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June 20, 2018

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SUBJECT: CONTRACT NO. DE-SC0014664
INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE CIRCULATING WATER DISCHARGE INTERIOR PIPING AT THE LA CROSSE BOILING WATER REACTOR, GENOA, WISCONSIN
RFTA No. 18-003; DCN 5299-SR-03-0

Dear Ms. Vaaler:

The Oak Ridge Institute for Science and Education (ORISE) is pleased to provide the subject report detailing the independent confirmatory surveys performed of the Circulating Water Discharge interior piping at the La Crosse Boiling Water Reactor. NRC comments have been incorporated into this final version.

Please feel free to contact me at 865.574.6273 or Erika Bailey at 865.576.6659 if you have any comments or concerns.

Sincerely,

Nick A. Altic, CHP
Health Physicist/Project Manager
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NAA:lw

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N. A. Altic, CHP
and
S. T. Pittman, PhD
ORISE

FINAL REPORT

Prepared for the
U.S. Nuclear Regulatory Commission

June 2018

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This document was prepared for U.S. Nuclear Regulatory Commission by the Oak Ridge Institute for Science and Education (ORISE) through an interagency agreement with the U.S. Department of Energy (DOE) (NRC FIN No. F-1244). ORISE is managed by Oak Ridge Associated Universities under DOE contract number DE-SC0014664.

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FINAL REPORT

JUNE 2018

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ACRONYMS

AA	alternate action
BWR	boiling water reactor
Co-60	cobalt-60
cpm	counts per minute
Cs-137	cesium-137
CWD	Circulating Water Discharge
DCGL	derived concentration guideline level
$DCGL_{BC}$	Base Case DCGL
$DCGL_{Ops}$	Operational DCGL
DPC	Dairyland Power Cooperative
DQO	data quality objective
Eu-152	europium-152
Eu-154	europium-154
FESW	fuel element storage well
FRS	final radiation survey
FSS	final status survey
ISFSI	Independent Spent Fuel Storage Installation
LACBWR	La Crosse Boiling Water Reactor
LSE	LACBWR site enclosure
LTP	license termination plan
m	meter
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
mrem/yr	millirem per year
NaI	sodium iodide
NRC	U.S. Nuclear Regulatory Commission
ORAU	Oak Ridge Associated Universities

ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
PSP	project-specific plan
PSQ	principal study question
Q	quantile
ROC	radionuclide of concern
STS	source term survey
SU	survey unit
VSP	Visual Sample Plan

INDEPENDENT CONFIRMATORY SUMMARY AND RESULTS FOR THE CIRCULATING WATER DISCHARGE INTERIOR PIPING AT THE LA CROSSE BOILING WATER REACTOR, GENOA, WISCONSIN

EXECUTIVE SUMMARY

At the request of the U.S. Nuclear Regulatory Commission, the Oak Ridge Institute for Science and Education performed an independent confirmatory survey of the interior of the circulating water discharge (CWD) piping during the period of April 24–26, 2018, at the La Crosse Boiling Water Reactor. The confirmatory survey consisted of gamma scans and surface activity measurements for gamma and beta radiation to independently assess the final radiological status of the CWD piping interior relative to the release criterion. None of the collected surface activity measurements—based on either gamma or beta radiation—were above the operational gross activity derived concentration guideline level (which corresponds to a dose of approximately 5.3 mrem/yr). Based on the confirmatory survey data, it is the opinion of ORISE that residual radiation levels in the CWD piping interior are less than the release criterion.

INDEPENDENT CONFIRMATORY SURVEY RESULTS AND SUMMARY FOR THE CIRCULATING WATER DISCHARGE INTERIOR PIPING AT THE LA CROSSE BOILING WATER REACTOR, GENOA, WISCONSIN

1. INTRODUCTION

The La Crosse Boiling Water Reactor (LACBWR), a 50-megawatt electric boiling water reactor (BWR) located in Genoa, Wisconsin, was originally a demonstration plant funded by the U.S. Atomic Energy Commission. The plant was later sold to Dairyland Power Cooperative (DPC) with a provisional operating license. The BWR achieved initial criticality on July 11, 1967 and operated for 19 years until being permanently shut down on April 30, 1987. After shutdown, DPC's authority to operate LACBWR under Provisional Operating License DPR-45 (issued by the U.S. Nuclear Regulatory Commission [NRC] on August 28, 1973) was amended via License Amendment 56 (August 4, 1987) to possession only authority (LS 2016).

Dismantling unused and offline systems and waste disposal operations began in 1994. The Reactor Pressure Vessel (head, internals, and 29 control rods sealed with concrete), stored waste in the Fuel Element Storage Well (FESW), and other Class B/C wastes were shipped offsite for disposal in June 2007. Other systems and components—such as spent fuel storage racks, gaseous waste disposal systems (excluding the underground gas storage tanks), condensate and feedwater system (excluding condensate storage tank and condenser), the turbine and generator, and various components located in the Turbine Building (cooling water system pumps, heat exchangers, piping, etc.)—have also been removed. In September 2012, 333 irradiated fuel assemblies from the FESW were packaged in five dry casks and transferred to the site's Independent Spent Fuel Storage Installation (ISFSI) (LS 2016). In May 2016, the NRC consented to having the possession, maintenance, and decommissioning authorities of the LACBWR site transferred from DPC to LaCrosse*Solutions,* LLC.

LACBWR has submitted a License Termination Plan (LTP) to the NRC requesting the removal of all remaining open-land and structures, except for the fenced area surrounding the ISFSI, from License DPR-45. The LACBWR Administration Building, Crib House, and Transmission Sub-Station Switch House will remain intact. All other LACBWR buildings and structures will be demolished and removed to a depth of three feet below grade—corresponding to the 636-foot

elevation—including the basements of the Reactor Building, Waste Treatment Building, Waste Gas Tank Vault, and other miscellaneous remaining basement structures.

The LTP requires a Final Radiation Survey (FRS) that will radiologically characterize the site and determine the potential for an average member of the critical group to receive a total effective dose equivalent greater than 25 millirem per year (mrem/yr). The FRS plan consists of two types of compliance surveys: a Final Status Survey (FSS) for open-land areas and buried piping based on NUREG-1575 (NRC 2000) and a Source Term Survey (STS) for the below-ground structures that will be backfilled prior to license termination.

NRC requested that the Oak Ridge Institute for Science and Education (ORISE) perform confirmatory survey activities to independently assess the final radiological condition of the Circulating Water Discharge (CWD) piping interior. This report summarizes the confirmatory survey activities associated with the CWD interior piping, at LACBWR.

2. SITE DESCRIPTION

The LACBWR site is located approximately 1.6 kilometers (1 mile) south from the village of Genoa, Wisconsin on the eastern shore of the Mississippi River. The 10 CFR Part 50 licensed site is shared with the non-nuclear Genoa-3 Fossil Station and comprises a total of 66.2 hectares (164 acres). The operational fossil plant's buildings and structures were classified as non-impacted and are not subject to the release surveys specified in the LTP (LS 2016). Figure 2.1 provides an aerial view of the licensed site, within which is located the 0.61-hectare LACBWR Site Enclosure (LSE) area, shaded red in the figure.

The CWD piping, designated as survey unit S1-011-102 CWD, is a 1.52-meter (m) diameter steel pipe with a length of approximately 128 m (420 feet). During reactor operations, it was the receiver of batch liquid discharges to the Mississippi River and is a Class 1 survey unit (LS 2018a). At the time of survey, pumps had largely removed river water from the pipe, with residual water and sediment remaining in limited areas.

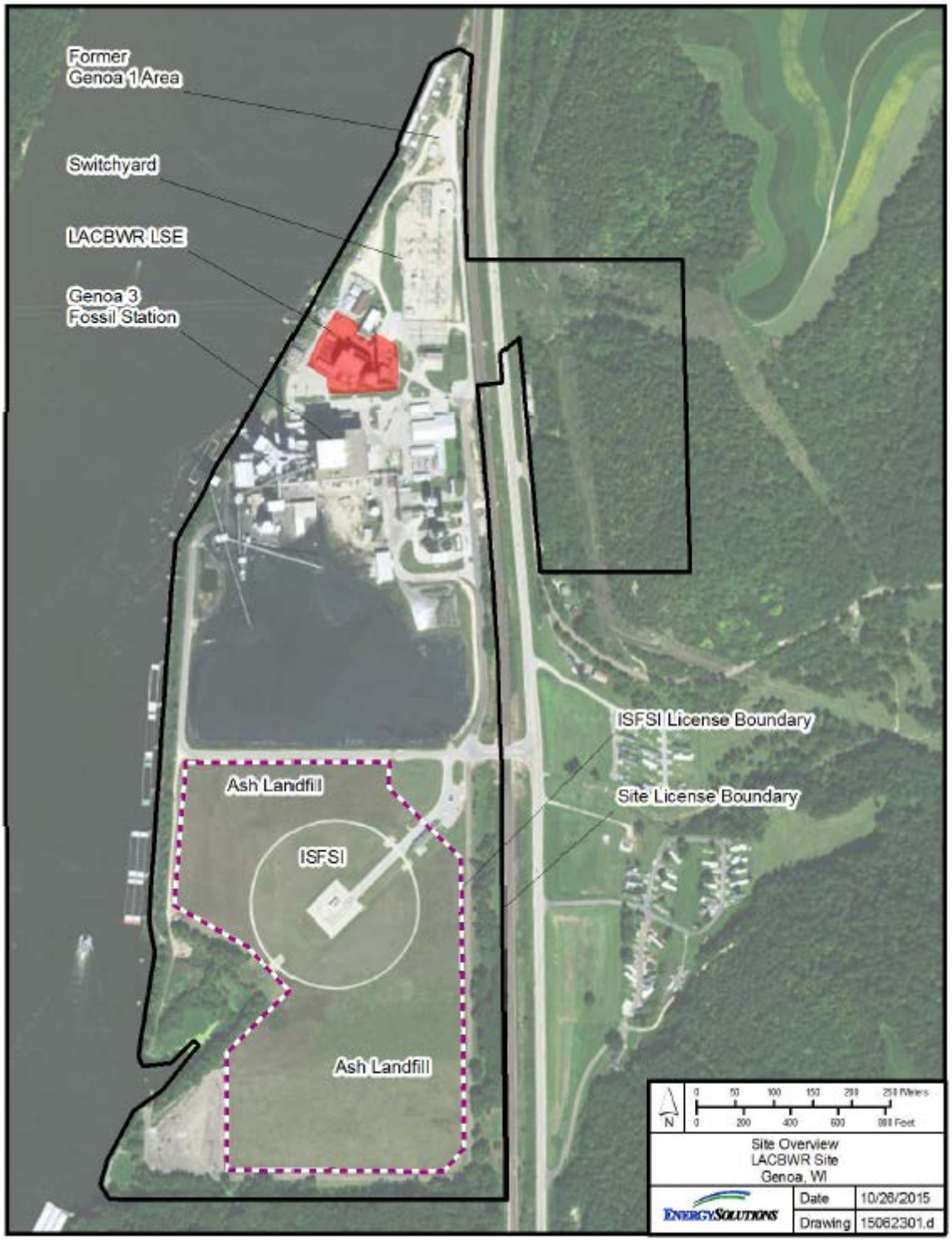

Figure 2.1. LACBWR Site Overview (LS 2016)

3. DATA QUALITY OBJECTIVES

The data quality objectives (DQOs) process was applied to the design of the confirmatory survey. The DQOs applied, and described herein, were consistent with the *Guidance on Systematic Planning Using the Data Quality Objectives Process* (EPA 2006) and provided a formalized method for planning confirmatory radiation surveys, improving survey efficiency and effectiveness, and ensuring that the type, quality, and quantity of data collected were adequate for the intended decision applications. The seven steps in the DQO process were as follows:

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

3.1 STATE THE PROBLEM

The first step in the DQO process defined the problem that necessitated the study, identified the planning team, and examined the project budget and schedule. LACBWR is in the process of dismantling remaining structures and remediating remaining lands. As part of this process, LACBWR is conducting an FSS to demonstrate compliance with the NRC's license termination criteria specified in 10CFR20.1402. To this end, the NRC requested that ORISE perform confirmatory surveys of the CWD piping interior to provide independent data for NRC's consideration in their evaluation of the FSS. Therefore, the problem statement was as follows:

> Confirmatory surveys are necessary to generate independent radiological data for NRC's consideration in the evaluation of the FSS design, implementation, and results for demonstrating compliance with the release criteria.

3.2 IDENTIFY THE DECISION

The second step in the DQO process identified the principal study questions (PSQs) and alternate actions (AAs); developed decision statements; and organized multiple decisions, as appropriate. This was done by specifying AAs that could result from a "yes" response to the PSQs and combining the PSQs and AAs into decision statements. Given that the problem statement introduced in Section 3.1 was fairly broad, multiple PSQs arose and were detailed in the confirmatory survey project-specific plan (PSP) (ORISE 2017). The PSQ, AAs, and combined decision statement applicable to the survey efforts detailed in this report are presented in Table 3.1.

Table 3.1. LACBWR CWD Piping Confirmatory Survey Decision Process

Principal Study Question	Alternative Actions
Are residual radioactivity levels in the CWD piping below the release criterion?	**Yes:** Confirmatory results indicate that residual radioactivity levels are below allowable limits—compile confirmatory survey data and present the results to the NRC for their decision making. **No:** Confirmatory survey results indicate that residual radioactivity levels exceed allowable limits—summarize unacceptable data points and provide technical comments and/or further evaluation(s) to the NRC for their decision making.
Decision Statement	
Determine if the residual radioactivity levels are below/above the allowable limits.	

3.3 IDENTIFY INPUTS TO THE DECISION

The third step in the DQO process identified both the information needed and the sources of this information; determined the basis for action levels; and identified sampling and analytical methods to meet data requirements. For this effort, information inputs included the following:

- LACBWR FSS Plan for the CWD piping (LS 2018a)
- LWCBWR derived concentration guideline levels (DCGLs); discussed in Section 3.3.1

- LACBWR Procedure No. LC-FS-PR-018, Radiation Surveys of Pipe Interiors Using Sodium/Cesium Iodide Detectors (LS 2018b)
- ORISE confirmatory survey results including: surface radiation scans and direct surface activity measurements

3.3.1 Radionuclides of Concern

The primary radionuclides of concern (ROCs) identified for LACBWR are beta-gamma emitters—fission and activation products—resulting from reactor operation. The DCGLs for the CWD piping are presented in Table 3.2.

Table 3.2. DCGLs for CWD Piping (dpm/100 cm2)

ROC	$DCGL_{BC}$	$DCGL_{Ops}$
Co-60	7.75E+04	1.63E+04
Sr-90	7.56E+05	1.59E+05
Cs-137	3.30E+05	6.94E+04
Eu-152	1.67E+05	3.51E+04
Eu-154	1.56E+05	3.27E+04

dpm/100 cm 2 = disintegrations per minute per 100 square-centimeters
$DCGL_{BC}$ = Base Case DCGL
$DCGL_{Ops}$ = Operational DCGL
Source: LS 2018a

In Table 3.2, the Base Case DCGL ($DCGL_{BC}$) is the residual radioactivity level—when considered individually—that results in a potential total effective dose equivalent of 25 mrem/yr (the release criterion) to a future receptor. To account for multiple source terms at the LACBWR site, the Base Case DCGLs are reduced to a fraction of the dose criterion; the reduced DCGL is termed the Operational DCGL ($DCGL_{Ops}$).

LACBWR has established gross gamma activity DCGLs using the surrogate approach to account for the pure beta emitter, Sr-90, and the expected activity fractions for other gamma-emitters. Gross activity DCGLs are presented in Table 3.3.

Table 3.3. Gross Activity DCGLs (dpm/100 cm$^{2)}$	
	Gross Activity Limit
$DCGL_{BC}$	2.28E+05
$DCGL_{Ops}$	4.80E+04

Source: LS 2018a

3.4 DEFINE THE STUDY BOUNDARIES

The fourth step in the DQO process defined target populations and spatial boundaries; determined the timeframe for collecting data and making decisions; addressed practical constraints; and determined the smallest subpopulations, area, volume, and time for which separate decisions must be made.

Physical boundaries of the confirmatory survey were limited to the CWD interior piping survey unit, identified as S1-011-102 CWD by the licensee. Three full days were allotted for this survey, constituting the temporal boundary of the study.

3.5 DEVELOP A DECISION RULE

The fifth step in the DQO process specified appropriate population parameters (e.g., mean, median); confirmed detection limits were below the action levels; and developed an if…then… decision rule statement.

The parameters of interest for this survey were individual surface activity measurements collected from the CWD piping survey unit. LACBWR established a measurement scheme where static gamma measurements were collected every linear foot of piping. Based on LACBWR's calibration and measurement methodology, each static surface activity measurement represented a total area of 3,315 cm^2, with a width of 32.5 cm along the central axis of the pipe. Measurements were collected with the detector approximately 14 cm from the bottom of the pipe, as this represents the highest potential for contamination. The confirmatory survey was designed such that, provided a representative number of surface activity measurements from the CWD piping were below the action level, one could conclude that a high percentage of the piping was below the action level with a specified degree of confidence. In this instance the action level was the $DCGL_{BC}$. Given the previous discussion, the decision rule was stated as follows:

If confirmatory survey measurements were below the $DCGL_{BC}$, then conclude that a high percentage of the CWD piping is below release limit; otherwise, perform further evaluation(s) and provide technical comments to the NRC.

3.6 SPECIFY LIMITS ON DECISION ERRORS

The sixth step in the DQO process specified the decision maker's limits on decision errors, which were then used to establish performance goals for the survey.

Decision errors were limited by establishing the confidence level for confirming that residual activity over a high percentage of the piping was less than the $DCGL_{BC}$. For this study, a 95% confidence level was selected. Visual Sample Plan (VSP), version 7.9, was used to determine the number of measurements necessary to conclude that 95% of the survey unit was acceptable at the 95% confidence level. Figure 3.1 depicts the VSP output. A total of 55 measurements were needed to achieve the desired confidence level.

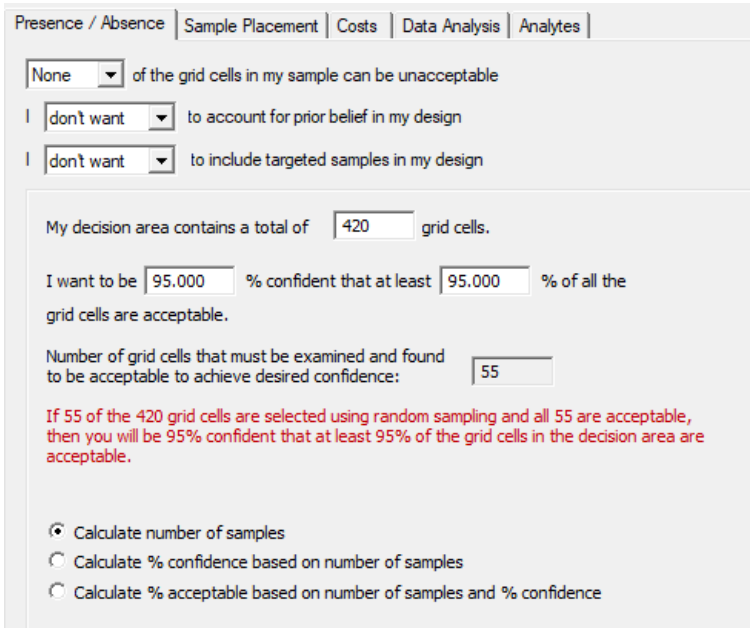

Figure 3.1. VSP Sample Size Determination

An additional level of control was to ensure that static measurement count times were such that the minimum detectable concentration (MDC) was less than the gross surface activity $DCGL_{BC}$ and that scan sensitivities were adequate for the identification of any localized, high activity hot spots.

3.7 OPTIMIZE THE DESIGN FOR OBTAINING DATA

The seventh step in the DQO process was used to review DQO outputs; develop data collection design alternatives; formulate mathematical expressions for each design; decide on the most resource-effective design of agreed alternatives; and document requisite details. The survey design was optimized by implementing the procedures outlined in Section 4.

4. PROCEDURES

The ORISE survey team performed visual inspections, measurements, and sampling activities within the CWD piping survey unit S1-011-102 CWD. Survey activities were conducted in accordance with the *ORAU Radiological and Environmental Survey Procedures Manual*, the *ORAU Environmental Services and Radiation Training Quality Program Manual*, and the approved confirmatory survey PSP Addendum (ORAU 2016a, ORAU 2016b, and ORISE 2018).

4.1 REFERENCE SYSTEM

ORISE referenced confirmatory measurement/sampling locations to the site's reference system, which was distance traveled from the south to north end of the CWD pipe.

4.2 SURFACE SCANS

LACBWR designed a detector carrier for scanning the interior of the CWD piping. The carrier is a wheeled device designed to keep a 2 × 2-inch sodium iodide (NaI) detector in a fixed geometry. ORISE used this device at the direction of NRC, along with an ORISE NaI detector, to perform surface scans of the bottom of the pipe interior. The NaI detectors used for interior piping scans were Ludlum Model 44-10 NaI scintillation detectors coupled to Ludlum Model 2221 ratemeter-scalers with audible indicators. High-density surface scans were performed on both the lower and upper portions of the entire length of the pipe per standard ORISE procedure (without the use of the carrier).

There were no locations of elevated direct gamma radiation identified. All ratemeter-scalers were attached to data-loggers to electronically capture the scan count rate data.

4.3 SURFACE ACTIVITY MEASUREMENTS

Surface activity measurement locations for the bottom of the pipe were laid out in a random start systematic fashion. As introduced in Section 3.6, the number of measurements required was 55; however, due to the systematic spacing, the number was adjusted to 60 to cover the entire length of the pipe. Gamma surface activity measurements were collected using a Ludlum Model 44-10 NaI detector coupled to a Ludlum Model 2221 ratemeter-scaler. Prior to survey, the efficiency was expected to be on the order of 2.3E-03 for a 15.24 cm detector offset, based on Monte Carlo modeling of the NaI detector and the estimated size of LACBWR's Cs-137 calibration standard. The choice of Cs-137 as a calibration standard is conservative because the instrument response from the other gamma-emitters is greater than that of Cs-137.

Efficiency factors for the NaI detector were validated by utilizing LACBWR's flexible large-area Cs-137 calibration standard once the survey team was onsite. The validated efficiency determined by using the site's calibration standard was 3.37E-03, which is slightly higher than the Monte Carlo-based estimated efficiency. All subsequent surface activity calculations were performed using the validated detector efficiency determined onsite. Appendix C.3.2 provides additional information related to the efficiency validation. The surface-activity MDC was 2,700 dpm/100 cm^2, averaged over an area of 3,315 cm^2.

In addition to the gamma measurements, beta surface activity measurements were also collected from the top and bottom of the CWD piping at a frequency of 25% to confirm the appropriateness of the gamma surrogate relationship to Sr-90. Beta measurements were collected using a Ludlum 44-142 beta scintillator detector connected to a Ludlum 2221 ratemeter-scaler. Additional gamma surface activity measurements were collected from the top of the piping at a frequency of 25%; at the same locations as the beta measurements.

5. SAMPLE ANALYSIS AND DATA INTERPRETATION

All confirmatory survey data collected onsite were transferred to the ORISE facility in Oak Ridge, Tennessee for analysis and interpretation. Static surface activity measurements are reported in units of dpm/100 cm^2. Surface activity calculations are discussed in greater detail in Appendix C.

Gamma surface scan data for the top and bottom portion of the pipe were graphed in a quantile-quantile (Q-Q) plot for assessment. The Q-Q plot is a graphical tool for assessing the distribution of a data set. In viewing the Q-Q plots provided, the Y-axis represents gross gamma surface activity in units of cpm. The X-axis represents the data quantiles about the median value. Values less than the median are represented in the negative quantiles, and the values greater than the median are represented in the positive quantiles. A normal distribution that is not skewed by outliers—i.e., a background population—will appear as a straight line, with the slope of the line subject to the degree of variability among the data population. More than one distribution, such as background plus contamination or other outliers, will appear as a step function.

6. FINDINGS AND RESULTS

The results of the confirmatory survey activities are discussed in the subsections below.

6.1 SURFACE SCANS

Overall, NaI detector scan responses ranged from approximately 2,600 to 5,100 cpm for the bottom of the pipe and 2,800 to 5,900 cpm for the top. Table A-2 in Appendix A provides summary statistics for the scan data. No anomalies were identified relative to the localized background. Higher NaI responses were encountered near the north end of the pipe, where the metal pipe had been removed for entry—as would be expected due to an increase in cosmic background radiation. The Q-Q plot for the scan data is presented as Figure 6.1. NaI detector response for the top portion of the pipe was approximately 300 to 800 cpm higher relative to the bottom portion throughout the length of the pipe. However, the increase is not expected to be due to contamination as the detector response was fairly uniform.

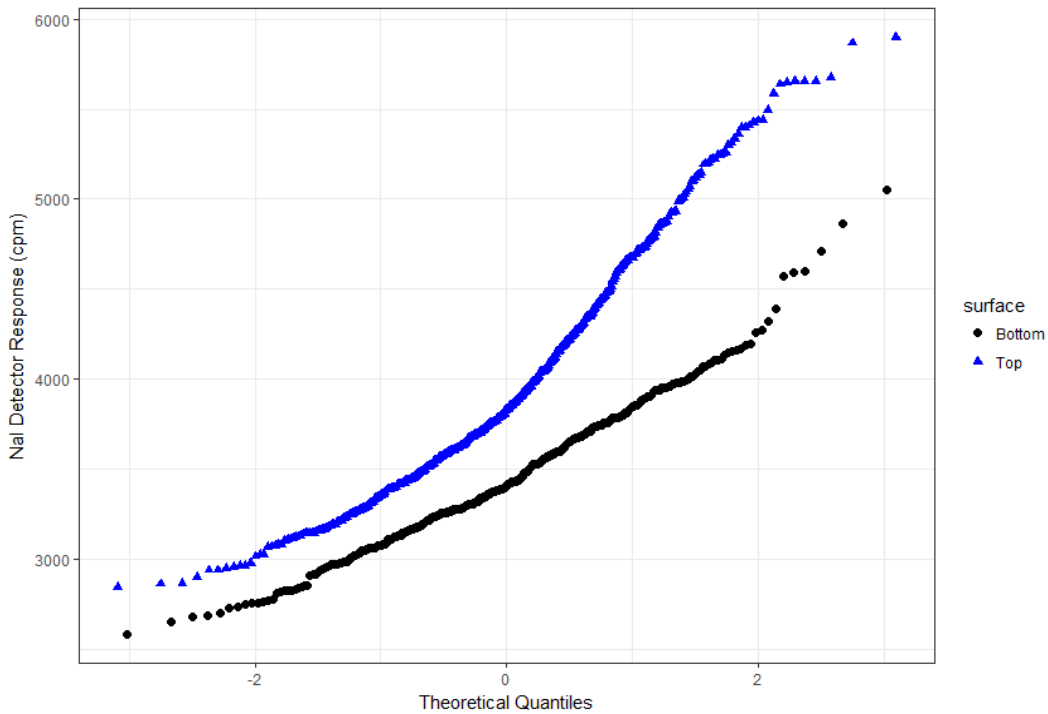

Figure 6.1. ORISE Scan Data for the CWD Pipe

6.2 SURFACE ACTIVITY MEASUREMENTS

General statistics for the ORISE surface activity measurements are provided in Table 6.1. Table A-1 provides individual surface activity measurements.

Table 6.1. General Statistics for ORISE Surface Activity Measurements

Parameter	Bottom (dpm/100 cm²)		Top (dpm/100 cm²)	
	Beta	Gamma	Beta	Gamma
Mean	680	39	4,200	7,700
Median	780	0	4,100	6,700
Standard Deviation	460	3,100	2,100	5,700
Min	-580	7,800	510	1,700
Max	1,300	17,000	8,600	20,000

The maximum observed gamma and beta surface activities on the bottom of the pipe were at 417 feet and 403 feet from the south end of the pipe, respectively. The maximum observed gamma and beta surface activities on the top of the pipe were at 95 feet from the south end of the pipe for both.

Figure 6.2 provides the Q-Q plot for surface activity measurements collected from the CWD piping interior. The plot is segregated by the top and bottom portion of the pipe with the respective beta and gamma data set plotted on the same facet. As indicated by the Q-Q plot, all data sets appear normal, with perhaps the exception of the gamma surface activity data set for the bottom—which appear to contain an outlier at 17,000 dpm/100 cm^2. However, contamination is not expected as this potential outlier was collected from the north end of the pipe where the ambient background was observed to be higher. The mean surface activity for the top portion of the pipe is approximately 7,700 dpm/100 cm^2, which is approximately 7,600 dpm/100 cm^2 higher than the mean of the bottom surface activity measurements. The corresponding count rate difference—in units of cpm—is approximately 860 cpm, which compares favorably to the mean difference in the top/bottom scan data sets, indicating the top portion and bottom portion of the pipe have different backgrounds. Due to the relatively small gamma efficiency, a small difference in count rates corresponds to a multiplicatively larger difference in surface activity. None of the beta or gamma surface activity measurements were above the operational gross activity $DCGL_{Ops}$. The beta surface activity measurements did not indicate the presence of Sr-90 at significant quantities relative to other gamma-emitting ROCs, however no hot spots were identified that would confirm this relationship.

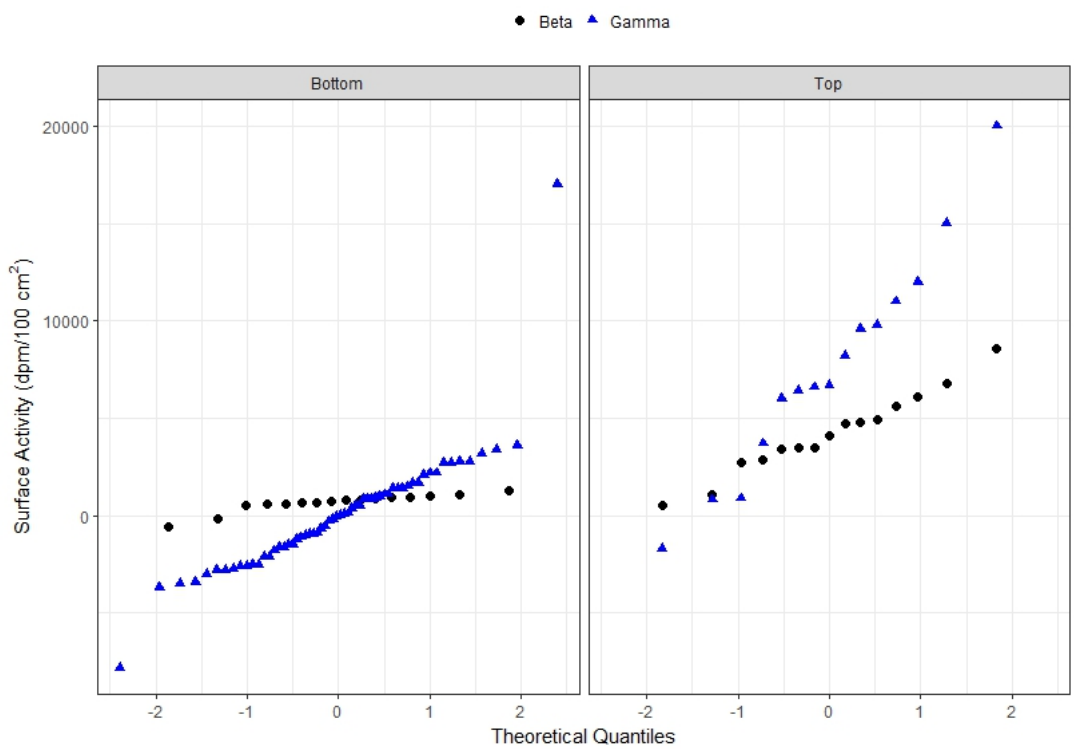

Figure 6.2. Q-Q Plot of Surface Activity for the Top and Bottom Portion of the CWD Piping

The beta surface activity data sets are both approximately normal as indicated by the shape of the Q-Q plot and the similarities of the mean and median. As with the gamma data set, the top portion of the pipe has a higher background than the bottom. Neither data set is centered around zero, as an ambient instrument background was subtracted instead of an SU-specific background, which is conservative.

7. CONCLUSIONS

During the period of April 24 through 26, 2018, ORISE performed an independent confirmatory survey of the interior of the CWD piping at the La Crosse Boiling Water Reactor. The confirmatory survey consisted of gamma scans and surface activity measurements for gamma and beta radiation to independently assess the final radiological status of the CWD piping interior relative to the release criterion. None of the collected surface activity measurements—based on either gamma or beta radiation—were above the $DCGL_{Ops}$. Based on the confirmatory survey data, it was concluded that at least 95% of the CWD piping interior is below the $DCGL_{Ops}$ at the 95% confidence level.

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APPENDIX A
ANALYTICAL RESULTS

Table A-1. Individual Surface Activity Measurements for the CWD Piping Interior									
DM ID	Distance from Start (ft)[a]	Static Count (cpm)				Surface Activity (dpm/100 cm²)			
		Bottom		Top		Bottom		Top	
		Gamma	Beta	Gamma	Beta	Gamma	Beta	Gamma	Beta
CDW-1	4	3187				-1,200			
CDW-2	11	2930	373	3131	310	-3,500	1,100	-1,700	510
CDW-3	18	2982				-3,000			
CDW-4	25	3030				-2,600			
CDW-5	32	3004				-2,800			
CDW-6	39	3047	241	3997	533	-2,500	-180	6,000	2,700
CDW-7	46	3120				-1,800			
CDW-8	53	3145				-1,600			
CDW-9	60	3005				-2,800			
CDW-10	67	3506	335	4980	870	1,700	760	15,000	6,100
CDW-11	74	3572				2,200			
CDW-12	81	3551				2,100			
CDW-13	88	3630				2,800			
CDW-14	95	3487	349	5586	1117	1,500	900	20,000	8,600
CDW-15	102	3618				2,700			
CDW-16	109	3702				3,400			
CDW-17	116	3635				2,800			
CDW-18	123	3157	317	4234	733	-1,500	580	8,200	4,700
CDW-19	130	2907				-3,700			
CDW-20	137	3145				-1,600			
CDW-21	144	3289				-290			
CDW-22	151	3207	363	4643	935	-1,000	1,000	12,000	6,800
CDW-23	158	3031				-2,600			
CDW-24	165	3319				-22			
CDW-25	172	3432				990			
CDW-26	179	2944	313	4419	671	-3,400	540	9,800	4,100
CDW-27	186	3016				-2,700			
CDW-28	193	3046				-2,500			
CDW-29	200	3220				-910			
CDW-30	207	3260	351	3413	544	-550	920	820	2,900
CDW-31	214	3154				-1,500			
CDW-32	221	3086				-2,100			
CDW-33	228	3324				22			
CDW-34	235	3216	340	3739	602	-940	810	3,700	3,400
CDW-35	242	3303				-170			
CDW-36	249	3474				1,400			

Table A-1. Individual Surface Activity Measurements for the CWD Piping Interior									
DM ID	Distance from Start (ft)[a]	Static Count (cpm)				Surface Activity (dpm/100 cm²)			
		Bottom		Top		Bottom		Top	
		Gamma	Beta	Gamma	Beta	Gamma	Beta	Gamma	Beta
CDW-37	256	2446				-7,800			
CDW-38	263	3248	328	4073	740	-660	690	6,700	4,800
CDW-39	270	3419				870			
CDW-40	277	3085				-2,100			
CDW-41	284	3226				-850			
CDW-42	291	3378	201	4060	607	510	-580	6,600	3,500
CDW-43	298	3330				76			
CDW-44	305	3567				2,200			
CDW-45	312	3508				1,700			
CDW-46	319	3624	338	4594	819	2,700	790	11,000	5,600
CDW-47	326	3723				3,600			
CDW-48	333	3675				3,200			
CDW-49	340	3422				900			
CDW-50	347	3376	328	4395	610	490	690	9,600	3,500
CDW-51	354	3364				380			
CDW-52	361	3419				870			
CDW-53	368	3447				1,100			
CDW-54	375	3336	316	4034	750	130	570	6,400	4,900
CDW-55	382	3419				870			
CDW-56	389	3482				1,400			
CDW-57	396	3441				1,100			
CDW-58	403	3195	389	3421	367	-1,100	1,300	890	1,100
CDW-59	410	3473				1,400			
CDW-60	417	5239	351			17,000	920		
	Mean	3325	327	4181	681	39	676	7667	4214
	Median	3322	337	4073	671	0	775	6700	4100
	SD	350	47	635	209	3119	464	5651	2097
	n	60	16	15	15	60	16	15	15

[a]Referenced from the south end of the CWD pipe

Table A-2. Gamma Scan Data Summary Statistics (cpm)				
CWD Pipe Portion	**Minimum**	**Maximum**	**Mean**	**SD**
Top	2,846	5,896	3,970	641
Bottom	2,587	5,050	3,460	389

APPENDIX B
MAJOR INSTRUMENTATION

The display of a specific product is not to be construed as an endorsement of the product or its manufacturer by the author or his employer.

B.1 SCANNING AND MEASUREMENT INSTRUMENT/DETECTOR COMBINATIONS

B.1.1 Gamma

Ludlum NaI(Tl) Scintillation Detector Model 44-10, Crystal: 2-inch × 2-inch
(Ludlum Measurements, Inc., Sweetwater, Texas)
coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)
coupled to:
Trimble Data Logger (Trimble Navigation Limited, Sunnyvale, California)

B.1.2 Beta

Ludlum Plastic Scintillation Detector Model 44-142, Physical Area: 100 cm^2
(Ludlum Measurements, Inc., Sweetwater, Texas)
coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)

APPENDIX C
SURVEY PROCEDURES

C.1 PROJECT HEALTH AND SAFETY

ORISE performed all survey activities in accordance with the *ORAU Radiation Protection Manual*, the *ORAU Health and Safety Manual*, and the *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2014, ORAU 2016c, and ORAU 2016a). Prior to on-site activities, a work-specific hazard checklist was completed for the project and discussed with field personnel. The planned activities were thoroughly discussed with site personnel prior to implementation to identify hazards present. Additionally, prior to performing work, a pre-job briefing and walk-down of the survey areas were completed with field personnel to identify hazards present and discuss safety concerns. Should ORISE have identified a hazard not covered in the *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2016a) or the project's work-specific hazard checklist for the planned survey and sampling procedures, work would not have been initiated or continued until it was addressed by an appropriate job hazard analysis and hazard controls.

C.2 CALIBRATION AND QUALITY ASSURANCE

Calibration of all field instrumentation was based on standards/sources, traceable to National Institute of Standards and Technology.

Calibration of field instrumentation was performed in accordance with procedures from the *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2016a)

Quality control procedures included:

- Daily instrument background and check-source measurements to confirm that equipment operation is within acceptable statistical fluctuations.
- Training and certification of all individuals performing procedures.
- Periodic internal and external audits.

C.3 SURVEY PROCEDURES

C.3.1 Surface Scans

Scans for elevated gamma radiation were performed by passing the detector slowly over the surface. The distance between the detector and surface was maintained at a minimum. Specific scan MDCs for the NaI scintillation detectors were not determined, as the instruments were used solely as a qualitative means to identify elevated gamma radiation levels in excess of local background.

Identifications of elevated radiation levels that could exceed the site criteria were determined based on an increase in the audible signal from the indicating instrument.

C.3.2 SURFACE ACTIVITY MEASUREMENTS

Calibration verification activities were performed using LACBWR's flexible Cs-137 calibration standard placed inside a 152.4-cm-diameter pipe surrogate. A static count was collected with the detector mid-point centered along the width of the source (i.e., along the length of the pipe). The count time was sufficiently long, such that over 10,000 counts were accumulated, thus limiting counting error to less than one percent. The static efficiency was determined to be 3.37E-03. The median of the gamma count rate data for the bottom of the pipe was selected as the SU-specific background for the CWD piping. A representative SU-specific background from a non-impacted area was not available. The SU-specific background selected from the study area was justified based on the approximate normal distribution of the count-rate data. The median of the data set was selected, such that the one data point (near the north end pipe opening) that was elevated would not bias the background estimate high. For a static 1-minute count, the *a priori* minimum detectable concentration (MDC) was determined to be 2,400 dpm/100cm^2 by:

$$MDC\left(\frac{dpm}{100\ cm^2}\right) = \frac{3 + 4.65\sqrt{Bkg}}{\varepsilon \times G}$$

Where:

Bkg = Background count rate, determined to be 3,322 cpm

ε = detector efficiency, estimated to be on the order of 3.37E-03

G = Source area modification factor, which is 33.15 (based on the calibration source area of 3,315 cm^2 (33.15 cm^2/100 cm^2). The detector field of view covers significantly more than this area. Previous ORISE pipe detector calibrations demonstrate that approximately 90% of the detector response to a point source (relative to the source centered with the detector midpoint) occurs within 15 cm from the detector midpoint.

Surface activity measurement data were converted to units of disintegrations per minute per 100 square centimeters (dpm/100 cm^2) using the following equation:

$$SA(dpm/100\ cm^2) = \frac{C - Bkg}{\varepsilon_{tot} \times G}$$

Where:

SA = surface activity

C = measured count rate (cpm)

B = background count rate (cpm) = 3,322 cpm

G = 33.15

ε_{tot} = total efficiency (unitless)

Surface activity measurements for the beta data set were calculated using the previous equation only a G of 1.00 was used and the total efficiency was 0.10, based on Co-60.