site-specific” in NUREG-0586, Supplement 1 (such as “Threatened” and “Endangered” species and environmental justice) have been evaluated, and there is no significant impact. The potential environmental impacts associated with decommissioning FCS have already been predicted in and will be bounded by the previously issued environmental impact statements (PSDAR, NUREG-0586, and FCS Environmental Statement). Therefore, there are no new or significant environmental change associated with decommissioning.

8.9 References

[1] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.179, Revision 2, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," 2019.

[2] "Updated Environmental Report Fort Calhoun Station File No. 127690-003" (LIC-20-0013, ML 20202A654), March 2019.

[3] U.S. Atomic Energy Commission, "Final Environmental Statement related to operation of Fort Calhoun Station Unit 1" (WIP 031037), March 1971.

[4] U.S. Nuclear Regulatory Commission, "NUREG-1437, Supplement 12, Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Fort Calhoun Station Unit 1 – Final Report," 2003.

[5] Omaha Public Power District, Fort Calhoun Station Unit 1, "Post Shutdown Decommissioning Activity Report".

[6] U.S. Nuclear Regulatory Commission, "NUREG-0586, Supplement 1, Volume 1, Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," 2002.

[7] Correspondence, Olsson Engineering to the Army Corps of Engineers, February 2021.

[8] U.S. Nuclear Regulatory Commission, "NUREG-1496, Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities," 1997.

[9] U.S. Census Bureau American Community Survey report - 2010.

[10] Radiation Safety and Control Services, "TSD 20-001, Historical Site Assessment for Fort Calhoun Station," 2020.

[11] National Centers for Environmental Information, National Oceanic and Atmospheric Administration, www.ncdc.noaa.gov/stormevents/.

[12] Omaha Public Power District, "FCS-SAF-103, FCS Deconstruction Health and Safety Plan".

[13] Omaha Public Power District, "CH-ODCM-0001, Offsite Dose Calculation Manual".

[14] Nebraska Department of Transportation, gis.ne.gov/portal/apps/webappviewer/index.html?id=bb00781d6653474d945d51f49e1e7c34.

[15] Omaha Public Power District, "Avian Protection Plan".

Table 8-1 Summary of the Environmental Impacts from Decommissioning

Section	Environmental Issue	GEIS	Impact
	Onsite/Offsite Land Use		
8.6.3.1	- Onsite Land Use	Yes	SMALL
8.6.3.2	- Offsite Land Use	No	Site Specific
8.6.3.3	Water Use	Yes	SMALL
8.6.3.4	Water Quality		
	Surface Water	Yes	SMALL
	Ground Water	Yes	SMALL
8.6.3.5	Air Quality	Yes	SMALL
8.6.3.6	Aquadic Ecology		
	Activities Within the Operational Area	Yes	SMALL
	Activities Beyond the Operational Area	No	Site Specific
8.6.3.7	Terrestrial Ecology		
	Activities Within the Operational Area	Yes	SMALL
	Activities Beyond the Operational Area	No	Site Specific
8.6.3.8	Threatened and Endangered Species	No	Site Specific
8.6.2	Radiological		
	Activities Resulting in Occupational Dose to Workers	Yes	SMALL
	Activiteis Resulting in Dose to the Public	Yes	SMALL
	Radiological Accidents	Yes	SMALL
8.6.3.9	Occupational Issues	Yes	SMALL
8.6.3.10	Cost*	N/A	N/A
8.6.3.11	Socioeconomics	Yes	SMALL
8.6.3.12	Environmental Justice	No	Site Specific
8.6.3.13	Cultural and Historic Impacts		
	Activities Within the Operational Areas	Yes	SMALL
	Activities Beyond the Operational Area	No	Site Specific
8.6.3.14	Astethics	Yes	SMALL
8.6.3.15	Noise	Yes	SMALL
8.6.3.17	Transportation	Yes	SMALL
8.6.3.16	Irretrevable Resources	Yes	SMALL
8.6.3.18	Placement of Clean Debris	No	Site Specific

Note: Cost, Section 4.3.11 in GEIS Supplement 1, is not evaluated using environmental significance levels and is not identified as a generic or site-specific issue.