

ATTACHMENT 25

CONSUMERS ENERGY
BIG ROCKPOINT

DOCKET NUMBERS 50-155 AND 72-043

TRANSMITTAL OF SURVEY PACKAGES IN SUPPORT OF BIG ROCK POINT PHASED
LICENSE TERMINATION

CLASS 1 FINAL STATUS SURVEY – SUBSURFACE (QUARRY), 09C_{q1}1,
SCREEN HOUSE AREA EXCAVATION

October 9, 2006

60 Pages

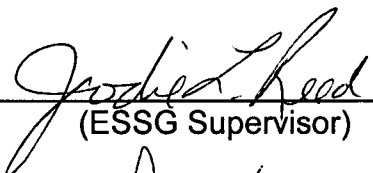
Final Status Survey 09C_{q1}1

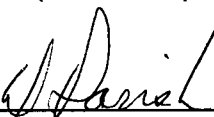
Screenhouse Area Excavation

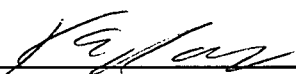
Survey Unit 09
Survey Date: 11-02-04

SURVEY PACKAGE CLOSURE

Final Status Survey Documentation is authorized for closure. All required reviews are complete and the evaluation of data results have satisfied the criteria established for unrestricted release and onsite use for excavation backfill.

Signed:  Date: 10-05-06
(ESSG Supervisor)

Signed:  Date: 10-05-06
(ES Superintendent)

Signed:  Date: 10-5-06
(RP & ES Manager)

This Survey Package Closure form is an addendum to original survey package 09C_{q1}1 conducted on 11-02-04. This addendum changes this survey package number to 09C_{q1}1. The addition of the subscript "q" identifies this as a final status support survey of an excavated surface. This nomenclature is consistent with guidance provided in RM-76, Final Status Survey Design. The performance of this survey predates the establishment of nomenclature criteria contained within RM-76. The data contained within this survey package remains intact, as originally approved on 12-20-04, with no changes or modifications.

Final Status Survey Area Requirements

Survey 09C₁1 Screenhouse Excavation Area

Survey Description

Final Status Survey 09C₁1 encompasses an area of 1820 m² at the northeast section of the Industrial Area in the former location of the Screenhouse Building. The survey area is an open excavation approximately eight meters below grade that results from demolition and removal of the following subsurface structures and components:

- Screenhouse intake and discharge foundations,
- Septic holding tanks and lift pump station,
- Emergency diesel storage tanks, and
- All support piping and conduit

The intake pipe that provided lake water to the Screenhouse for distribution to the Industrial Area is located inside the north boundary of the survey area. The pipe has been plugged and terminated in an anchor of concrete to maintain the barrier preventing lake water entry into the excavation area. The intake pipe will be abandoned in place to satisfy the requirements specified for "Greenfield Condition" as detailed in the License Termination Plan (LTP, Section 1.5). No other piping, components or materials of plant origin exist in this survey area.

History

During power operations the Screenhouse was the transfer station for well water, lake water, and fire-water supporting plant systems in the Industrial area. The Screenhouse also contained the process monitoring system for permitted effluent release to Lake Michigan from the liquid radwaste discharge line that once traversed this survey area. The Historical Site Assessment (HSA) has identified this survey unit to potentially contain residual radioactivity in area soils (LTP, 2E-1).

Current Radiological Status

Soil Characterization surveys and radiological evaluations for the release of demolition materials do not indicate the presence of residual radioactivity in this survey area. Based on proximity to material transport pathways and radioactive systems the radiological status of this survey area is Class 1. Input for this evaluation includes the following survey data:

- Characterization Survey Unit 9 (LTP, 2 E -52),
- BMR Survey Package 2004-0024,
- Final Status Survey 09C_{x1}1, dated 09-09-2004,
- Final Status Survey 09C_{x2}1, dated 09-17-2004,
- Final Status Survey 09C_{x3}1, dated 09-23-2004,
- Final Status Survey 09C_{x4}1, dated 10-01-2004,
- Final Status Survey 09C_{x5}1, dated 10-05-2004,
- Final Status Survey 09C_{x6}1, dated 10-18-2004.

Quality Assurance/Quality Control

As a minimum 5% of the sample population of this survey shall be selected for QA/QC verification in accordance with BRP Procedure RM-79, *Final Status Survey Quality Control*. Both split samples and sample recounts will take place. In addition, a minimum of 5% of the survey area will receive a verification scan. QA/QC soil samples and verification scan locations will be selected using the RAND function in Microsoft 2000 software program.

Additional Sample Analysis Requirements

This survey area intersects the identified waterborne pathway for Tritium migration and shall require Tritium in soil analyses for a minimum of 10% of the sample population. Soil for Tritium analysis will be collected in the same locations as those collected for QA/QC evaluation. Tritium samples will be sent to an independent laboratory for analysis.

Post-Construction Expectations

Survey 09C₁1 will be performed in the following activity sequence:

1. Walkdown: Site Characterization personnel will perform a walkdown assessment to insure survey area preparations are complete and confirm that the following post-construction expectations have been satisfied:
 - Groundwater and Surface water control is adequate
 - All construction debris has been removed from the survey area
 - The current survey area status meets all applicable safety requirements
2. Survey Area Isolation and Control: Control measures will be established to ensure that any potential ongoing decommissioning activities in adjacent locations do not impact the current survey area status. Isolation and control measures include postings, barriers, access points, and the evaluation of ongoing work activities in adjacent areas.
3. Survey Design and Execution: Survey design and execution will follow the Data Quality Objectives for Survey 09C₁1 in accordance with the survey requirements established in RM-76, *Final Status Survey Design*, RM-77, *Final Status Survey Implementation*, and LTP, Chapter 5. Survey size will be based on the statistical requirements of the Sign Test for Class 1 areas with soil samples collected in random start, systematic data point locations. Surface scanning will be performed with 100% survey area coverage. This survey will be conducted in accordance with approved BRP procedures and follow the guidance of NUREG 1575.
4. Data Quality Assessment: Isolation and control of the survey area will be maintained until the survey Data Quality Assessment demonstrates that the regulatory requirements for unrestricted site release have been satisfied. Once released for unrestricted use, this area will be backfilled and restored to original grade elevation.

DATA QUALITY OBJECTIVES

Survey 09C₁ Screenhouse Excavation Area

1. STATE THE PROBLEM

The Problem:

To demonstrate that the level of residual radioactivity in the excavated area of the former Screenhouse does not exceed the release criteria of 25 mrem/year Total Effective Dose Equivalent (TEDE) as specified in the License Termination Plan (LTP). This Class 1 survey area includes all exposed sub-surface soils in the former location of the Screenhouse. It must be demonstrated that this survey area meets the criteria established for unrestricted release prior to backfill and return to original grade elevation.

Stakeholders:

The primary stakeholders interested in the answer to this problem are Consumers Energy Co., and the general public as represented by the Michigan Department of Environmental Quality (MDEQ), and the US Nuclear Regulatory Commission (USNRC).

The Planning Team:

The planning team consists of members of the BRP Environmental Services Survey Group (ESSG). The primary decision maker will be the Final Status Survey (FSS) Supervisor. The Final Status Survey Supervisor will obtain input from the site Construction Group and Scheduling Group for issues relating to schedule and costs.

Schedule:

Approximately five (5) working days are projected to implement the Final Status Survey to collect and analyze field data.

Resources:

The primary resources needed to determine the answer to the problem are two (2) technicians to perform fieldwork, one (1) technician to prepare the samples and conduct laboratory analyses, and two (2) site characterization team members to prepare and review the design, generate maps, coordinate field activities and evaluate data.

2. IDENTIFY THE DECISION

Several decisions need to be defined to address the stated problem.

Principal Study Question (1):

Does the mean concentration of residual radioactivity in the survey unit exceed the release criteria stated above?

Decision (1):

Determine whether the mean concentration of residual radioactivity in the survey exceeds the release criteria stated in the problem.

Actions (1):

Alternative actions include failure of the survey unit, remediation, or no action required.

Principal Study Question (2):

Do any areas of elevated activity in the survey unit exceed the release criteria?

The Decision (2):

Determine if any areas of elevated activity in the survey unit exceed the release criteria.

Actions (2):

Alternative actions include confirmation and investigation, performing the elevated measurement comparison (EMC), remediation, or no action required.

Principal Study Question (3):

Is the potential dose from residual radioactivity in the survey unit ALARA as stated?

The Decision (3):

Determine if the potential dose from residual radioactivity in the survey unit is ALARA. ALARA requirements for soil remediation are defined in Chapter 4 of the LTP.

Actions (3):

Alternative actions include remediation or no action required.

3. IDENTIFY INPUTS TO THE DECISION

Information Needed:

Characterization measurements are required to define the radionuclides present and determine the extent and variability of residual radioactivity in the survey area for design and implementation of the FSS. Survey area classification, ALARA analysis, potential radionuclides of interest, and site-specific DCGL values are also required inputs to the decision process. The primary information required for evaluation is the analytical results of FSS measurements.

Source of the Information:

The soil sample data to be used for FSS development are the radionuclide-specific measurements of soil samples collected within the affected local coordinate grids during the characterization process. This data also include the results of multiple surveys performed during soil excavation and the removal of demolition debris. The ALARA analysis for potential soil remediation is provided in LTP, Section 4.4. Site-specific DCGL values and BRP radionuclides of interest are defined in LTP Section 5, Table 5-1 and Procedure RM-76, *Final Status Survey Design*.

The FSS will be conducted in accordance with LTP Section 5 for Class 1 areas and associated BRP survey procedures. Soil samples will be utilized for radionuclide-specific measurements in this evaluation.

4. BOUNDARIES OF THE STUDY

Boundaries of the Survey:

The target population for this survey is the upper 15 cm of soil in a defined survey area of 1820 m². The physical boundary includes the base elevation in the survey area and all non-vertical sides of the excavation where accessibility is unrestricted by regulatory concerns for industrial safety. The location of the Screenhouse excavation survey area can be identified on site maps in an area defined by local coordinates 5S-2N by 9E-15E.

Temporal Boundaries:

Scanning and sampling in this survey unit will only be performed during daylight hours during acceptable weather conditions. Collection of data will take place when surface conditions are most favorable. Surface soils must be free of excessive snow cover and significant standing water prior to surface scanning. Soils must be in a non-frozen state or fragmented for collection to satisfy BRP procedural sampling requirements. The anticipated start date for the survey is 11-02-04.

Constraints:

Cold weather or excessive rain conditions may effect the operation of electronic equipment. Adverse weather conditions that include accumulations of rain or snow may limit area access and delay survey efforts.

5. DEVELOP A DECISION RULE

The following decision rules have been developed to define a logical process for choosing among alternative actions for the principal study questions associated with this survey area.

Decision Rule (1):

If all reported concentrations for residual radioactivity are less than the site-specific DCGL's and the unity rule has been satisfied for each sample, then the survey unit meets release criteria. No further action is required.

Decision Rule (2):

If the mean value of activity in the survey unit is greater than the DCGL, then the survey unit fails to meet the release criteria.¹ Remediate, resurvey, and evaluate the results relative to the decision rule.

Decision Rule (3):

If the mean activity in the survey unit is less than the DCGL and any individual sample measurement exceeds this value conduct the Sign Test and the elevated measurement comparison (EMC) per LTP, Chapter 5 and Procedure RM-76, *Final Status Survey Design*. If the EMC and the Sign Test have been satisfied then the survey unit meets the release criteria and no further action is required. If the EMC or the Sign Test has not been satisfied then remediate the area(s) of elevated activity, resurvey as appropriate, and evaluate the results relative to the decision rule.

¹ When multiple radionuclides are present the mean activity value is determined as the average of the weighted sum. The DCGL of the weighted sum is 1.

Decision Rule (4):

If the potential dose from residual radioactivity in the survey unit is ALARA, then no further action is necessary. If the potential dose from residual radioactivity in the survey unit is not ALARA, then remediate and resurvey.

6. SPECIFY TOLERABLE LIMITS ON DECISION ERRORS

The Null Hypothesis:

It is assumed that residual radioactivity in the survey unit exceeds the release criterion.

Type I Error (α):

The α error is the maximum probability of rejecting the null hypotheses when it is true. The α error is defined in the LTP at a value of at 0.05 (5%) and cannot be changed to a less restrictive value unless prior approval is granted by the USNRC. The α error value of 0.05 will be used for survey planning and data assessment for this survey area.

Type II Error (β):

The β error is the probability of accepting the null hypothesis when it is false. A value of 0.05 (5%) will be used for survey planning and data assessment for this survey area.

The Lower Bound of the Gray Region (LBGR):

The LBGR is initially set at 0.5 for this survey unit. The LBGR may be adjusted during survey design to achieve an optimum relative shift between 1.0 and 3.0.

Relative Shift (Δ/σ):

The relative shift will be maintained within the range of 1.0 and 3.0 by adjusting the LBGR as appropriate.

7. OPTIMIZE DESIGN FOR OBTAINING DATA

Statistical Test

Sign Test:

Radionuclides of potential plant origin also present in soil as background activity resulting from fallout constitute only a small fraction of the DCGL. Therefore, the Sign Test will be used where applicable in the FSS evaluation to determine if the survey area meets the requirements for unrestricted release.

Number of Samples Determined:

The number of samples required for this survey will be determined based on the relative shift as defined by the requirements of the Sign Test (LTP, Chapter 5) and Procedure RM-76, *Final Status Survey Design*. The LBGR is initially set at 0.5 and may be adjusted as necessary for optimizing the survey design to achieve a relative shift between 1.0 and 3.0. Sample point locations are to be determined using a random start, systematic grid spacing. For sample point locations where access is impractical or unsafe, alternate locations will be randomly selected to achieve the sample size requirement.

Biased Sampling:

Co-60 is the most limiting radionuclide for identification by surface scanning; biased surface and subsurface core samples will be collected in any location that exceeds the scan investigation level.

Scan Coverage:

Scanning for this survey area will provide 100% coverage.

Number of Samples for Quality Control:

A minimum of 5% of the sample population will be collected for quality evaluation. These samples may include sample splits, sample recounts, or 3rd party sample analysis. Quality analyses will be conducted as defined in LTP, Chapter 5 and Procedure RM-79, *Final Status Survey Quality Control*.

Additional Sample Analysis Requirements:

An additional quantity of soil shall be collected for Tritium Analysis in the same locations as samples selected for QA/QC. A minimum of 10% of the sample population will be sampled. Tritium analyses will be performed by an independent laboratory. Data results will be provided in the FSS package.

Investigation Levels:

Investigation levels are defined in LTP, Chapter 5 and Procedure RM-76, *Final Status Survey Design*, by individual survey area classification; however, prior to regulatory approval of the LTP a more conservative approach for investigation will be established for this survey as shown below.

Investigation Levels for Survey 09C₁1

Classification	Scan Measurement	Soil Sample Analysis
Class 1	> DCGL	> DCGL _w

The investigation levels for soil sample measurements are meant to include any individual radionuclide result greater than the site-specific DCGL or where the combined radionuclide values exceed the unity rule. Co-60 is the most limiting radionuclide for identification by surface scanning; further investigation will be initiated at any location that exceeds the Co-60 Scan _{DCGL} of 1818 CPM above background as detailed in the survey design.

SURVEY DESIGN

Survey 09C₁1
Final Status Survey Design
Screenhouse Excavation Area

Survey Unit Description

Final Status Survey 09C₁1 encompasses an area of 1820 m² at the northeast section of the Industrial Area in the former location of the Screenhouse Building. The survey area is an open excavation approximately eight meters below grade that results from demolition and removal of the following subsurface structures and components:

- Screenhouse intake and discharge foundations,
- Septic holding tanks and lift pump station,
- Emergency diesel storage tanks, and
- All support piping and conduit

The intake pipe that provided lake water to the Screenhouse for distribution to the Industrial Area is located inside the north boundary of the survey area. The pipe has been plugged and terminated in an anchor of concrete to maintain the barrier preventing lake water entry into the excavation area. The intake pipe will be abandoned in place to satisfy the requirements specified for "Greenfield Condition" as detailed in the License Termination Plan (LTP, Section 1.5). No other piping, components or materials of plant origin exist in this survey area.

Soil Sample Design

Scoping Data

Sample measurements obtained to determine suitability for transport and FSS of excavated soil only identified background or MDA levels of residual radioactivity. As a conservative measure, input values for survey design were estimated based on surveys conducted for characterization of this survey area (LTP Appendix 2-E).

Table 1
Input Data for Survey Design (pCi/g)

Radionuclides	Cs-137	Co-60
σ	0.43	0.43
DCGL	11.93	3.21

Sample Requirements

The number of sample data points for this survey is based on the requirements of the Sign Test. The Unity Rule is used for the presence of multiple radionuclides. The Standard Deviation of the weighted sum is described by the following:

$$\sigma = \sqrt{\left(\frac{\sigma_{CS137}}{DCGL_{CS137}}\right)^2 + \left(\frac{\sigma_{CO80}}{DCGL_{CO80}}\right)^2}$$

$$\sigma = \sqrt{\left(\frac{0.43}{11.93}\right)^2 + \left(\frac{0.43}{3.21}\right)^2}$$

$$\sigma = 0.14$$

Relative Shift

The DCGL for the weighted sum is 1.0. The relative shift is determined using an LBGR value set at 72% of the DCGL_w.

$$\text{Relative Shift} = \frac{DCGL - LBGR}{\sigma}$$

$$\text{Relative Shift} = \frac{1 - 0.72}{0.14}$$

$$\text{Relative Shift} = 2.0$$

With α and β error levels set at 0.05 and the relative shift of 2.0, the Sign Test requires 15 sample data points (Table 5.5 NUREG 1575). As a conservative measure a minimum of 18 samples will be collected in this survey unit.

Sample Locations

Sample locations are selected in a random-start systematic pattern with the southwest corner of the survey unit as origin (X=0, Y=0). Two random numbers between 0 and 1 are generated using the RAND function within Microsoft 2000 Excel software program (Table 2). The numbers are applied to the survey unit maximum X and Y dimensions to determine the random start location.

Table 2
Random Numbers

Random #, X Axis	Random #, Y Axis
0.649064	0.777624

Survey Unit 09C₁1 Dimensions: X (E/W) = 53 meters
Y (N/S) = 69 meters

Random Start Location: X = (0.649064)(53) = 34.4 meters
Y = (0.777624)(69) = 53.7 meters

Using the site local coordinate grid, the origin of this survey unit is placed in Grid 214, X= 5.0 meters, Y= 6.5 meters. The random start location is located in Grid 122, X= 9.4 meters, Y= 0.2 meters.

Sample Spacing

Samples are located in a square grid pattern with sample spacing determined by the following:

$$L = \sqrt{\frac{A}{n}}, \quad \text{where } A = \text{area of survey unit and} \\ n = \text{number of samples.}$$

$$L = \sqrt{\frac{1820}{18}} = 10.1 \text{ meters}$$

With sample spacing established at 10.1 meters, 20 data point locations are available for survey as identified in Attachment 1.

QA/QC Sampling

A minimum of 5% of the sample population and 5% of the scan survey area are required to be selected for QA/QC verification in accordance with BRP Procedure RM-79, *Final Status Survey Quality Control*. As a conservative measure, three (3) soil samples and 10% of the scan survey area will be selected for QA/QC evaluation. Data point locations will be selected using the RAND function in the Microsoft 2000 Excel software program:

$\text{RAND()}*(b-a)+a$ where $a = 1$ and $b = \text{total number of soil samples to be collected}$.

Verification scan start point and track direction is determined using the above function. The first sample location selected will determine the start point of the verification scan, and the second sample location will determine the direction in which the scan will track. QA/QC location results are listed in Table 3:

Table 3
Random Numbers Generated for QA/QC

QA/QC Soil Samples	Random Sample Number	Verification Scan	Random Sample Number
Split Sample:	4	Start Point:	7
Sample Recount:	15	Scan Towards :	10
Sample Recount:	8	Minimum Scan Area Requirement:	182 m ²

Surface Scanning

The coverage requirement for surface scanning in this Class 1 area is 100%. The Scan_{MDC} has been established at fractional values of the DCGL_w for typical background activity levels at Big Rock Point. Scan_{MDC} values for varying backgrounds are provided in Attachment 2. The investigation level for identification of potential areas of elevated activity in this survey area will be the Scan_{DCGL} as defined by the following:

$$\text{SCAN}_{\text{DCGL}} = \text{Detector Rating} \frac{\text{CPM}}{\text{uR/hr}} * \text{Exposure Model} \frac{\text{uRi/hr}}{\text{pCi/g}} * \text{DCGL}_w$$

$$\text{Scan}_{\text{DCGL}} \text{ for Co-60} = 1818 \text{ cpm}$$

$$\text{Scan}_{\text{DCGL}} \text{ for Cs-137} = 3518 \text{ cpm}$$

Where:¹

$$\text{Detector Rating} = \frac{1200 \text{ CPM}}{\text{uR/hr}} \text{ Cs-137 and } \frac{565 \text{ CPM}}{\text{uR/hr}} \text{ Co-60}$$

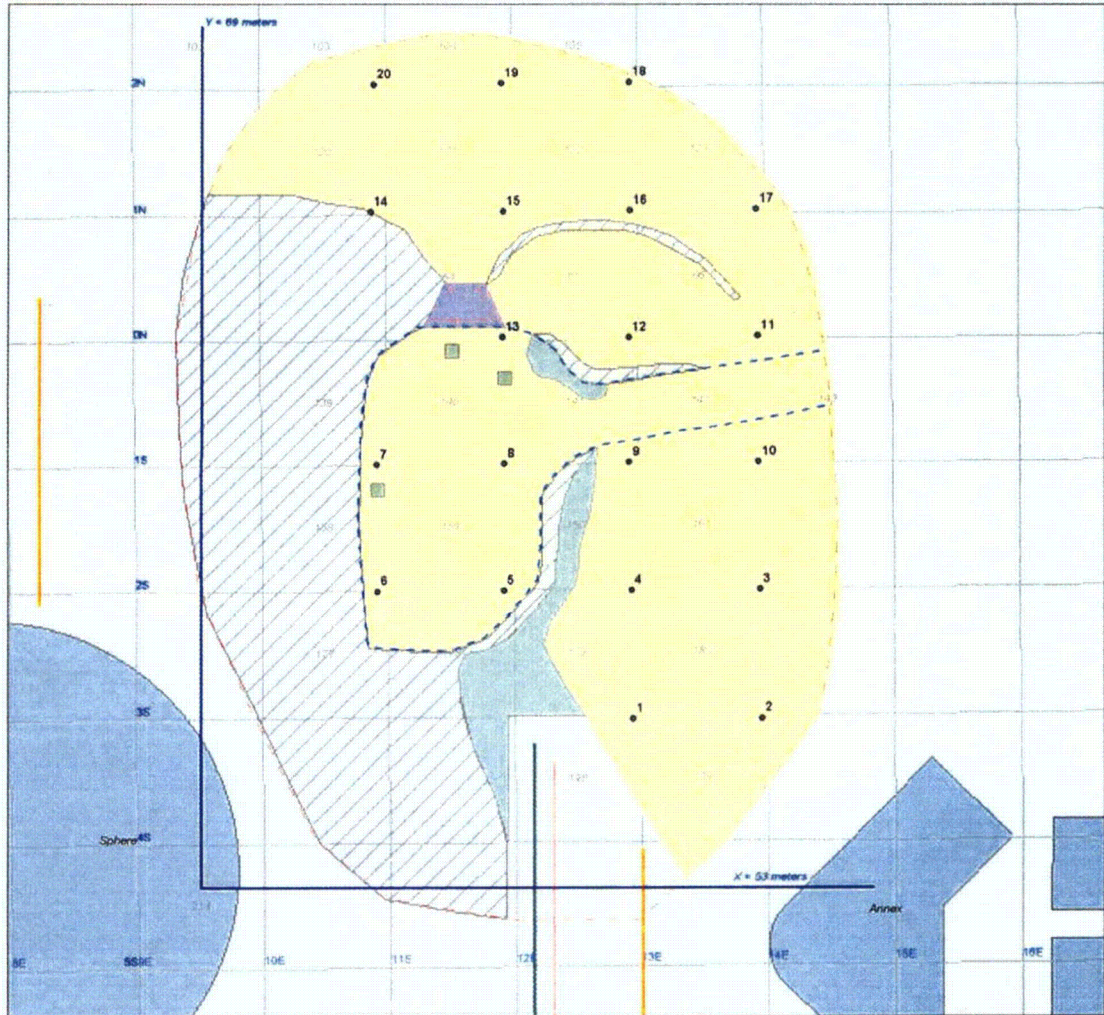
$$\text{Exposure Model} = \frac{1.229 \text{ uRi/hr}}{5 \text{ pCi/g}} \text{ Cs-137 and } \frac{5.029 \text{ uRi/hr}}{5 \text{ pCi/g}} \text{ Co-60}$$

$$\text{DCGL}_w = 11.93 \text{ pCi/g Cs-137 and } 3.21 \text{ pCi/g Co-60}$$

The DCGL_w for Co-60 is the most limiting value for scanning measurements performed to identify areas of potentially elevated activity. Scanning conducted for this Final Status Survey will assume all residual radioactivity to originate from Co-60 and the instrument response at the Co-60 DCGL_w (1818 cpm) will be used as the scanning investigation level for Survey 09C₁.

¹ These values established in EA-BRP-SC-0201, *Nal Scanning Sensitivity For Open Land Survey*

Attachment 1
Final Status Survey 09C,1
Soil Sample Locations
Screenhouse Excavation Area
11-02-2004



Legend

- Soil Sample Locations
- Sumps
- - - Extent of Excavation
- - - Ramp and Footprint of Base Elevation
- - - RCP Discharge Pipe
- - - Stormwater Drain Pipe
- - - Batch Pipe
- Intake Pipe Anchor Footprint

- Survey Area
- Site Local Coordinate Grid
- Safety Hazard: Unable to Access
- Engulfment Hazard
- Near Vertical Slopes

0 5 10 20 Meters

Attachment 1
Soil Sample Coordinate Locations
Final Status Survey 09C,1
Screenhouse Excavation Area
11-02-2004

Sample No.	Grid	X Coord.	Y Coord.
1	198	9.4	9.8
2	199	9.5	9.8
3	180	9.5	9.9
4	179	9.4	9.9
5	178	9.3	9.9
6	177	9.2	9.9
7	158	9.2	0
8	159	9.3	0
9	160	9.4	0
10	161	9.5	0

Sample No.	Grid	X Coord.	Y Coord.
11	66	9.5	0.1
12	65	9.4	0.1
13	64	9.3	0.1
14	120	9.2	0.2
15	121	9.3	0.2
16	122	9.4	0.2
17	123	9.5	0.2
18	105	9.4	0.3
19	104	9.3	0.3
20	103	9.2	0.3

*Sample No. 16 is the Random Start Location

Attachment 2

Scan MDC In Varying Backgrounds

				CPM	MDER uR/hr		Scan MDC pCi/g	
Background	d'	i	s _i	MDCR _{surveyor}	Cs-137	Co-60	Cs-137	Co-60
2000	2.48	4	28.64	607.47	0.51	1.08	2.06	1.07
2500	2.48	4	32.02	679.18	0.57	1.20	2.30	1.20
3000	2.48	4	35.07	744.00	0.62	1.32	2.52	1.31
3500	2.48	4	37.88	803.61	0.67	1.42	2.72	1.41
4000	2.48	4	40.50	859.10	0.72	1.52	2.91	1.51
4500	2.48	4	42.95	911.21	0.76	1.61	3.09	1.60
5000	2.48	4	45.28	960.50	0.80	1.70	3.26	1.69
5500	2.48	4	47.49	1,007.38	0.84	1.78	3.42	1.77
6000	2.48	4	49.60	1,052.17	0.88	1.86	3.57	1.85
6500	2.48	4	51.63	1,095.14	0.91	1.94	3.71	1.93
7000	2.48	4	53.57	1,136.48	0.95	2.01	3.85	2.00
7500	2.48	4	55.45	1,176.37	0.98	2.08	3.99	2.07
8000	2.48	4	57.27	1,214.95	1.01	2.15	4.12	2.14
8500	2.48	4	59.04	1,252.34	1.04	2.22	4.25	2.20
9000	2.48	4	60.75	1,288.65	1.07	2.28	4.37	2.27
9500	2.48	4	62.41	1,323.96	1.10	2.34	4.49	2.33
10000	2.48	4	64.03	1,358.35	1.13	2.40	4.61	2.39
10500	2.48	4	65.61	1,391.90	1.16	2.46	4.72	2.45
11000	2.48	4	67.16	1,424.65	1.19	2.52	4.83	2.51
11500	2.48	4	68.67	1,456.67	1.21	2.58	4.94	2.56
12000	2.48	4	70.14	1,488.00	1.24	2.63	5.04	2.62
12500	2.48	4	71.59	1,518.68	1.27	2.69	5.15	2.67
13000	2.48	4	73.01	1,548.76	1.29	2.74	5.25	2.73
13500	2.48	4	74.40	1,578.26	1.32	2.79	5.35	2.78
14000	2.48	4	75.77	1,607.22	1.34	2.84	5.45	2.83
14500	2.48	4	77.11	1,635.67	1.36	2.89	5.55	2.88
15000	2.48	4	78.42	1,663.63	1.39	2.94	5.64	2.93
Modeled Exposure (uR/hr) @ 5 pCi/g								
	Cs-137	1.23E+00						
	Co-60	5.03E+00						

Attachment 3

Area Factors for Open Land Survey Evaluation

Contaminated Area (m ²)	Calculated Area Factors at Time of Peak Dose								
	H-3	Mn-54	Fe-55	Co-60	Sr-90	Cs-137	Eu-152	Eu-154	Eu-155
8094	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
4047	1.00	1.01	1.00	1.01	1.00	1.02	1.02	1.01	1.02
2024	1.00	1.03	1.00	1.03	1.00	1.03	1.03	1.03	1.03
1012	1.35	1.04	1.00	1.04	1.00	1.04	1.05	1.04	1.04
506	2.91	1.09	1.98	1.08	1.98	1.13	1.07	1.07	1.06
253	6.05	1.14	3.95	1.13	3.94	1.20	1.11	1.11	1.09
126	12.4	1.20	7.93	1.20	7.87	1.29	1.17	1.16	1.14
63	24.9	1.30	15.8	1.30	15.6	1.41	1.27	1.26	1.23
32	49.2	1.49	31.2	1.49	30.5	1.62	1.44	1.45	1.39
16	98.9	1.78	62.0	1.78	59.9	1.93	1.72	1.73	1.63
8	198	2.38	123	2.38	117	2.58	2.30	2.31	2.14
4	397	3.61	243	3.62	230	3.91	3.49	3.52	3.19
2	794	5.68	473	5.75	452	6.14	5.48	5.55	4.90
1	1590	9.57	905	9.73	887	10.3	9.24	9.39	7.88

RM-76-5
FINAL STATUS SURVEY APPROVAL
AND AUTHORIZATION FOR IMPLEMENTATION

Survey Code FSS 09C₁1


Survey Area Description:

Final Status Survey 09C₁1 encompasses an area of 1820 m² in the northeast section of the Industrial Area at the former location of the Screenhouse Building. This survey area is an open excavation that extends to approximately eight meters below grade. This survey unit is a Class 1 area.

The survey area is authorized for Final Status Survey Implementation.


Designed by

11-01-04
Date


Technical Review by

11-01-04
Date

RM-77-1
SURVEY IMPLEMENTATION CHECKLIST
Page 1 of 3

Step
(+)

Initial

Date

1.0 PREPARATION FOR SURVEY 09C.1
Survey #

1.1 Survey Area Status:

- ✓ a. Final Status Survey Design has been approved for implementation (see RM-76-5, Final Status Survey Approval and Authorization for Supplementation).
1. Survey area walkdown complete
 2. Survey area determined ready for FSS
 3. Decommissioning activities that may impact the environmental status of the survey area have been completed.
 4. Survey area environment is controlled by barriers and postings or other approved method to restrict access.

W. Davis 11/01/04
ESSG

- ✓ b. Survey area has been turned over to the Environmental Services Survey Group (ESSG) in acceptable condition for FSS.

W. Davis 11/01/04
ESSG

1.2 Field Preparation:

- ✓ a. Survey unit boundaries delineated (Step 6.1.1)
- ✓ b. Statistical soil samples predetermined in the survey design are located and marked within the survey unit. (Step 6.1.2)
- ✓ c. Soil sample locations verified (Step 6.1.2.c)
- ✓ d. Instruments and equipment have been collected and calibrated for data measurement and collection (Step 6.1.3)
- ✓ e. Field documentation is prepared (Step 6.1.4)


W. Davis 11/01/04
ESSG

RM-77-1
SURVEY IMPLEMENTATION CHECKLIST
Page 2 of 3

		<u>Initial</u>	<u>Date</u>
2.0	DATA COLLECTION		
2.1	Soil Survey:		
<u>✓</u>	All soil samples collected and controlled (Step 6.2.1).	<u>MDL</u> ESSG	<u>11/02/04</u>
2.2	Surface Scan:		
<u>✓</u>	Surface Scan complete. Action response requirements have been conducted on any identified areas exceeding the investigation level (Step 6.3).	<u>MDL</u> ESSG	<u>11/02/04</u>
2.3	Judgmental Soil Samples:		
<u>✓</u>	a. Judgmental soil samples have been collected and controlled (Step 6.2.3).		
<u>NA</u>	b. Deep core profiles performed in areas identified to contain elevated residual activity (Step 6.2.3).	<u>MDL</u> ESSG	<u>11/02/04</u>
3.0	SAMPLE PREPARATION AND LABORATORY ANALYSIS		
3.1	Sample Preparation (Step 6.4.1):		
<u>✓</u>	a. Soil samples are homogenous		
<u>✓</u>	b. Soil samples are visibly dry prior to packing		
<u>✓</u>	c. Non-soil materials have been removed from sample		
<u>✓</u>	d. Soil samples have been transferred to one-liter Marinelli containers and are labeled and sealed.	<u>MDL</u> ESSG	<u>11/03/04</u>

RM-77-1
SURVEY IMPLEMENTATION CHECKLIST
Page 3 of 3

- | | | <u>Initial</u> | <u>Date</u> |
|----------|--|--------------------|-----------------|
| 3.2 | Laboratory Analysis: | | |
| <u>✓</u> | Isotopic analyses are complete. The spectroscopy report requires a signature of completion by the laboratory analyst and a signature of evaluation documenting that a second level review has been performed (Step 6.4.2). | <u>MAC</u>
ESSG | <u>11/04/04</u> |
| 3.3 | Sample Control and Documentation: | | |
| | Chain of custody documentation exhibits control of soil samples (Step 6.4.3). | <u>MAC</u>
ESSG | <u>11/04/04</u> |

<u></u>	<u>11/10/04</u>
Reviewed by	Date

Date: 11-02-2004	Time: 08:41	Location: Screenhouse Excavation	Tech: jlr/trs
SURVEY IDENTIFICATION / DESCRIPTION			
Survey 09C ₁ , Final Status Survey of Screenhouse Excavation Area.			
SURVEY TYPE			
Survey Type:	Characterization <input checked="" type="checkbox"/> Scan (Motive)	Remediation <input checked="" type="checkbox"/> Final Scan (Static)	
SURVEY DESIGN			
Sample Collection:	Judgmental <input type="checkbox"/> Random <input checked="" type="checkbox"/> Systematic	Scan Coverage: 100 %	
ANALYSIS			
Inst./Serial No. Detector 6 Scan 23501/186201	DAILY CHECK: <input checked="" type="checkbox"/> SAT <input type="checkbox"/> UNSAT INIT: _____		
Investigation Of Unidentified Peaks:	N/A <input checked="" type="checkbox"/> SAT <input type="checkbox"/> UNSAT INIT: _____		
Minimum Detectable Activity (Section 5.3.3)	X SAT <input type="checkbox"/> UNSAT INIT: _____		
COMMENTS			
Final status survey of the screenhouse excavation involved 100% mobile scan with a NaI Detector and soil samples. Sample Design: Random start with a square grid systematic pattern. Twenty statistical soil samples were collected all of which showed concentrations of radioactivity at a fraction of the DCGL _w . 100% mobile scan did not identify activity above established background. Judgmental samples were taken at each sump location. Sample data results showed no activity of plant origin.			
Technician Signature: [Signature]		Date: 11-04-04	
Second Level Review: Signature: [Signature]		Date: 11/10/04	

Activity Summary
Final Status Survey 09C,1
Excavated Soil from Screenhouse Area
11-02-2004

Excavation Area

delete date

Sample No.	Grid	X Coord.	Y Coord.	Cs-137 (pCi/g)		Co-60 (pCi/g)	
				Activity	MDA	Activity	MDA
1	198	9.4	9.8	nd	0.05	nd	0.06
2	199	9.5	9.8	nd	0.05	nd	0.07
3	180	9.5	9.9	nd	0.05	nd	0.06
4	179	9.4	9.9	nd	0.04	nd	0.07
5	178	9.3	9.9	nd	0.05	nd	0.06
6	177	9.2	9.9	nd	0.04	nd	0.06
7	158	9.2	0.0	nd	0.05	nd	0.07
8	159	9.3	0.0	nd	0.05	nd	0.06
9	160	9.4	0.0	nd	0.06	nd	0.06
10	161	9.5	0.0	nd	0.05	nd	0.06
11	66	9.5	0.1	0.04		nd	0.07
12	65	9.4	0.1	0.11		0.09	
13	64	9.3	0.1	nd	0.04	nd	0.05
14	120	9.2	0.2	nd	0.05	nd	0.05
15	121	9.3	0.2	0.05		nd	0.06
16	122	9.4	0.2	0.08		nd	0.05
17	123	9.5	0.2	0.18		nd	0.07
18	105	9.4	0.3	0.87		nd	0.07
19	104	9.3	0.3	0.07		nd	0.05
20	103	9.2	0.3	1.07		nd	0.05
J1	140	5.1	9.1	nd	0.05	nd	0.07
J2	140	9.5	6.8	nd	0.07	nd	0.07
J3	158	9.4	8.0	nd	0.04	nd	0.06

*Coordinate location relative to SW Corner of survey unit where X=0 m. and Y=0 m.

Note: nd indicates activity not detected above MDA values.

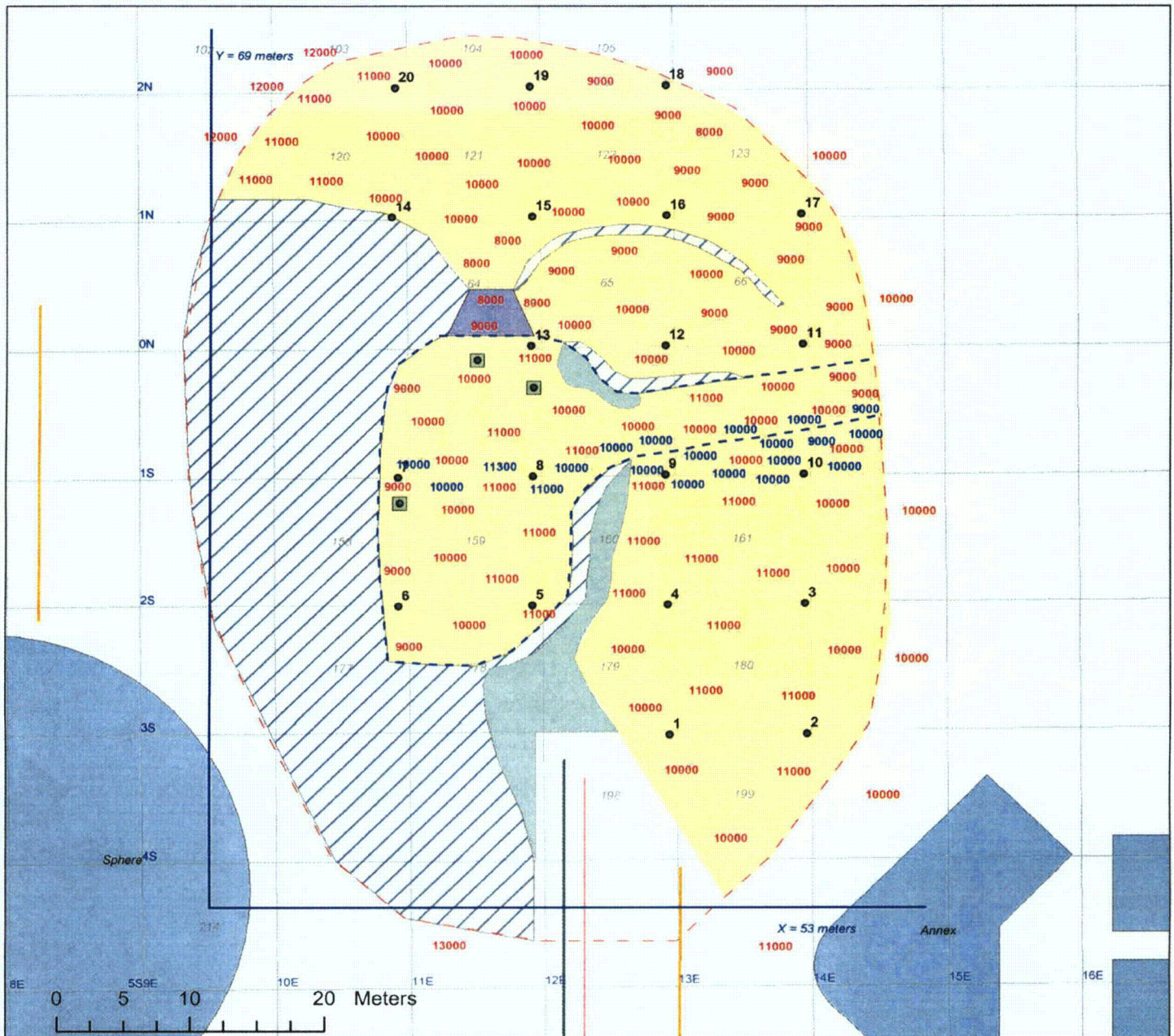


- Soil Sample Locations
- Sumps
- - - Extent of Excavation
- - - Ramp and Footprint of Base Elevation
- RCP Discharge Pipe
- Stormwater Drain Pipe
- Batch Pipe
- Intake Pipe Anchor Footprint

Site Local Coordinate Grid

Engulfment Hazard
Near Vertical Slopes

**Final Status Survey 09C,1
Mobile Scan
Screenhouse Excavation Area
11-02-2004**



Legend

- Soil Sample Locations
- Sumps
- - - Extent of Excavation
- - - Ramp and Footprint of Base Elevation
- RCP Discharge Pipe
- Stormwater Drain Pipe
- Batch Pipe
- Intake Pipe Anchor Footprint

- Survey Area
- Site Local Coordinate Grid
- Safety Hazard: Unable to Access**
- Engulfment Hazard
- Near Vertical Slopes

Primary Scan : 100 %

Technician Signature: [Signature] Date: 11-02-04
Time: 11:35

QC Verification Scan: 10+ %

Technician Signature: [Signature] Date: 11-02-04
Time: 12:00

Numbers in **Red** indicate Average General Area Activity (cpm)
Identified During Mobile Scan

Numbers in **Blue** indicate Average General Area Activity (cpm)
Identified During QC Verification Scan

SAMPLE CHAIN-OF-CUSTODY

FINAL STATUS SURVEY O9C.1
SCREENHOUSE EVALUATION AREA

11-02-04

PAGE 1 OF 2

RM-72-1

CHAIN-OF-CUSTODY RECORD

Sample Number	Sampling Location	Date	Time	Final Disposition of Sample
1	GR10193 (9.4)(9.8)	11/02/04	09:51	77B
2 (GR10194)	GR10199 (9.5)(9.8)		09:54	77B
3	GR10180 (9.5)(9.9)		09:56	77C
4 (SPLIT ² 2)	GR10179 (9.5)(9.9)		10:06	77C
5	GR10178 (9.3)(9.9)		08:41	77C
6	GR10177 (9.2)(9.9)		08:47	77C
7	GR10158 (9.2)(0.0)		08:54	77D
8 (REWORK)	GR10159 (9.3)(0.0)		09:04	77D
9	GR10160 (9.4)(0.0)		09:23	77D
10	GR10161 (9.5)(0.0)		10:16	77D
11	GR10166 (9.5)(0.1)		10:19	77D
12	GR10165 (9.4)(0.1)		10:32	77E
13	GR10164 (9.3)(0.1)		09:27	77E
14	GR10120 (9.2)(0.2)		10:44	77E
15 (REWORK)	GR10121 (9.3)(0.2)		10:28	77E
16	GR10122 (9.4)(0.2)		10:23	77E
17	GR10123 (9.5)(0.2)		10:38	77F
18	GR10105 (9.4)(0.3)		1041	77F
19	GR10124 (9.3)(0.3)		1043	77F
20	GR10103 (9.2)(0.3)	✓	10:47	77F

(Samples may be analyzed and stored, shipped for offsite evaluation or analyzed and disposed of.)

1. Relinquished by: <i>Joshua Reed</i>	Date 11/02/04	Time 10:58	Received in good condition by: locked in Chem Lab locker
2. Relinquished by: <i>Joshua Reed</i>	Date 11/02/04	Time 12:28	Received in good condition by: locked in oven in Chem Lab
3. Relinquished by: <i>Joshua Reed</i>	Date 11/03/04	Time 11:20	Received in good condition by: locked in Chem Lab locker
4. Relinquished by: <i>Joshua Reed</i>	Date 11/10/04	Time 15:35	Received in good condition by: Permanent Storage

RM-72.doc

WORKING CONTROLLED COPY

RM-72-1
CHAIN-OF-CUSTODY RECORD

[illegible]

(Samples may be analyzed and stored, shipped for offsite evaluation or analyzed and disposed of.)

1. Relinquished by: <i>Joshua Reed</i>	Date <i>11/02/04</i>	Time <i>10:58</i>	Received in good condition by: <i>Locked in Chem Lab Locker</i>
2. Relinquished by: <i>Joshua Reed</i>	Date <i>11/02/04</i>	Time <i>12:28</i> <i>12:28</i>	Received in good condition by: <i>Locked in oven in Chem Lab</i>
3. Relinquished by: <i>Joshua Reed</i>	Date <i>11/07/04</i>	Time <i>1020</i>	Received in good condition by: <i>Locked in Chem Lab Locker</i>
4. Relinquished by: <i>Joshua Reed</i>	Date <i>11/07/04</i>	Time <i>15:35</i>	Received in good condition by: <i>Permanent Storage</i>

RM-78-3
DATA ASSESSMENT REPORT
Page 1 of 8

FINAL STATUS SURVEY: 09C₁1

1.0 DATA VERIFICATION

1.1 Data Acceptance

 x Review the Implementation Checklist (RM-77-1) to verify that survey isolation and control measures were executed prior to FSS and are being maintained.

 x Review RM-77, Final Status Survey Implementation, to verify that methods, techniques, and survey activities required for FSS have been applied in accordance with the appropriate procedures.

1.2 Field QC Records:

 n/a Review all assessments, Condition Reports and audits to ensure that identified issues have been resolved.

Comments: _____

 x Verify scan instrumentation was in calibration and the QC source checks were performed prior to and after surveys.

 x Verify daily QC source checks for Canberra gamma spectroscopy detector properly logged prior to soil sample analysis.

1.3 Review Verification:

 x Verify that the Data Quality Objectives are complete.

 x Verify that the survey design has been technically reviewed.

RM-78-3
DATA ASSESSMENT REPORT
Page 3 of 8

2.0 DATA VALIDATION

2.1 Documentation Review:

Perform documentation review for quality control purposes and validate the data collected is complete and appropriate for use as defined by the survey design. Documentation includes:

<u> x </u>	Field measurement records
<u> x </u>	Chain-of-custody
<u> x </u>	Quality Control (QC) measurement records
<u> x </u>	Current qualification of survey personnel
<u> n/a </u>	Corrective Action Reports
<u> x </u>	Data inputs (laboratory spectroscopy)
<u> x </u>	Sample preparation techniques

2.2 Detection Limit Review:

<u> x </u>	Scan MDCs are below established site DCGLs.
<u> x </u>	Forced-count values are assigned as necessary when activity is not detected in a sample.
<u> x </u>	Minimum Detectable Concentration (MDC) values of gamma spectroscopy are below established DCGLs.

2.3 Quality Control (QC) Data Review:

<u> x </u>	Quality Control (QC) data results have received required reviews and are complete and consistent.
<u> x* </u>	Results of judgmental samples have been reviewed and evaluated. <i>*Refer to RM-78-3, Attachment 1 for details.</i>
<u> x </u>	Review to ensure that the analytical results of judgmental samples do not impact the evaluation for unrestricted release of the survey area.

RM-78-3
DATA ASSESSMENT REPORT
Page 4 of 8

2.4 Qualification of Data:

Statistical radionuclide-specific measurements for completeness. Evaluate the survey for determination of data usability and confirm that sufficient qualified data are present for the decision process.

a. Total number of statistical samples planned for the survey: 18

b. Total number of statistical samples determined as valid: 20

c. Calculate % Completeness: $\frac{b \times 120}{a} = \underline{133\%}$

x Qualified data are $\geq 100\%$ completeness and are sufficient to support the Sign Test requirement for determination of unrestricted release.

Data Validation Completed: Yes No

Comments: Refer to RM-78-3 Attachment 2 for verification of analysis results, 2.4.a, 2.4.b,
and 2.4.c.
Refer to the Additional Sampling Requirements section for Tritium in Soil data
results.


Assessor

11-10-04
Date

RM-78-3
DATA ASSESSMENT REPORT
Page 5 of 8

3.0 DATA QUALITY ASSESSMENT

3.1 Review the DQOs and Survey Design:

- x Confirm that all inputs to the decision have been reviewed and are complete.
- x Verify that boundaries or constraints identified in the survey area have not affected the quality of the data.
- x Review the Statement of Hypothesis and confirm that it remains relevant.
- x Confirm that Type I and Type II error limits are consistent with DQOs.
- x Confirm that the survey design is consistent with DQOs and that the appropriate number of data points were obtained.

3.2 Preliminary Review:

3.2.1 Preliminary Evaluation:

- x Quality Assessment (QA) reports consistent with procedure RM-79, Final Status Survey Quality Control.
- x Survey is of sufficient intensity to satisfy classification requirement.
- x Potential trends of radioactivity levels in the survey area do not impact a decision for unrestricted release.

Comments: _____

RM-78-3
DATA ASSESSMENT REPORT
Page 6 of 8

3.2.2 Calculate Basic Statistical Quantities:*

- a. Number of qualified data points 20
- b. Calculation of the Mean 0.013 (SOR)
- c. Calculation of the Median 0.007 (SOR)
- d. Calculation Standard Deviation 0.020 (SOR)

**Note: Statistical Quantities are calculated on RM-78-3, Attachment 2.*

n/a Attach graphic representation of the data if any radionuclide-specific measurements exceed 50% of the DCGL.

x Sample QA/QC measurements consistent with FSS data.

3.3 Statistical Evaluation:

NOTE: If all measurement data are less than the $DCGL_w$, statistical testing is not required and the survey unit meets the regulatory requirement for unrestricted release.

x All survey measurements are below the $DCGL_w$.

3.3.1 Verify Assumptions of the Statistical Test

n/a Review the posting plot to verify that if data exhibits spatial independence. Spatial trends must be investigated and resolved prior to further assessment.

n/a Review to verify dispersion symmetry. The appearance of skewed data must be investigated for cause and documented prior to further assessment.

RM-78-3
DATA ASSESSMENT REPORT
Page 7 of 8

n/a Review the dataset standard deviation and range for data variance. Questionable data must be investigated for cause and documented prior to further assessment.

n/a Compare the prospective power curve with the retrospective power curve. Verify that the data exhibits adequate power and confirm that the sample size is sufficient to satisfy the DQOs.

3.4 Draw Conclusions from the Data:

3.4.1 Investigation Levels and Response Actions

x Determine if data results have exceeded any investigation level. Document findings.

3.4.2 Evaluation for Unrestricted Release

Select applicable conclusion:

x Survey area acceptance criteria met and survey area satisfies the requirements for unrestricted release:

x All concentrations are less than the $DCGL_w$. The Null Hypothesis is rejected.

n/a The mean concentration of the survey area is below the $DCGL_w$ but individual measurements in the survey unit exceed the $DCGL_w$. The Sign Test and EMC evaluation are successful and the Null Hypothesis is rejected.

RM-78-3
DATA ASSESSMENT REPORT
Page 8 of 8

n/a Survey area acceptance criteria not met and survey area fails to satisfy the requirements for unrestricted release:

n/a The mean concentration in the survey area exceeds the DCGL_w and the null hypothesis is confirmed.

n/a The mean concentration of the survey area is below the DCGL_w but individual measurements in the Unit exceed the DCGL_w. The Sign Test and EMC evaluation are unsuccessful and the null hypothesis is confirmed.

Data Quality Assessment Completed: ☒ Yes ☐ No

Comments _____



Assessor

11/10/04

Date

Reviews:



Technical Review

11/10/04

Date



ES Superintendent

11/30/04

Date



RP&ES Manager

12-20-04

Date

**RM 78-3, Attachment 1
Lake Water Intake Pipe
Final Status Survey 09C₁1
Screenhouse Excavation Area**

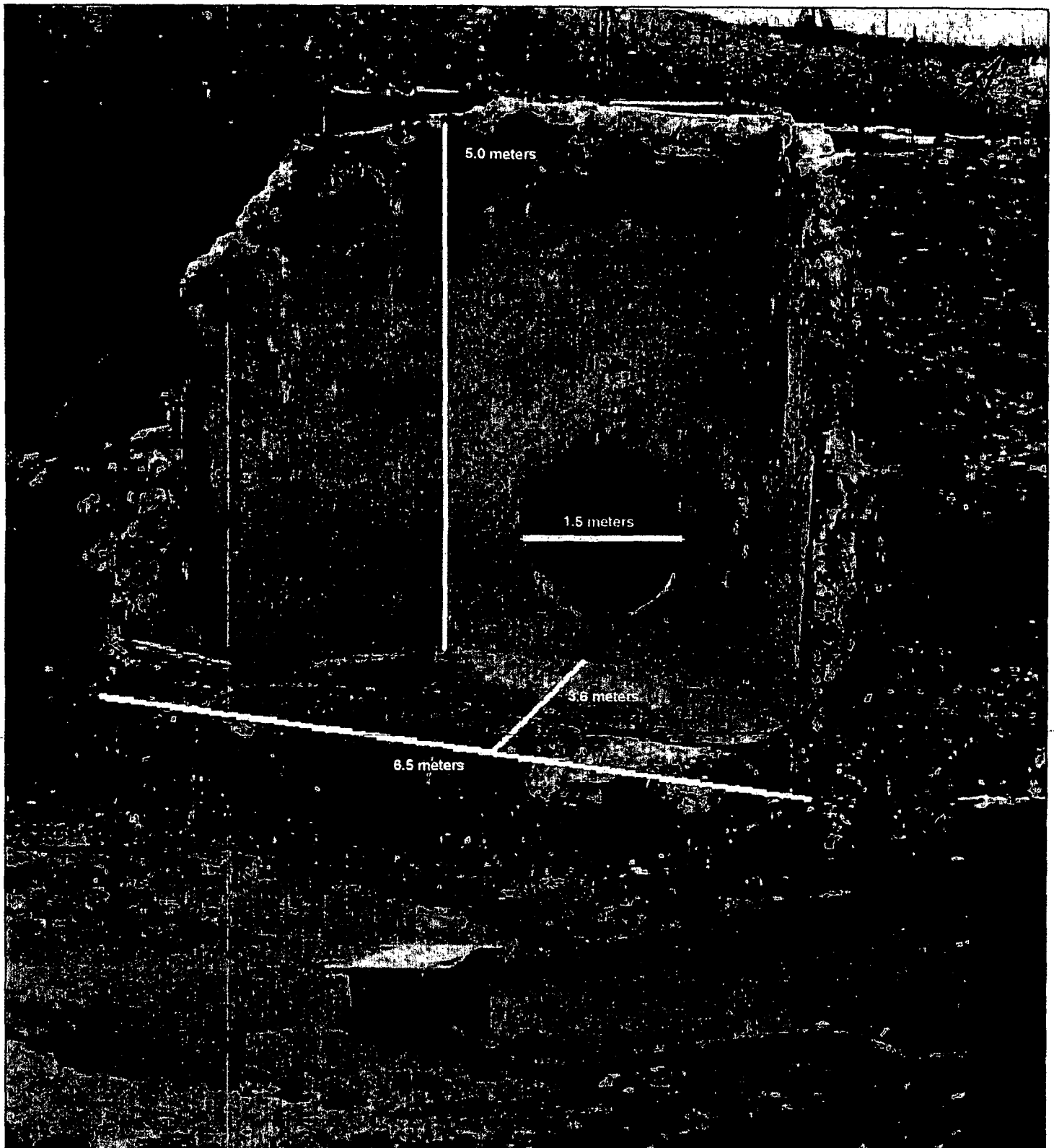
The lake water intake pipe extends below grade from Lake Michigan to the north boundary of the screen house excavation. The pipe has been plugged and terminates in an anchor of concrete that provides structural stability and maintains the barrier preventing lake water entry into the excavation area.

The Historical Site Assessment (HSA) documents an operational event that once allowed radioactive contaminants to briefly enter the screen-house intake bay from the discharge canal in Lake Michigan (LTP, Appendix 2-b, Event Data 57). These contaminants were the result of wave-suspended radioactivity that was previously discharged to the lake by permitted release. The exposed surfaces of the intake pipe and anchor were in the contaminant flow path during this event; however, the potential for any residual radioactivity to be present in these components is considered to be extremely remote. The following surveys were conducted to verify the radiological status of these components:

- Prior to screen-house demolition, the intake bay was surveyed in accordance with the provisions of 10 CFR 20.2002 for disposition of demolition debris to a State of Michigan Type II landfill. All measurements were consistent with standard background radioactivity levels (BMR survey package 2004-0024, rooms 500/501/502).
- Laboratory analyses of soils adjacent to the intake pipe and anchor do not identify the presence of reactor generated residual radioactivity
- Laboratory analyses of judgmental samples collected from the concrete anchor do not identify the presence of reactor generated radioactivity
- ~~Horizontal surfaces of the anchor block scanned with the NaI detection system have not identified any potential areas of elevated radioactivity~~

Based on historical data, process knowledge, and the results of the survey analyses noted above, the intake pipe and concrete anchor do not impact the conclusion for unrestricted release of this survey unit. These components will be abandoned in place to satisfy the requirements specified for "Site Greenfield" as detailed in the License Termination Plan (LTP, Section 1.5).

RM-78-3, Attachment 1
Concrete Anchor



**RM 78-3, Attachment 2:
Analysis of Data Results
Final Status Survey 09C,1
Screenhouse Excavation Area
11-02-2004**

Sample Number	Cs-137 (pCi/g)	Co-60 (pCi/g)	Weighted Sum	*Weighted Sum <DCGLw?	DCGL-W. Sum	Sign
1	0.0024	-0.0024	-0.001	yes	0.999	+1
2	0.0217	0.0218	0.009	yes	0.991	+1
3	0.0130	0.0257	0.009	yes	0.991	+1
4	0.0071	0.0263	0.009	yes	0.991	+1
5	0.0179	-0.0142	-0.003	yes	0.997	+1
6	0.0094	0.0088	0.004	yes	0.996	+1
7	0.0194	0.0159	0.007	yes	0.993	+1
8	0.0126	0.0047	0.003	yes	0.997	+1
9	0.0212	0.0055	0.004	yes	0.996	+1
10	0.0022	0.0089	0.003	yes	0.997	+1
11	0.0405	0.0401	0.016	yes	0.984	+1
12	0.1079	0.0912	0.037	yes	0.963	+1
13	0.0079	0.0077	0.003	yes	0.997	+1
14	0.02472	-0.0180	-0.004	yes	0.996	+1
15	0.0537	0.0133	0.009	yes	0.991	+1
16	0.0796	0.0041	0.008	yes	0.992	+1
17	0.1841	0.0693	0.037	yes	0.963	+1
18	0.8692	0.0235	0.080	yes	0.920	+1
19	0.0652	-0.0114	0.002	yes	0.998	+1
20	1.074	0.003	0.091	yes	0.909	+1

St. Deviation (SOR): 0.020
Mean (SOR): 0.013
Median (SOR): 0.007

Number of Positive Differences (S+): n/a

Critical Value, k , Table I.3 of Marssim: n/a

S+ > than k ? : n/a

Survey Unit Pass or Fail: *Pass

Note: Forced-Count values are used for samples with activity levels below the MDA.

* If all measurement data are less than the DCGL_w, then the Sign Test is not required.

RM-79-1
FSS QUALITY CONTROL EVALUATION RESULTS

FSS Package # 09C₁1 QC Package # 09C₁1

QC Measurement Type	Acceptance Criteria Met*?	Reference
<u>x</u> 1. Replicate Scan	<u>Yes</u> / No	Step 5.1.3
2. Sample Recounts		Step 5.1.4.1
<u>x</u> a. In-house	<u>Yes</u> / No	
_____ b. Third party	Yes / No	
3. Split Samples		Step 5.1.4.2
<u>x</u> c. In-house	<u>Yes</u> / No	
_____ d. Third party	Yes / No	

*NOTE: If Acceptance Criteria is not met, completion of Attachment RM-79-2, FSS Quality Control Investigation Results, is required.

Comments:

1. Replicate Verification Scan Information is documented in the Implementation Section of this report, FSS Mobile Scan Map.

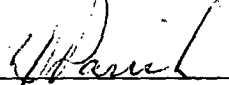
2a. In-House Sample Recount Worksheet is attached.

3c. Split Sample Verification Worksheet is attached.

Reviews:


Evaluator

11-04-04
Date


Technical Review

11/10/04
Date

**FSS QA Verification Worksheet
In-House Sample Recounts**

Date: 11/2/2004

QA Package: 09C.1, Screenhouse Area Excavation

Type of QA: Sample Recounts

Lab: In House

**Table 1:
NRC 84750 Criteria**

Resolution	Ratio
<4	N/A
4-7	0.5-2.0
8-15	0.8-1.66
16-50	0.75-1.33
51-200	0.8-1.25
>200	0.85-1.18

		A	B	C	D	E	F	G			
Sample No.	Plant Nuclide	BRP Result Below MDA	BRP Results (pCi/g)	BRP 1-sigma Error (pCi/g)	BRP Resolution (pCi/g) A/B	Recount Result Below MDA	Recount Results (pCi/g)	Ratio A/D	Resolution (Compare C w/ Table 1)	Ratio (Table 1)	*Results in Agreement (Compare E with G)
8	Co-60	<	0.0047	0.0147	0.32	<	-0.0059	0.79	<4	n/a	YES
8	Cs-137	<	0.0126	0.0107	1.18	<	-0.0101	1.25	<4	n/a	YES
15	Co-60	<	0.0133	0.0131	1.01	<	0.0166	0.80	<4	n/a	YES
15	Cs-137		0.0537	0.0172	3.13	<	0.0438	1.23	<4	n/a	YES

< indicates results less than the MDA.

*Note: All analyses comparisons not in agreement must be investigated per RM-79.

FSS QA Verification Worksheet In-House Split Samples

Date: 11/2/2004

QA Package: 09C.1 Screenhouse Area Excavation

Soil Sample Type: Split Samples

Lab: In-House

Table 1:

NRC 84750 Criteria

Resolution	Ratio
<4	N/A
4-7	0.5-2.0
8-15	0.6-1.66
16-50	0.75-1.33
51-200	0.8-1.25
>200	0.85-1.18

[illegible]

< indicates results less than the MDA.

*Note: All analyses comparisons not in agreement must be investigated per RM-79.

**Tritium in Soil
Analysis of Data Results
Final Status Survey 09C₁1
Screenhouse Excavation Area
11-02-2004**

Sample Number	Tritium in Soil (pCi/g)
4	0.05
7	0.48
8	0.12
15	0.01

**Mean: 0.17
Median: 0.09
St. Dev: 0.21**

Note: DCGL for Tritium is 327 pCi/g.
Sample data results are a fraction of the DCGL.

GENERAL ENGINEERING LABORATORIES, LLC

2040 Savage Road Charleston SC 29407 - (843) 556-8171 - www.gel.com

10 CFR Part 50/61 Certificate of Analysis

GEL Sample ID: 125136004
Client Sample ID: #4 FSS 09C1-1
Matrix: Soil
Amount of Sample Received:

Client: Big Rock Nuclear Facility
Collect Date: November 02, 2004
Receive Date: November 08, 2004
Report Date: November 18, 2004

Analyte	Aliquot ()	Run Date	Activity ²	Uncertainty	MDA ¹	RL	Units	Qualifier
H-3		11/14/04	6.20E+02	1.65E+02	2.48E+02	5.00E+02	pCi/L	3
Moisture		11/10/04	7.45E+00				percent	

Note(s): 1. Calculated MDAs are a-posteriori values.

2. Activity concentration net +/- 2 sigma overall on reference date.

3. Results are statistically positive at the 99.9% confidence level (activity is greater than three times the uncertainty)

U Indicates the target analyte was analyzed for but not detected above the detection limit.

GENERAL ENGINEERING LABORATORIES, LLC

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10 CFR Part 50/61 Certificate of Analysis

GEL Sample ID: 125136005
Client Sample ID: #7 FSS 09C1-1
Matrix: Soil
Amount of Sample Received:

Client: Big Rock Nuclear Facility
Collect Date: November 02, 2004
Receive Date: November 08, 2004
Report Date: November 18, 2004

Analyte	Aliquot ()	Run Date	Activity ²	Uncertainty	MDA ¹	RL	Units	Qualifier
H-3		11/14/04	2.00E+03	2.24E+02	2.79E+02	5.00E+02	pCi/L	3
Moisture		11/10/04	2.84E+01				percent	

Note(s): 1. Calculated MDAs are a-posteriori values.
2. Activity concentration net +/- 2 sigma overall on reference date.
3. Results are statistically positive at the 99.9% confidence level (activity is greater than three times the uncertainty)
U Indicates the target analyte was analyzed for but not detected above the detection limit.

GENERAL ENGINEERING LABORATORIES, LLC

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10 CFR Part 50/61 Certificate of Analysis

GEL Sample ID: 125136006
Client Sample ID: #8 FSS 09C1-1
Matrix: Soil
Amount of Sample Received:

Client: Big Rock Nuclear Facility
Collect Date: November 02, 2004
Receive Date: November 08, 2004
Report Date: November 18, 2004

Analyte	Aliquot ()	Run Date	Activity ²	Uncertainty	MDA ¹	RL	Units	Qualifier
H-3		11/14/04	8.44E+02	1.86E+02	2.72E+02	5.00E+02	pCi/L	3
Moisture		11/10/04	2.18E+01				percent	

Note(s): 1. Calculated MDAs are a-posteriori values.

2. Activity concentration net +/- 2 sigma overall on reference date.

3. Results are statistically positive at the 99.9% confidence level (activity is greater than three times the uncertainty)

U Indicates the target analyte was analyzed for but not detected above the detection limit.

GENERAL ENGINEERING LABORATORIES, LLC

2040 Savage Road Charleston SC 29407 - (843) 556-8171 - www.gel.com

10 CFR Part 50/61 Certificate of Analysis

GEL Sample ID: 125136007

Client: Big Rock Nuclear Facility

Client Sample ID: #15 FSS 09C1-1

Collect Date: November 02, 2004

Matrix: Soil

Receive Date: November 08, 2004

Amount of Sample Received:

Report Date: November 18, 2004

Analyte	Aliquot 0	Run Date	Activity ²	Uncertainty	MDA ¹	RL	Units	Qualifier
H-3		11/14/04	2.14E+02	1.61E+02	2.66E+02	5.00E+02	pCi/L	U
Moisture		11/10/04	4.83E+00				percent	

Note(s): 1. Calculated MDAs are a-posteriori values.

2. Activity concentration net +/- 2 sigma overall on reference date.

3. Results are statistically positive at the 99.9% confidence level (activity is greater than three times the uncertainty)

U Indicates the target analyte was analyzed for but not detected above the detection limit.

Filename : H3VAC.WAT

File type : Excel

Version # : 13

Batch : 379714

Analyst : ATH1

Date : 11/11/04

BKG Count time : 60 min

TRITIUM WATER/SOIL

Using the Vacuum Distillation Rig

Procedure Code : LSC_VH3S

Parmname : Tritium

Batch Counted on : LSCBLUE

Sample ID

Sample Dup

Run Date

1200736661

125136006

11/14/04 11:30

1200736662

125136006

11/14/04 12:33

1200736663

11/14/04 13:35

1200736664

11/14/04 14:38

BRP SAMPLE ID	Sample ID	Sample Volume mL	Position#	Count Time min	Raw CPM	Net Sample CPM	Counting Efficiency %	Bkg CPM	Time Counted
	125136001	10.00	34-2	60	17.30	11.70	25.70	5.60	11/14/04 3:10
	125136002	10.00	34-3	60	15.70	10.10	24.01	5.60	11/14/04 4:12
	125136003	10.00	34-4	60	12.40	6.80	24.60	5.60	11/14/04 5:15
QC1 #4	125136004	10.00	34-5	60	9.27	3.67	26.65	5.60	11/14/04 6:17
QC1 #7	125136005	10.00	34-6	60	16.15	10.55	23.73	5.60	11/14/04 7:20
QC1 #8	125136006	10.00	34-7	60	10.15	4.55	24.30	5.60	11/14/04 8:23
QC1 #15	125136007	10.00	34-8	60	6.78	1.18	24.89	5.60	11/14/04 9:25
	1200736660	10.00	34-9	60	8.18	2.58	25.02	5.60	11/14/04 10:28
	1200736661	10.00	34-10	60	9.97	4.37	24.77	5.60	11/14/04 11:30
	1200736662	10.00	34-11	60	90.15	84.55	23.86	5.60	11/14/04 12:33
	1200736663	10.00	34-12	60	38.33	32.73	24.49	5.60	11/14/04 13:35
	1200736664	10.00	7-1	60	43.13	37.53	26.51	5.60	11/14/04 14:38

General Engineering Laboratories, LLC

BRP Sample ID	Tritium MDA pCi/mL	Tritium RESULT pCi/mL	Tritium ERROR pCi/mL	Tritium MDA pCi/G	Tritium RESULT pCi/G	Tritium ERROR pCi/G	MDA Met?	Error Met?	Total Moisture	Initial Wt. grams
09C11 #1	0.2575	2.05	0.2123	0.0184	0.15	0.0152	Yes	Yes	70.02	979.8
09C11 #2	0.2755	1.89	0.2191	0.0182	0.13	0.0145	Yes	Yes	62.54	944.76
09C11 #3	0.2689	1.25	0.1966	0.0198	0.09	0.0145	Yes	Yes	68.82	935.41
09C11 #4	0.2483	0.62	0.1649	0.0193	0.05	0.0128	Yes	Yes	66.39	852.62
09C11 #7	0.2788	2.00	0.2240	0.0663	0.48	0.0532	Yes	Yes	248.4	1045.4
09C11 #8	0.2723	0.84	0.1862	0.0397	0.12	0.0271	Yes	Yes	144.47	991.42
09C11 #15	0.2658	0.21	0.1611	0.0170	0.01	0.0103	Yes	Yes	68.52	1070
09C11 #16	0.2644	0.46	0.1691	0.2644	0.46	0.1691	Yes	Yes	20	20
09C11 #17	0.2671	0.79	0.1815	0.0423	0.13	0.0288	Yes	Yes	144.47	911.42
09C11 #18	0.2773	15.96	0.4674	0.0440	2.53	0.0741	Yes	Yes	144.47	911.42
09C11 #19	0.2702	6.02	0.3085	0.2702	6.02	0.3085	Yes	Yes	20	20
09C11 #20	0.2495	6.38	0.3001	0.2495	6.38	0.3001	Yes	Yes	20	20

Page 2

**Michigan Department of Environmental Quality
Radiological Protection Program**

Dave,

Thanks for the sample results. Attached is our analysis of samples 4 and 7.

T.R.

"Theodore Wentworth" <wentwort@michigan.gov>

12/02/04 09:11 AM

Screenhouse Excavation Area Final Status Survey

Sample Collection Date: 11/2/04

Sample Name	Sample Type	Co-60 (pCi/g)	Cs-137 (pCi/g)	Sample Mass (g)
#4	Soil	LT .07	LT 0.06	2290.0
#7	Soil	LT 0.1	LT 0.07	952.1

* LT indicates less than the minimum detectable activity.

0000 4929 0997

FSS QA Verification Worksheet **3rd Party Split Samples**

Date: 11/2/2004

QA Package: 09C.1 Screenhouse Area Excavation

Soil Sample Type: Split Samples

Lab: MDEQ

Table 1:
NRC 84750 Criteria

Resolution	Ratio
<4	N/A
4-7	0.5-2.0
8-15	0.6-1.66
16-50	0.75-1.33
51-200	0.8-1.25
>200	0.85-1.18

		A		B		C		D		E		F		G	
Sample No.	Plant Nuclide	BRP Result Below MDA	BRP Results (pCi/g)	BRP 1-sigma Error (pCi/g)	BRP Resolution (pCi/g) A/B	Split Sample Result Below MDA	Split Sample Results (pCi/g)	Ratio A/D	Resolution (Compare C w/ Table 1)	Ratio (Table 1)	*Results in Agreement (Compare E with G)				
4	Co-60	<	0.0263	0.0169	1.56	<	0.0700	0.38	<4	n/a	YES				
4	Cs-137	<	0.0071	0.0097	0.73	<	0.0600	0.12	<4	n/a	YES				
7	Co-60	<	0.0159	0.0157	1.01	<	0.1000	0.16	<4	n/a	YES				
7	Cs-137	<	0.0194	0.0127	1.52	<	0.0700	0.28	<4	n/a	YES				

< Indicates results less than the MDA.

*Note: All analyses comparisons not in agreement must be investigated per RM-79.

January 18, 2005

Mr. Kurt M. Haas
General Manager
Big Rock Point Nuclear Plant
Consumers Energy Company
10269 U.S. 31 North
Charlevoix, MI 49720

SUBJECT: BIG ROCK POINT INSPECTION REPORT 05000155/2004-003(DNMS) AND
NOTICE OF VIOLATION

Dear Mr. Haas:

On January 7, 2005, the NRC completed inspection activities at the Big Rock Point Nuclear Plant. The purpose of the inspection was to determine whether decommissioning activities were conducted safely and in accordance with NRC requirements. Specifically, during onsite inspections from October 12 through 15, 2004, November 1 through 5, 2004, and December 13 through 16, 2004, the inspectors evaluated decommissioning support activities, final status surveys, and radiological safety. At the conclusion of the onsite inspections on October 15, November 5, and December 16, 2004, the inspectors discussed the inspection findings with you and members of your staff. On January 7, 2005, the inspectors completed an in-office review of laboratory analysis results for split soil samples that were collected during the November 1 through 15 inspection. The inspectors conducted a telephone exit interview with Mr. Ken Pallagi, Radiation Protection & Environmental Services Manager, on January 7, 2005, to discuss the results of the in-office review of the laboratory results.

The onsite inspections consisted of an examination of decommissioning activities at the Big Rock Point Nuclear Plant as they relate to safety and compliance with the Commission's rules and regulations. Areas examined during the inspections are identified in the enclosed report. Within these areas, the inspections consisted of a selective examination of procedures and representative records, observations of activities in progress, and interviews with personnel.

Based on the results of these inspections, the NRC has determined that one Severity Level IV violation of NRC requirements occurred. The violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the Enforcement Policy. The current Enforcement Policy is included on the NRC web site at www.nrc.gov; select **What We Do, Enforcement**, then **Enforcement Policy**. The NCV is described in the subject inspection report. If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with a copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's

document system (ADAMS). The NRC's document system is accessible from the NRC Web site at <http://www.nrc.gov/reading-rm/adams.html> (the Public Electronic Reading Room).

Please note that on October 25, 2004, the NRC terminated public access to ADAMS and initiated an additional security review of publicly available documents to ensure that potentially sensitive information is removed from the ADAMS database accessible through the NRC's web site. Interested members of the public may obtain copies of the referenced documents for review and/or copying by contacting the Public Document Room pending resumption of public access to ADAMS. The NRC Public Documents Room is located at NRC Headquarters in Rockville, MD, and can be contacted at (800) 397-4209.

We will gladly discuss any questions you may have regarding this inspection.

Sincerely,

/RA/

Jamnes L. Cameron, Chief
Decommissioning Branch

Docket No. 05000155
License No. DPR-6

Enclosure: Inspection Report 05000155/2004-003(DNMS)

cc w/encl: R. A. Fenech, Senior Vice President, Nuclear, Fossil, and Hydro Operations
John King, Michigan Public Service Commission
L. Shekter Smith, Michigan Department of Environmental Quality
Chief, Nuclear Facilities Unit, Michigan Department of Environmental Quality
Department of Attorney General (MI)
Emergency Management Division, Michigan Department of State Police

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No. 050-00155

License No. DPR-6

Report No. 05000155/2004-003(DNMS)

Licensee: Consumers Energy Company

Facility: Big Rock Point Restoration Project

Location: 10269 U.S. 31 North
Charlevoix, MI 49720

Dates: October 12 through 15, 2004 (Onsite inspection)
November 1 through 5, 2004 (Onsite inspection)
December 13 through 16, 2004 (Onsite inspection)
January 7, 2005 (In-office review)

Inspectors: William G. Snell, Senior Health Physicist (Inspector)
Christopher R. Martin, Reactor Inspector (Decommissioning)
Sam Mulay, Materials Inspector

Approved by: Jamnes Cameron, Chief
Decommissioning Branch
Division of Nuclear Materials Safety

EXECUTIVE SUMMARY

Consumers Energy Company Big Rock Point Restoration Project NRC Inspection Report 05000155/2004-003(DNMS)

These routine decommissioning inspections involved a review of the Consumers Energy Company's and its contractors' current performance related to decommissioning support activities and radiological safety. During the inspection period, major activities reviewed included facility demolition and decontamination, and radiological and environmental surveys.

Decommissioning Support Activities

- The inspectors concluded that the licensee and its contractors conducted decommissioning activities in accordance with appropriate regulatory requirements and in a safe manner. Management oversight of decommissioning activities was commensurate with the scope and complexity of the activities observed. (Section 1.0)

Radiation Protection Program

- The inspectors identified one Non-Cited Violation of 10 CFR 20.1801 for failure to secure from unauthorized removal or limit access to licensed material (activated concrete) located in the radioactive waste building compound, which was a controlled area, nor did the licensee control and maintain constant surveillance of this licensed material. This finding is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. (Section 2.0)

Final Status Survey

- The inspectors determined that the licensee's final status survey of the screen house was consistent with procedural requirements and that the procedures used were consistent with the Final Survey Plan found in Chapter 5 of the License Termination Plan. The inspectors concluded that the licensee implemented the survey program in accordance with approved decommissioning procedures. (Section 3.0)
-

Report Details¹

1.0 Decommissioning Support Activities (71801)

a. Inspection Scope

The inspectors evaluated decommissioning activities to verify that the licensee and its contracted workforce were conducting work in accordance with licensed requirements. In addition, the inspectors evaluated the licensee's management and oversight of decommissioning activities.

b. Observations and Findings

The inspectors completed numerous site tours to observe licensee staff conduct decommissioning activities such as the removal of surface contamination from segments of the dismantled stack, decontamination and surveys of equipment and building surfaces, final status surveys of the pump house area, radiation protection work practices, movement of heavy loads, activated concrete removal, and demolition of the Turbine Building.

The inspectors noted that the licensee and its contractors were knowledgeable of their work assignments and attentive to their individual tasks. The inspectors verified that the licensee and its contractors' staff were cognizant of the radiological conditions in their work area and aware of actions that could cause the radiation or contamination levels to change. The inspectors observed that the licensee and its contractors communicated effectively, demonstrated appropriate concern for industrial and radiological safety, conducted work in accordance with procedural requirements, and employed good work practices. The inspectors discussed work activities with management, health physics (HP) technicians and contractors during the tours to verify that they understood the radiological issues pertinent to their assigned activities.

The inspectors observed the material condition of facilities and equipment and determined it to be commensurate with the current decommissioning activities. The inspectors noted that general housekeeping was adequate.

c. Conclusion

The inspectors concluded that the licensee and its contractors conducted decommissioning activities in accordance with appropriate regulatory requirements and in a safe manner. Management oversight of decommissioning activities was commensurate with the scope and complexity of the activities observed.

¹A list of acronyms used in the report is included at the end of the Report Details.

2.0 Radiation Protection Program (83750)

a. Inspection Scope

The inspectors reviewed selected radiation protection procedures, observed licensee and contractor staff implement the program requirements, and interviewed licensee and contractor staff, to verify that the program was appropriate for the radiological hazards associated with current decommissioning activities.

b. Observations and Findings

The inspectors observed the licensee conduct daily briefings prior to the beginning of authorized work activities. The inspectors noted that the briefings, as a minimum, consisted of a discussion of the current industrial and radiological conditions at the work sites.

The inspectors observed the licensee and its contractors perform decommissioning activities within the turbine building, the containment sphere, and the discharge canal. The workers performed the decommissioning activities in accordance with approved radiation work permits (RWPs), and/or procedures.

The inspectors noted that the contractors had begun the removal process of the remaining activated concrete from the containment sphere. The contractors' plan required the staff to soften the concrete by mechanical means prior to removal. The contractor discovered significantly more reinforcing bar than expected, and as a result, the work became more labor intensive than initially planned. The HP staff recognized the potential for increased dose and closely monitored the activity to ensure that worker doses were maintained as low as is reasonably achievable (ALARA).

The inspectors identified that several activated concrete blocks, recently removed from the containment sphere (reactor containment), were stored in shipping containers within the licensee's radioactive waste processing area, which is outside the controlled area. The radioactive waste processing area was neither adequately secured nor under surveillance by licensee staff or its contractors. Radiation levels, resulting from the radioactive material contained within the shipping containers, ranged from 60 to 100 millirem per hour (mrem/hr) on contact with the shipping container.

The inspector had previously discussed the security of licensed materials with the Radiation Protection Manager (RPM). Specifically, in April 2004 the inspectors expressed concern that if the activated concrete blocks were not adequately secured from unauthorized access a member of the public could exceed the public dose limit of 100 mrem in a short period of time. The RPM acknowledged the inspectors' concern. The RPM subsequently engaged the work planning staff to ensure that the activated blocks would remain in the controlled area when removed from the containment sphere. However, scheduling conflicts caused the licensee's contractors to store the blocks in the radioactive waste building compound. The radioactive waste building compound is a controlled area and did not have a contiguous fence to prevent unauthorized access. Specifically, a 20 foot section of fencing was previously removed and a rope barrier was utilized to prevent access. A member of the public could have exited the state road adjacent to the site on foot and followed a cleared path (abandoned power line right of way) for approximately 300 yards to the radioactive waste processing area.

Title 10 CFR 20.1801 requires that the licensee secure from unauthorized removal or access licensed materials that are stored in controlled or unrestricted areas. Title 10 CFR 20.1802 requires that the licensee control and maintain constant surveillance of licensed material that is in a controlled or unrestricted area and that is not in storage. As defined in 10 CFR 20.1003, *controlled area* means an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason; and *unrestricted area* means an area, access to which is neither limited nor controlled by the licensee.

Contrary to the above, between September 24 and November 3, 2004, the licensee did not secure from unauthorized removal or limit access to licensed material (activated concrete) located in the radioactive waste building compound, which was a controlled area, nor did the licensee control and maintain constant surveillance of this licensed material. The licensee entered this issue in its corrective action program (CAP) as No. C-BRP-04-0209, entitled "Issue Identified Related to 10 CFR 20.1801 and 1802 Storage of Licensed Material." This finding is considered to be a Non-Cited Violation of 10 CFR 20.1801 (Violation 05000155/2004-003-01) consistent with Section VI.A of the NRC Enforcement Policy.

The inspectors discussed this issue with licensee management, and the Site Director immediately had his staff erect a contiguous fence around the radioactive waste building compound to prevent unauthorized access. In addition, the site Director directed his management team to determine if additional vulnerabilities to security of licensed materials existed. The management team determined that the controlled area physical barrier (fence line) required additional attention to ensure compliance with regulatory requirements. The staff subsequently added to or modified the existing controlled area fence line.

The inspectors reviewed the licensee's corrective actions and determined that the corrective actions were appropriate to address all the immediate and potential generic aspects of the violation.

c. Conclusion

The inspectors identified one Non-Cited Violation of 10 CFR 20.1801 for failure to ~~secure from unauthorized removal or limit access to licensed material~~ (activated concrete) located in the radioactive waste building compound, which was a controlled area, nor did the licensee control and maintain constant surveillance of this licensed material. This finding is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy.

3.0 Final Status Survey (83801)

a. Inspection Scope

The inspectors evaluated final status survey documentation to verify that areas had been decontaminated to radiological levels consistent with procedural requirements. In addition, the inspectors performed independent confirmatory surveys during the licensee's final status survey of the excavated area resulting from the demolition of the screen house.

b. Observations and Findings

The licensee's screen house final status survey encompassed approximately 1820 square meters at the northeast section of the industrial area. The survey area was an open excavation approximately eight meters below grade that resulted from the demolition and removal of the screen house subsurface structures and components (e.g., screen house intake and discharge foundations, septic holding tanks and lift pump station, emergency diesel generator fuel oil storage tanks, and support piping and conduit). The intake pipe that provided Lake Michigan water to the screen house for distribution to the industrial area was located inside the north boundary of the survey area. The licensee and its contractors plugged the pipe to prevent lake water from entering the excavation area. The pipe and a portion of the screen house wall (serving as a concrete anchor for the end of the pipe) were abandoned in place in accordance with the License Termination Plan (LTP). No other material or components from the facility remained in the survey area.

The licensee staff conducted the screen house final status survey described in its survey plan. The inspectors performed side-by-side independent confirmatory surveys with the licensee staff and a representative from the Michigan Department of Environmental Quality. The inspectors performed the independent confirmatory surveys using radiation detection equipment (Ludlum Model 2241-2 with sodium-iodide probe) which was comparable to the licensee's radiation detection equipment (Ludlum Model 2350-1 with sodium-iodide probe). The NRC and licensee radiation detection instruments were verified as operable and met the annual calibration periodicity.

The inspectors initiated the confirmatory survey by performing a background check of the instruments concurrent with the licensee staff. The inspectors noted that both the NRC and licensee radiation detection instrumentation indicated similar background radiation levels of 5,000 to 7,000 disintegrations per minute (dpm). The inspectors then performed side-by-side scanning surveys covering approximately 75 percent of the screen house excavation site. The remaining area was moist and could not be surveyed at that time; however, the area was previously characterized by the licensee and found to be below the applicable release criteria.

The licensee staff determined that cobalt-60 was the most limiting radioisotope expected to be present and assumed that all residual radioactivity was cobalt-60. The licensee staff established an instrument response value of 1818 counts per minute (cpm) above background radiation levels based on the cobalt-60 Derived Concentration Guideline Level (DCGL) as the scanning investigation level for this area. The inspectors observed that the licensee staff denoted survey points found to be greater than the established instrument response value for the collection of "judgmental samples" and subjected them to further radiological analysis in accordance with the previously established investigation levels.

The inspectors noted that none of the licensee's judgmental sample results exceeded the DCGL for cobalt-60.

The site specific DCGL for the radioisotopes expected to be present were: 11.93 picocuries per gram (pCi/g) for cesium-137 and 3.21 pCi/g for cobalt-60. Based on a statistical evaluation of the survey unit, the licensee staff collected 20 random soil samples to demonstrate the discharge canal survey unit was adequately remediated.

The licensee staff also selected 5 percent of these samples (i.e., one sample) for quality assurance/quality control (QA/QC) in accordance with approved procedures.

The inspectors collected split soil samples for two of the samples. The inspectors sent the samples to the Oak Ridge Institute for Science and Education (ORISE) for analysis. The analytical results for the NRC and licensee split soil samples collected were below the applicable DCGL levels and are documented in Table 1. The licensee's results for samples 1 through 6 and 9 through 20 were also below the applicable DCGL levels.

Table 1 - Final Status Survey Soil Sample Results¹

Sample No.	Licensee cobalt-60	Licensee cesium-137	NRC cobalt-60	NRC cesium-137
7	0.05 ²	0.07 ²	0.03 ²	0.03 ²
8	0.05 ²	0.06 ²	0.03 ²	0.03 ²

¹ sample results are in picocuries per gram soil (pCi/g)

² minimum detectable concentration

The inspectors verified by observation that the survey was conducted in accordance with the approved procedures found in the licensee's "Final Status Survey Program." These procedures were: Procedure No. RM-77, entitled "Final Status Survey Implementation;" Procedure No. RM-76, entitled "Final Status Survey Design;" and Procedure No. RM-78, entitled "Final Status Survey Assessment."

The inspectors reviewed the following information, survey, and verification work packages: 2004-0103, entitled "Turbine Building Instrument Shop/Electrical Shop Rooms 122/123A;" 2004-0098, entitled "Turbine Building Lay-Down/Condensate Pump Rooms 124/125;" 2004-0099, entitled "Turbine Building foundations, Rooms 117/118/119;" 2003-0089, entitled "Turbine Building Air Ejector Room 115;" 2004-0004, entitled "Pipe Tunnel Roof;" 2004-0016, entitled "Screen House Discharge Canal/Apron/Canal Walls Room 505;" 2003-0088, entitled "Turbine Building Pipe Tunnel Room 114;" 2003-0012, entitled "Turbine Building Condenser Area Rooms 117/118/119;" and 2004-0040, entitled "East Office Building Annex, Septic Tanks and Associated Manholes."

c. Conclusion

The inspectors determined that the licensee's final status survey of the screen house was consistent with procedural requirements and that the procedures used were consistent with the Final Survey Plan found in Chapter 5 of the License Termination Plan. The inspectors concluded that the licensee implemented the survey program in accordance with approved radiological criteria for release.

4.0 Exit Meeting Summary

The inspectors presented preliminary inspection findings to members of the licensee management team at the conclusion of onsite inspection activities on October 15, November 5, and December 16, 2004. On January 7, 2005, the inspectors conducted a telephone exit interview with the Radiation Protection & Environmental Services Manager to discuss the results of the in-office review of the laboratory results. The licensee acknowledged the findings presented. The licensee did not identify any documents or processes reviewed by the inspector as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

Consumers Energy Company

- * Kurt Haas, Site General Manager
- * Ken Pallagi, Radiation Protection & Environmental Services Manager
- * Greg Withrow, Engineering, Operations & Licensing Manager
- * William Trubilowicz, Cost, Scheduling and Purchase Manager

State of Michigan

T.R. Wentworth, Michigan Department of Environmental Quality

* Indicates those individuals present at the preliminary and/or final exit meetings.

INSPECTION PROCEDURES USED

IP 71801	Decommissioning Performance and Status Review
IP 83750	Occupational Radiation Exposure
IP 83801	Inspection of Final Surveys at Permanently Shutdown Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>	<u>Type</u>	<u>Summary</u>
05000155/2004-003-01	NCV	Failure to secure licensed material.

Closed

05000155/2004-003-01	NCV	Failure to secure licensed material.
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Discussed

None

PARTIAL LIST OF DOCUMENTS REVIEWED

Licensee documents reviewed and utilized during the course of this inspection are specifically identified in the "Report Details" above.

LIST OF ACRONYMS USED

ADAMS	AgencyWide Documents Access and Management System
ALARA	As Low As Reasonably Achievable
BRP	Big Rock Point
CAP	Corrective Action Program
CFR	Code of Federal Regulations
cpm	counts per minute
DCGL	Derived Concentration Guideline Level
DNMS	Division of Nuclear Material Safety
dpm	disintegrations per minute
HP	Health Physics
LTP	License Termination Plan
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
RWP	Radiation Work Permit