Cleanup Plan
Former Benrus Clock Factory Site
Waterbury, Connecticut

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DEFINITIONS

Action level - The numerical value that will cause the decision maker to choose one of the alternative actions. It may be a regulatory threshold standard (e.g., Maximum Contaminant Level for drinking water), a dose- or risk-based concentration level (e.g., DCGL), or a reference-based standard. See investigation level.

ALARA (acronym for As Low As Reasonably Achievable) - A basic concept of radiation protection which specifies that exposure to ionizing radiation and releases of radioactive materials should be managed to reduce collective doses as far below regulatory limits as is reasonably achievable considering economic, technological, and societal factors, among others. Reducing exposure at a site to ALARA strikes a balance between what is possible through additional planning and management, remediation, and the use of additional resources to achieve a lower collective dose level. A determination of ALARA is a site-specific analysis that is open to interpretation because it depends on approaches or circumstances that may differ between regulatory agencies. An ALARA recommendation should not be interpreted as a set limit or level.

Area factor - A factor used to adjust $DCGL_W$ to estimate $DCGL_{EMC}$ and the minimum detectable concentration for scanning surveys in Class 1 survey units—$DCGL_{EMC} = DCGL_W \bullet A_m$. Am is the magnitude by which the residual radioactivity in a small area of elevated activity can exceed the $DCGL_W$ while maintaining compliance with the release criteria.

Arithmetic standard deviation - A statistic used to quantify the variability of a set of data. It is calculated in the following manner: 1) subtracting the arithmetic mean from each data value individually, 2) squaring the differences, 3) summing the squares of the differences, 4) dividing the sum of the squared differences by the total number of data values less one, and 5) taking the square root of the quotient. The calculation process produces the Root Mean Square Deviation (RMSD).

Background radiation - Radiation from cosmic sources, naturally occurring radioactive material,

Calibration - Comparison of a measurement standard, instrument, or item with a standard or instrument of higher accuracy to detect and quantify inaccuracies and to report or eliminate those inaccuracies by adjustments.

Chain of custody - An unbroken trail of accountability that ensures the physical security of samples, data, and records.

Characterization survey - A type of survey that includes facility or site sampling, monitoring, and analysis activities to determine the extent and nature of contamination. Characterization surveys provide the basis for acquiring necessary technical information to develop, analyze, and select appropriate cleanup techniques.

CHP - Certified Health Physicist

CIH - Certified Industrial Hygienist

Class 1 survey - A type of final status survey that applies to areas with the highest potential for contamination, and meets the following criteria: (1) impacted; (2) potential for delivering a dose above the release criterion; (3) potential for small areas of elevated activity; and (4) insufficient evidence to support classification as Class 2 or Class 3.

Class 2 survey - A type of final status survey that applies to areas that meet the following criteria: (1) impacted; (2) low potential for delivering a dose above the release criterion; and (3) little or no potential for small areas of elevated activity.

Class 3 survey - A type of final status survey that applies to areas that meet the following criteria: (1) impacted; (2) little or no potential for delivering a dose above the release criterion; and (3) little or no potential for small areas of elevated activity.

Composite sample - A sample formed by collecting several samples and combining them (or selected portions of them) into a new sample which is then thoroughly mixed.

Confidence interval - A range of values for which there is a specified probability (e.g., 90%, 95%) that this set contains the true value of an estimated parameter.

Contamination - The presence of residual radioactivity in excess of levels which are acceptable for release of a site or facility for unrestricted use.

DAC - Derived Air Concentration

DCGL (derived concentration guideline level) - A derived, radionuclide-specific activity concentration within a survey unit corresponding to the release criterion. The DCGL is based on the spatial distribution of the contaminant and hence is derived differently for the nonparametric statistical test ($DCGL_W$) and the Elevated Measurement Comparison ($DCGL_{EMC}$). DCGLs are derived from activity/dose relationships through various exposure pathway scenarios.

Decommissioning - The process of removing a facility or site from operation, followed by decontamination, and license termination (or termination of authorization for operation) if appropriate. The objective of decommissioning is to reduce the residual radioactivity in structures, materials, soils, ground water, and other media at the site so that the concentration of each radionuclide contaminant that contributes to residual radioactivity is indistinguishable from the background radiation concentration for that radionuclide.

Decontamination - The removal of radiological contaminants from a person, object or area to within levels established by governing regulatory agencies. Decontamination is sometimes used interchangeably with remediation, remedial action, and cleanup.

Derived concentration guideline level - See DCGL.

Detection limit - The net response level that can be expected to be seen with a detector with a fixed level of certainty.

Detection sensitivity - The minimum level of ability to identify the presence of radiation or radioactivity.

Exposure pathway - The route by which radioactivity travels through the environment to eventually cause a person or a group to be exposed to radiation.

Final status survey - Measurements and sampling to describe the radiological conditions of a site, following completion of decontamination activities (if any) in preparation for release.

FSS - Final status survey

Gamma radiation - Penetrating high-energy, short-wavelength electromagnetic radiation (similar to X-rays) emitted

Half-life ($t_{1/2}$) - The time required for one-half of the atoms of a particular radionuclide present to disintegrate.

HASP - Health and Safety Plan

Hypothesis - An assumption about a property or characteristic of a set of data under study. The goal of statistical inference is to decide which of two complementary hypotheses is likely to be true. The null hypothesis (H0) describes what is assumed to be the true state of nature and the alternative hypothesis (Ha) describes the opposite situation.

Impacted area - Any area that is not classified as non-impacted. Areas with a possibility of containing residual radioactivity in excess of natural background or fallout levels.

Infiltration rate - The rate at which a quantity of a substance moves from one environmental medium to another (e.g., the rate at which a quantity of rainwater moves into and through a volume of soil).

Investigation level - A derived media-specific, radionuclide-specific concentration or activity level of radioactivity that - 1) is based on the release criterion, and 2) triggers a response, such as further investigation or cleanup, if exceeded. See action level.

License - A license issued under the regulations in parts 30 through 35, 39, 40, 60, 61, 70 or part 72 of 10 CFR.

License termination - Discontinuation of a license, the eventual conclusion to decommissioning.

MARSSIM - Multi-Agency Radiation Survey and Site Investigation Manual

MDA - Minimum detectable activity

Minimum detectable concentration (MDC) - A priori activity level that a specific instrument and technique can be expected to detect 95% of the time. When stating the detection capability of an instrument, this value should be used. The MDC is the detection limit, LD, multiplied by an appropriate conversion factor to give units of activity.

Minimum detectable count rate (MDCR) - The minimum detectable count rate (MDCR) is the a priori count rate that a specific instrument and technique can be expected to detect.

Non-impacted area - Areas where there is no reasonable possibility (extremely low probability) of residual contamination. NRC - United States Nuclear Regulatory Commission

NVLAP - National Voluntary Laboratory Accreditation Program

OSHA - Occupational Safety and Health Administration

QAPP - Quality Assurance Project Plan

QA/QC - Quality Assurance/Quality Control

Quality assurance (QA) - An integrated system of management activities involving planning, implementation, assessment, reporting, and quality improvement to ensure that a process, item, or service is of the type and quality needed and expected by the customer.

Quality control (QC) - The overall system of technical activities that measure the attributes and performance of a process, item, or service against defined standards to verify that they meet the stated requirements established by the customer, operational techniques and activities that are used to fulfill requirements for quality.

Radioactivity - The mean number of nuclear transformations occurring in a given quantity of radioactive material per unit time. The International System (SI) unit of radioactivity is the Becquerel (Bq). The standard unit is the Curie (Ci).

Radiological survey - Measurements of radiation levels and radioactivity associated with a site together with appropriate documentation and data evaluation.

Radionuclide - An unstable nuclide that undergoes radioactive decay.

Release criterion - A regulatory limit expressed in terms of dose or risk.

Remedial action - Those actions that are consistent with a permanent remedy taken instead of, or in addition to, removal action in the event of a release or threatened release of a hazardous substance into the environment, to prevent or minimize the release of hazardous substances so that they do not migrate to cause substantial danger to present or future public health or welfare or the environment.

Representative measurement - A measurement that is selected using a procedure in such a way that it, in combination with other representative measurements, will give an accurate representation of the isotope being studied.

RESRAD - A computer code used to determine residual radioactivity in the environment by analysis of various exposure pathways.

Sample - A part or selection from a medium located in a survey unit or reference area that represents the quality or quantity of a given parameter or nature of the whole area or unit; a portion serving as a specimen.

Site - Any installation, facility, or discrete, physically separate parcel of land, or any building or structure or portion thereof, that is being considered for survey and investigation.

Soil activity (soil concentration) - The level of radioactivity present in soil and expressed in units of activity per soil mass (typically Bq/kg or pCi/g).

Survey - A systematic evaluation and documentation of radiological measurements with a correctly calibrated instrument or instruments that meet the sensitivity required by the objective of the evaluation.

Survey plan - A plan for determining the radiological characteristics of a site.

Survey unit - A geographical area of specified size and shape defined for the purpose of survey design and compliance testing.

TEDE (total effective dose equivalent) - The sum of the effective dose equivalent (for external exposure) and the committed effective dose equivalent (for internal exposure). TEDE is expressed in units of Sv or rem. See CEDE.

1.0 INTRODUCTION

Decontamination Decommissioning and Environmental Services (DDES), LLC was retained by Cherry Avenue Partners, LP to assemble a Cleanup Plan (CP) for the Former Benrus Clock Factory Site at 145 Cherry Avenue in Waterbury, Connecticut. The former clock factory structure encompasses approximately 60,000 square feet. This Characterization Survey was performed from April 19th – May 5th, 2017. The isotope of concern for the site has been limited to Radium-226 (^{226}Ra) based on the available historical information and clock manufacturing practices. Characterization activities were performed in accordance with the Benrus Clock Factory Site Radiological Characterization Plan dated April 12, 2017 and DDES's Massachusetts Radioactive Materials License 56-0623, via reciprocity.

This Characterization Survey was designed to identify and quantify the current contamination levels present throughout the complex. The survey was designed to identify contamination levels that would exceed a total effective dose equivalent (TEDE) of nineteen (19) mrem/year to an individual member of the public. The goal of this characterization survey was to identify and quantify radiological contamination on site and use this data to assemble a comprehensive cost estimate for unconditional release of the site.

The surveys were designed and executed using the guidance provided in NUREG 1757, "Consolidated NMSS Decommissioning Guidance"; and NUREG 1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM). Individual survey units were subdivided into individual floor elevations. The scan, total and removable contamination survey results from each survey unit was compared to the 19 mrem/year criteria to determine compliance with the release criteria for unrestricted use.

1.1 Introduction and Objective of the Cleanup Plan

This CP describes the remedial actions that will be implemented and defines the site specific radiological release criteria that will be used to show the site has been remediated to meet the 19 mrem/year criteria. Once decontamination activities have been

completed in accordance with this CP, A Final Status Survey Report will be assembled to support the release of the site for unrestricted use, as governed by the U.S. Nuclear Regulatory Commission (NRC) License Termination Rule (LTR), Subpart E of Title 10 of the Code of Federal Regulations (CFR) Part 20.1402 "Radiological Criteria for Unrestricted Use."

This plan was developed using the guidance provided in NUREG 1757, “Consolidated NMSS Decommissioning Guidance” and NUREG 1575, “Multi-Agency Radiation Survey and Site Investigation Manual” (MARSSIM). It provides the approach, methods, and techniques for the radiological decontamination and decommissioning of impacted building surfaces. D&D 2.0 was used to establish derived concentration guideline levels (DCGL) for ^{226}Ra.

1.2 Site Description

The former Benrus Clock Company Site is a 60,000-square foot structure located at 145 Cherry Avenue in Waterbury, Connecticut. This seven-story building is made up of brick and wood structure. The original building is currently owned and occupied by Bender Plumbing. According to ATSDR (1999), Bender Plumbing had extensive testing done at the Benrus factory during the purchase of the property; however, routine testing for radium was not part of a pre-sale environmental study.

A Site Plan depicting the current Site layout and pertinent Site features is provided as Figure 1-1.

Figure 1-1

Former Benrus Clock Factory Site

1.3 Planned Cleanup Activities

The proposed decontamination methods have the potential to disturb interior surfaces and structures with in the historic building envelope. The Connecticut State Historic Preservation Office (SHPO) will review will review site plan, demolition/abatement plans, and architectural drawings relative to historic materials. SHPO will provide comment and assistance to promote compatibility between remediation and preservation of the state's cultural heritage.

Impacted areas will be removed or decontaminated using aggressive techniques. The majority of areas to be remediated are floor surfaces. Floor surfaces that found to have average concentrations above the DGCL will be torn up, planned or sanded until levels

are reduced sufficiently. Contaminated brick/concrete with be scarified or needle gunned to remove the upper surfaces to reduce contamination levels.

We believe this phase of work presents the highest level of hazard to site staff. Both engineering controls and personal protective equipment (PPE) will be required to assure proper protection for staff and to maintain control of the radioactive material. An operational work area will be established to limit the potential emissions of radioactive particulate. Immediate work areas will be separated from the remaining areas of the building by plastic sheeting. High Efficiency Particulate Air (HEPA) filtered air scrubbers will be used to establish negative pressure within the work area. A number of these units will be used to clean the air in the immediate work area. HEPA vacuums and lockdown agent will be used to control radioactive emissions during remediation activities. Elevated levels of respiratory protection, disposable coveralls, nitrile gloves and shoe covers will be required during invasive project activities. Area sampling and personal air sampling will be performed to document the effectiveness of the engineering controls and support the selection of PPE.

We believe a graded approach to the remediation of impacted flooring is necessary to reduce the total volume of radioactive waste generated under this decommissioning. We have estimated that one layer of flooring will require removal to meet the site's remediation goals.

Since only a single layer of flooring will be removed, evaluation by a structural engineer will not be required. If additional floor layers require removal a structural engineer will be retained to assess if the floor/building structure will be compromised by the removal actions to maintain the stability of the building after floor removal. As a waste reduction method, the second layer of flooring will be surveyed after the first layer is removed to confirm if remediation is required.

A negative pressure waste accumulation area will be established by erecting a full containment around the area. HEPA filtration units will be used to maintain negative pressure within the area and create negative airflow in the work area. Area air sampling will be required daily to demonstrate that the engineering controls are effective. Waste will be packaged into cubic yard waste containers and lowered moved to the ground floor by elevator.

Each area where remediation is required will be HEPA vacuumed to remove loose contamination from the remaining surfaces and followed by a comprehensive Final Status Survey of the survey unit to document the area is acceptable for unrestricted release.

The Final Status Survey of the Former Benrus Clock Factory Site is intended to demonstrate the requirements in 10 CFR 20.1402 for unrestricted release have be met. The Final Status Survey is described in greater detail in Appendix B "Final Status Survey Plan".

2.0 FACILITY OPERATING HISTORY

2.1 History

The Benrus Clock Company was founded in New York City in 1921. Benrus also had a factory in Waterbury, CT, which was once the Movement Factory for the Waterbury Clock Company. The Benrus Clock Company in Waterbury, CT historically produced watches with radium-luminous dials.

One Public Health Assessment Survey from 1998 issued by the Agency for Toxic Substances & Disease Registry (ATSDR) was located. This report provided limited dose rate surveys. Readings in excess of twice background were noted on the 4th, 5th and 7th floors of the facility.

A limited spot-check survey was performed on the 4th, 5th and 7th floors of the facility by Scientech in 2003. Elevated levels were noted in each of the areas previously surveyed. These areas of elevated readings needed further quantification and areas of contamination bound to determine the level of remediation necessary to release the Site. Previous surveys were limited due to the amount of material being stored in the facility by Bender Plumbing.

Recent Site History

Bender Plumbing company has recently vacated the premises to a new business facility. Bender Plumbing is a plumbing distribution company that utilized the former clock factory as its corporate headquarters and distribution center.

2.2 Radioactive Material

Radioactive materials onsite are in the form of residual contamination from historic clock making operations in various locations throughout the facility. The building is currently unoccupied. The structure will be converted into condominium units in the near future.

2.3 Potential Contaminants

Table 2-1 lists the potential radioactive contaminants. The site would have used radium luminescent paints while these radioactive materials were exempt from regulation. These legacy ^{226}Ra sites are now regulated under the NRC. No records relating to the use or storage of radium paint onsite could be located.

Nuclides were evaluated by utilizing Default Screening Values (DSV's) generated from a screening analysis using the default parameters contained in the DandD Code v.2.1. Table 2-2 presents the contaminant of concern.

Table 2-2
Contaminant of Concern

Radionuclide	Half-Life	Dispersible Form	Half Life >120 Days
^{226}Ra	1,600 years	Yes	Yes

2.4 Radiological Surveys

To our knowledge, the Former Benrus Clock Factory did not perform radioactive surveys when ^{226}Ra paints were used during the facilities use. DDES performed personnel surveys during scoping and characterization efforts. No detectable personal contamination was found on personnel or visitors during the characterization project. There several areas of removable contamination identified during the Characterization Survey on the 7th floor but all were below 120 dpm/100 cm^2.

2.5 Spills and Uncontrolled Release of Radioactivity

Since no radiological records were required to be kept, no radiological spills have been reported over the history of the site.

2.6 Potentially Impacted Facilities

Table 2-3 lists the survey units that have been potentially impacted by the use of radioactive material and require remediation to meet the cleanup criteria. The majority

of areas identified were flooring, brick surfaces and steam radiators. Table 2-3 presents the list of impacted areas at the site.

Table 2-3

Potentially Impacted Areas (ft^2)

Floor	Wood Floors	Concrete Floors	Brick Walls	Radiators	Doors	Total
1	0	101	0	0	0	101
2	4	0	0	0	0	4
3	180	0	40	2	0	222
4	264	0	100	9	6	379
5	200	20	20	3	16	259
6	72	0	26	0	0	98
7	2,136	80	258	23	6	2,503
Total Area	2,856	201	444	37	28	3,566

2.7 Non-Impacted Areas

The following areas are considered non-impacted based on the characterization survey results:

- Exterior Areas

2.8 Previous Decommissioning Activities

No previous radiological decommissioning activities have been performed at the site to this date.

3.0 FACILITY DESCRIPTION

The National Environmental Policy Act (NEPA) of 1969 (42 USC 4321 et seq) requires Federal agencies, as part of their decision-making process, to consider the environmental impacts of their proposed actions. NUREG-1748, "Environmental Review Guidance for Licensing Actions Associated with NMSS Programs, Final Report", was used to guide the preparation of an Environmental Report for the site to support the NRC's environmental review of this CP.

3.1 Current/Future Land Use

Currently there are no occupants of the building envelope. Developers are currently considering renovating the site for residential use.

4.0 RADIOLOGICAL STATUS OF THE FACILITY

4.1 Building Surface and Structures Contamination

Total contamination measurements were obtained from a systematic grid for Class 1 survey units. The number and spacing of each sample location was determined in compliance with MARSSIM guidelines. Additionally, measurement data were reviewed and compared with the DCGLs and administrative limits to confirm the correct classification of survey units. All calculations of average, standard deviations, minimum and maximum values and all comparisons between survey data, DCGLs and administrative limits are presented in Table 4-1.

Table 4-1

Total Activity Summary

Floor	Class	Total Number	Avg.	Standard Deviation	Min.	Max.	Number Exceeding DCGLw
			(dpm/100 cm^2)				(819 dpm/100 cm^2)
1st	1	114	0	10	0	90	0
2nd	1	114	3	5	0	26	0
3rd	1	114	2	6	0	31	0
4th	1	114	9	19	0	152	0
5th	1	114	4	15	0	148	0
6th	1	121	1	5	0	15	0
7th	1	126	153	676	0	5557	4

A total of 4 locations on the 7th floor had total contamination results exceeding the 819 dpm/100cm^2 criteria. Additional measurements exceeding the DCGL were identified during characterization scan surveys. These locations are presented in Appendix A Radiological Characterization Report.

Removable Contamination Results

Additionally, measurement data were reviewed and compared with the DCGLs and administrative limits to confirm the correct classification of survey units. All calculations of average, standard deviations, minimum and maximum values and all comparisons between survey data, DCGLs and administrative limits are presented in Table 4-2.

Table 4-2

Removable Activity Summary

Floor	Class	Total Number	Avg.	Standard Deviation	Min.	Max.	Number Exceeding DCGLw
			(dpm/100 cm^2)				(20 dpm/100 cm^2)
1st	1	114	0	2	0	6	0
2nd	1	114	0	2	0	6	0
3rd	1	114	0	1	0	6	0
4th	1	114	0	2	0	3	0
5th	1	114	0	4	0	19	0
6th	1	121	0	2	0	3	0
7th	1	126	1	16	0	160	3

A total of 3 locations on the 7th floor had a removable contamination result exceeding the 20 dpm/100cm^2 criteria.

Removable contamination measurements were compared directly to the applicable DCGL scenarios. The removable contamination measurements, apart from three measurements, were less than the administrative limit for removable activity of 20 dpm/100cm^2. The three removable contamination measurements were 22, 66, and 160 dpm/100cm^2

Scanning, total activity and removable contamination measurement results for all surface and structure survey units are provided in Appendix A and B. Survey units where scanning and total contamination measurements did not identify contamination above these limits meet the 19 mrem/year criteria and would not require further remediation. However, a number of survey units had surface scans and total contamination measurements that

exceeded the DCGL of 819 dpm/100cm2 for radium that would require remediation if the suggested limit was accepted by the NRC. The majority of areas identified were large floor areas and window ledges. Cast iron radiators were also identified as remediation areas.

Determining Compliance for Building Surfaces and Structures

Total contamination measurements were compared directly to the applicable DCGL to determine compliance with the 19 mrem/year TEDE requirement for Connecticut DEEP. Surface scans were performed on horizontal surfaces to identify areas of elevated activity that exceeded the DCGL. Additionally, removable contamination measurements were compared to the applicable administrative limit to determine compliance with the applicable administrative limit of 20 dpm/100cm^2. The removable activity measurements, with the exception of three 7th floor locations, collected during the characterization surveys were less than the applicable administrative limit. A total of 26 survey units were found to be in compliance with the 19 mrem/year TEDE limit for unrestricted release and no further radiological evaluation is recommended.

4.2 Building Systems

The designated remediation areas do not contain ventilation systems or drainage systems that would have been used during clock manufacturing. In the event building systems are uncovered during cleanup operations, they will be evaluated for radiological content and removed if contamination levels exceed release criteria.

5.0 DOSE MODELING

An important aspect of the Cleanup Plan (CP) is to assess what the potential radiation dose could result to a potential receptor from the remaining residual radioactivity after decommissioning activities have been completed. The Derived Concentration Guideline Level (DCGL) development analyses simulate the behavior of residual radioactivity over one year, a period during which peak annual doses from the radionuclides of primary interest would be expected to occur. DCGLs were developed for residual radioactivity that will result in 19 mrem per year dose to the average member of the critical group.

The NRC has published default screening values in NUREG 1757 for commonly used radionuclides that is based on 25 mrem/year. DandD v.2.1 software was used to calculate the site DCGL based on 25 mrem/year. Surface contamination limits were derived using the Building Occupancy scenario together with default parameter values. Screening values were selected such that the 0.9 quantile of projected doses was less than or equal to 25 mrem/year (i.e., when probabilistic dose assessment calculations were performed, there was a 90% probability the calculated dose would be less than 25 mrem/year). Total Effective Dose Equivalent calculations are provided as part of the Radiological Characterization Report in Appendix A.

The nuclide of concern (NOC) has been limited to ^{226}Ra. The NOC screening values for surfaces under default conditions (generic screening levels) from the NRC DandD code v.2.1 are provided in Table 5-1, while Administrative Limits are provided in Table 5-2.

Table 5-1
Established $DCGL_w$ for Survey 25 mrem/year

Isotope	Total (DPM/100 cm2)	Removable (DPM/100 cm2)
^{226}Ra	1,116	111

Table 5-2

Established Surface CT DEEP Limit's 19 mrem/year

Isotope	Total Average (DPM/100 cm2)	Removable (DPM/100 cm2)
^{226}Ra	819	20

The term DCGLs will be generically used to describe the proposed levels specified in Table 5-1 and Table 5-2. These criteria are the basis for developing the DCGLs to compare with the survey results. $DCGL_W$ is the concentration limit if the residual activity is essentially evenly distributed over a large area of the survey unit.

In the case of non-uniform contamination, higher levels of activity are permissible over small defined areas. The $DCGL_{EMC}$ is derived separately for these small areas. The $DCGL_{EMC}$ is the $DCGL_W$ increased by an area factor depending on the size of the elevated area. The default area factors for ^{226}Ra listed in MARSSIM will be used.

5.1 ALARA Goals

Due to the extremely low doses associated with the release criteria used for this characterization project, a quantitative ALARA analysis is not required. Default screening values were used to establish DCGLs.

NUREG 1727 states in part: "In light of the conservatism in the building surface and surface soil generic screening levels developed by the NRC staff, the staff presumes, absent information to the contrary, that licensees or responsible parties that remediate building surfaces or soil to the generic screening levels do not need to demonstrate that these levels are ALARA. However, licensees or responsible parties should remediate their facility below these levels through practices such as good housekeeping. In addition, licensees or responsible parties should provide a description in the final status survey report of how these practices were employed to achieve the final activity levels."

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6.0 ALARA ANALYSIS

The proposed decommissioning of the Former Benrus Clock Factory includes the removal and/or decontamination of impacted building surfaces that exceed the established administrative levels for total contamination based on 19 mrem/year (819 dpm/100cm^2) to comply with the Connecticut Department of Energy and Environmental Program (DEEP) requirements. The NRC has published default screening values in NUREG 1757 for commonly used radionuclides based on 25 mrem/year (1,116 dpm/100cm^2) to show compliance with the license termination rule. Further the site will use 20 dpm/100cm^2 as a removable limit which is substantially less than the default value of 111 dpm/100cm^2. This approach is consistent with the ALARA (As Low As Reasonably Achievable) principle. Removal of loose residual radioactivity from buildings is almost always cost-effective except when very small quantities of radioactivity are involved. Therefore, loose residual radioactivity normally should be removed, and if it is removed, the analysis would not be needed, per NUREG-1757, Vol. 2, APPENDIX N

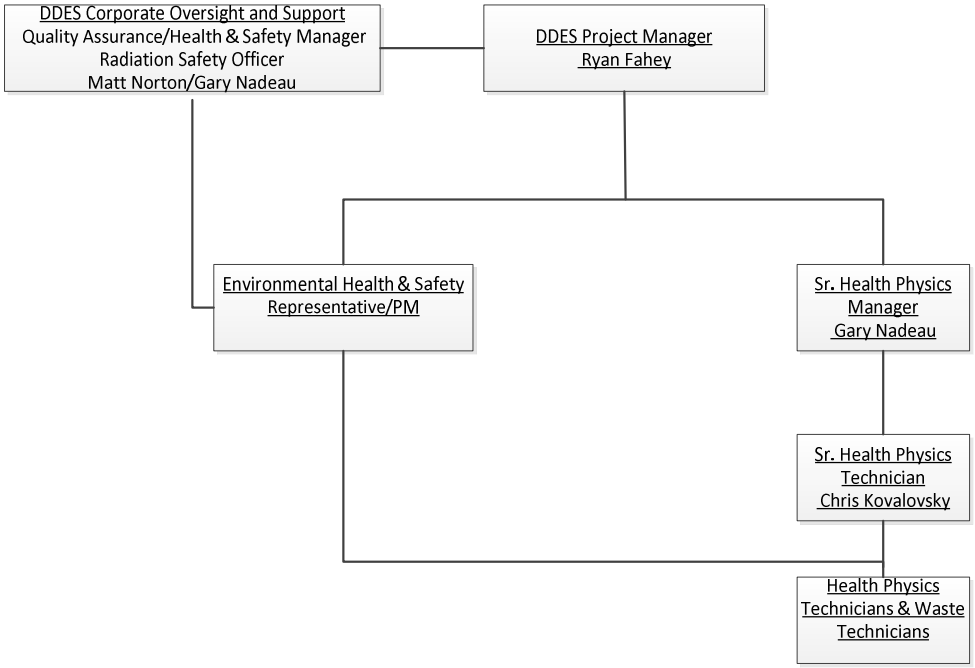
7.0 PROJECT MANAGEMENT AND ORGANIZATION

7.1 Decommissioning Management Organization

Decontamination Decommissioning and Environmental Services (DDES) LLC will maintain primary responsibility for all activities conducted under the requirements of the CP Massachusetts Radioactive Material License No. 56-0623 under reciprocity with the NRC. The point of contact between applicable regulatory authorities and DDES will be the DDES Principal. Figure 7-1 shows the organizational structure for the implementation of the CP.

Figure 7-1

Project Organization



7.2 Decommissioning Task Management

Radiation Work Permits (RWPs) will be used for the administrative control of personnel entering or working in areas that have radiological hazards present. Work techniques will be specified in such a manner that the exposures for all personnel, individually and

collectively, are maintained ALARA. RWPs will not replace work procedures, but will act as a supplement to such procedures. Radiation work practices will be considered when procedures are developed for work which will take place in a radiologically controlled area.

Project RWPs will describe the job to be performed, define protective clothing and equipment to be used, and personnel monitoring requirements. RWPs will also specify any special instructions or precautions pertinent to radiation hazards in the area to be entered including listing the radiological hazards present, known soil and groundwater concentrations and the presence and intensity of hot spots, loose surface radioactivity, and other hazards as appropriate. The radiation safety program will ensure that radiation, surface radioactivity, and equipment surveys are performed as required to define and document the radiological conditions for each job.

RWPs for jobs with low dose commitments (less than 25 millirem total effective dose equivalent or TEDE) will be approved at the health physics (HP) technician or HP supervisor while RWPs for jobs with potentially significant radiological hazards will be approved by the RSO. Examples of topics covered by implementing procedures for the RWPs are:

- Requirements, classifications and scope for RWPs;
- Initiating, preparing and using RWPs; and
- Terminating RWPs.

Work procedures and practices associated with radioactive materials, such as RWPs, will be developed, reviewed, implemented and managed pursuant to DDES health physics procedures.

7.3 Decommissioning Management Positions and Qualifications

7.3.1 Radiation Safety Officer

The RSO will be an employee of DDES and will have a Bachelors' degree in the physical sciences, health physics, industrial hygiene or engineering from an accredited college or university, with at least one (1) year of work experience in applied health physics, industrial hygiene or similar work relevant to radiological hazards, and a thorough knowledge of the proper application and use of all radiation safety equipment used in connection with the radioactivity present at the site, the chemical and analytical procedures used for radiological sampling and monitoring, and methodologies used to calculate personnel exposure to the radionuclides present at the site. The name and qualifications of the individual serving as RSO for this work, who will be available full-time for the decommissioning, will be provided to the NRC prior to the start of the on-site efforts.

7.3.2 Other Management Positions

7.3.2.1 Decommissioning Contractor

DDES will prepare the final work plans, perform decontamination activities, perform and document Final Status Surveys, facilitate communications with federal and state regulatory authorities, and provide on-site project management and site-specific health and safety support (radiological, industrial hygiene, and industrial safety support) during the implementation phase.

7.3.2.2 Project Manager

The DDES will designate an individual to serve as the Project Manager. The Project Manager, who will have training and education in applicable

radiological, engineering and environmental aspects of decommissioning, as well as expertise in managing projects of this magnitude, will be responsible for the following:

- Verifying that the personnel are provided with the proper radiation protection, industrial safety training and possess the requisite knowledge for the job assignment;
- Observing work in progress to verify adherence to the radiological and industrial safety rules and procedures;
- Recommending changes to operational and radiological protection practices;
- Enforcing compliance with DDES facility rules and license requirements;
- Reviewing reports and results; and
- Establishing and maintaining a records management system to verify that project documents, such as correspondence, procedures, drawings, specifications, contract documents, changes to documents, and inspection records are controlled.

The Project Manager has the responsibility and authority to stop or terminate any work activities that do or may violate regulatory or contract requirements.

7.3.2.3 Site Health and Safety Officer

Reporting to the Project Manager will be the Site Health and Safety Officer (Site HSO). This individual will be present at the Former Benrus Clock Factory Site for the duration of all on-site decommissioning work, and is to have a combination of education and experience in the following radiation protection and industrial safety subjects:

- Principles and practices of radiation protection;

- Radioactivity measurements, monitoring techniques, and the use of instruments;
- Mathematics and calculations basic to the use and measurement of radioactivity;
- Biological effects of radiation;
- Safety practices applicable to protection from radiation, chemical toxicity, and other properties of the materials that may be encountered during the decommissioning;
- Conducting radiological surveys and evaluating results;
- Evaluating and implementing the final work plans for proper operations from a radiological safety standpoint;
- Applicable NRC, EPA, and OSHA regulations, as well as the terms and conditions of any licenses and permits issued by regulatory agencies; and the requirements contained in License No. 56-0623 The responsibilities of the Site HSO will include, but are not limited to the following:
- Establishing the health and safety program requirements for field activities;
- Verifying that the requirements of the industrial safety and radiation protection program are implemented adequately;
- Reviewing the results of surveys, sampling, and environmental monitoring to identify trends and potential for personnel exposure;
- Evaluating the effectiveness of engineering and administrative control including the requirements for personnel protective equipment;
- Developing new safety protocols and procedures necessary for new field activities;
- Providing internal review and approval for work related documents;
- Auditing key aspects of the safety and health program; and
- Making recommendations to the Project Manager regarding the control of existing and potential industrial, chemical and radiological hazards.

The Site HSO has the responsibility and authority to terminate any work activities if conditions indicate the potential for unnecessary radiation exposure to site personnel or members of the public, or for unsafe working conditions.

7.3.2.4 Quality Assurance Officer

An individual, with a reporting line directly to the RSO, will be assigned to serve as the Quality Assurance Officer (QAO) for the project." The QAO, who will have training in the implementation of quality programs, will perform the following:

- Technical assistance and peer review of all deliverables;
- Prepare and review the Quality Assurance Project Plan (QAPP);
- Coordinate with analytical laboratories, as necessary;
- Oversee subcontractor QA activities to ensure compliance with the QAPPs;
- Track laboratory submittals and sample analyses and verify delivery of data, as necessary;
- Coordinate validation of analytical data;
- Monitor the on-site activities; and
- Prepare and submit QA reports, as required.

The QAO has the responsibility and authority to stop or terminate any work activities that may lead to conditions adverse to the quality requirements of the CP.

7.4 Training

All employees, contractors, and visitors with unescorted access to the restricted area of the facility will be trained on the types and magnitudes of the radiological hazards they might face. All personnel performing the on-site work described in this CP will be current

in the training required in 29 CFR 1910.120. The following subsections briefly describe the various training programs that will be implemented as part of the CP.

7.4.1 Visitor Training

Visitors to the work zone will be trained by a combination of reading, oral briefing, and signing a briefing form. The briefing form will contain information about the hazards present in the work zone, and the requirement that all visitors be escorted while in the work zone.

7.4.2 Radiation Worker Training

Radiation Worker Training (RWT) will be administered to all employees performing invasive activities with radioactive materials. RWT will address the following topics:

- Radioactivity and radioactive decay;
- Characteristics of ionizing radiation;
- Man-made radiation sources;
- Acute effects of exposure to radiation;
- Risks associated with occupational radiation exposures;
- Special considerations in the exposure of women of reproductive age;
- Dose-equivalent limits;
- Modes of exposure - internal and external;
- Dose-equivalent determinations;
- Basic protective measures - time, distance, shielding;
- Specific procedures for maintaining exposures as low as reasonably achievable (ALARA);
- Radiation survey instrumentation, calibration, use and limitations;
- Radiation monitoring programs and procedures;
- Contamination control, including protective clothing, equipment and work place design;
- Personnel decontamination;
- RWP issue, modification, termination and use;

- Emergency procedures;
- Warning signs, labels, and alarms;
- Responsibilities of employees and management; and
- How to contact project radiation safety staff.

RWT will consist of a classroom lecture and procedure review, a half-hour practical and demonstration. The duration of training will be approximately three (3) hours. An exam to test employee proficiency in the class topics shall be administered and a passing score of 80% is required. Refresher training will be provided annually thereafter.

7.4.3 Tailgate Safety Training

Tailgate safety meetings will be conducted at the beginning of each work shift, whenever significant changes are made in job scope, or whenever new personnel arrive at the job site. The meetings will present radiation safety and health and safety procedures and issues for the day, any unique hazards associated with an activity and review any significant topics from previous activities. The information discussed will be recorded, which will serve as confirmation that the information was presented to those persons whose signatures are on the form.

7.4.4 Training Records

A form will be developed to demonstrate that training commitments are being met. The form will capture the following information: the facility, date, time, type of work, hazardous/radioactive materials used, protective clothing/equipment, chemical hazards, radiological hazards, physical hazards, emergency procedures, special equipment, and any other safety topics that may be relevant.

8.0 HEALTH AND SAFETY PROGRAM

DDES is committed to implementing this Cleanup Plan (CP) in a manner that ensures the health and safety of workers, the surrounding environment and the public. Consequently, comprehensive health and safety requirements and access controls will be specified in the final work plans. These requirements will remain in effect during all decommissioning activities. DDES will also verify there is sufficient documentation to demonstrate the effectiveness of the health and safety program.

This chapter of the CP describes those measures that will be used to control and monitor the impacts of ionizing radiation on workers. The Radiation Protection Program described herein is designed to be compliant with U. S. Nuclear Regulatory Commission (NRC) regulations in 10 CFR Parts 19 and 20 and provisions in License No. 56-0623 and is implemented through a set of approved radiation safety procedures.

At a minimum, DDES will maintain a copy of DDES radiation safety procedures for regulatory inspection.

Each member of the decommissioning project team will assume certain health and safety responsibilities. These will include, but are not limited to, the following:

- The RSO is responsible for providing oversight for implementation of the Decommissioning Work Plan and making changes to reflect field situations that were not anticipated during the plan's initial development. Changes in the radiation protection program can only be made with the concurrence of the RSO.
- The designated health and safety contact for each subcontractor is responsible for verifying field implementation of the radiation protection program provisions. This verification includes communicating site requirements to all personnel on the job, field supervision, and consultation with the RSO regarding appropriate changes to this Cleanup Plan.
- All on-site project personnel are responsible for understanding and complying with all site health and safety requirements, including proper maintenance of health and safety

equipment and facilities. This understanding will be documented by signature prior to any team member being authorized to work on decommissioning operations.

DDES will provide a work-place environment in which employees, visitors and contractors are adequately protected from hazards, including the hazards associated with exposure to radiation and radioactive material. While the exposures associated with the planned decommissioning operations are low, all exposures are assumed to entail some risk to the employee.

The ALARA requirement will be communicated to all personnel at the outset of this project. All individuals must understand their responsibilities to reduce their radiation exposure. Methods to be used to achieve exposure reduction will be reviewed during General Employee Training and Tailgate Safety Training. Monitoring and surveillance information will be summarized and reviewed by the work force on a planned and periodic basis.

8.1 Radiation Safety Controls and Monitoring for Workers

Radiation, airborne radioactivity and contamination surveys during decommissioning will be conducted in accordance with approved procedure(s) (RSP-008, RSP-009, RSP-010). The purposes of these surveys will be to:

- Protect the health and safety of workers,
- Protect the health and safety of the general public, and
- Demonstrate compliance with applicable license, federal and state requirements, as well as Cleanup Plan commitments.

Radiation safety personnel assigned to the project will verify the presence and adequacy of posted radiological warning signs during the conduct of these surveys. Surveys will be conducted in accordance with procedures utilizing survey instrumentation and equipment suitable for the nature and range of hazards anticipated. Equipment and instrumentation will be calibrated and, where applicable, operationally tested prior to use in accordance with procedural requirements. Routine surveys will be conducted at a specified frequency to ensure that contamination and radiation levels in unrestricted

areas do not exceed license, federal, state or site limits. Radiation protection staff will also perform surveys during decommissioning whenever work activities create a potential to impact radiological conditions.

Control levels have been established for the decommissioning actions. Based upon knowledge of the radiological constituents present at the site and existing exposure rates, it is expected that maximum individual personnel exposures will not exceed 100 millirem TEDE over the life of the project. Surveillances will be performed to verify that exposures are minimized and within applicable limits.

Because the exposure potential is expected to be less than 500 millirem TEDE, individual monitoring for on-site personnel is not required, however DDES will require both internal and external dose monitoring for the decommissioning project as a conservative measure.

8.1.1 Work Area Air Sampling Program

The air sampling program during decommissioning will be implemented to assure that workers are adequately protected from inhalation of radioactive material. Air sampling will be performed for decommissioning activities involving disturbance or handling of contaminated material where employee exposures could be expected to receive an internal exposure of more than ten (10%) percent of an Annual Limit on Intake (ALI, as specified in 10 CFR 20, Appendix B, Table 1) for Ra-226. DDES will collect both personal and area sampling using approved procedures for collecting representative air samples.

Following collection of air samples, a screening analysis for gross alpha air activity will be performed. Samples will be analyzed on a gas-flow proportional counter (or instrument of similar sensitivity). In order to account for Radon-222/220 interference during this initial analysis and determine if airborne radioactivity

concentrations are at acceptable levels, alpha to beta ratios will be determined and used. Final air sampling results will not be available for three days following collection to allow for decay of Radon-222/220 short-lived daughters. Once the final results are available, airborne radioactivity concentrations will be documented in accordance with approved procedures.

8.1.2 Respiratory Protection Program

In controlling the concentrations of radioactive materials in air, the use of process controls, engineering controls or administrative procedures will be used. Examples may include the use of exhaust ventilation, diversion of air flow, dust suppression, fixative coatings, etc. The use of respiratory protection will be implemented to reduce the projected work dose in compliance with the ALARA principal.

The program will require use of National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA) certified equipment, and procedures that comply with 10 CFR 20, Subpart H. At a minimum, respiratory protection procedures will address the following elements:

- Monitoring, including air sampling and bioassays;
- Supervision of the program, including program audits;
- Training and minimum qualifications of respirator program supervisors and implementing personnel;
- Training of respirator users, including the requirement for each user to inspect and perform a user seal check (for face-sealing devices) or an operational check (non-face-sealing devices) on a respirator each time it is donned;
- Fit-testing;
- Selecting respirators;
- Maintaining breathing air quality;
- Inventory and control of respiratory protection equipment;

- Storage and issuance of respiratory protection equipment;
- Maintenance, repair, testing, and quality assurance of respiratory protection equipment;
- Recordkeeping; and
- Limitations on periods of respirator use and relief from respirator use.

The Project Manager and the RSO will jointly determine the need for and on the procedural requirements prior to implementing a respiratory protection program.

NIOSH approved loos fitting positive pressure air purifying respirators will be used (i.e., full face piece assemblies with air purifying elements to provide respiratory protection against hazardous vapors, gases, and/or particulate matter).

Respiratory protective equipment will be kept in proper working order. When any respirator shows evidence of excessive wear or has failed inspection, it will be repaired or replaced. Respiratory protective equipment that is not in use will be stored in a clean dry location.

8.1.3 Internal Exposure Determination

A combination of bioassay and breathing zone air sampling will be used to determine internal exposures incurred by decommissioning workers while on site. Workers will submit a bioassay at the beginning of the project and at the end of the project due to the short project timeframe. In addition, special bioassays may be required in the event air sample data and/or process knowledge warrants stricter monitoring. All samples will be analyzed by a laboratory that meets the performance criteria in ANSI N13.30.

The RSO (or designee) will determine the validity of bioassay and air monitoring results prior to their inclusion in the internal dose assessment process. The RSO will typically evaluate the following items to ascertain the validity of monitoring results:

- sample collection errors;
- radiation background interference during counting;
- calibration errors;
- computer software errors;
- errors due to counting geometry; and
- statistical errors.

Only valid bioassay or air monitoring results, as determined by the RSO, will be used for assessment of internal radiation dose. If the data are not valid, the RSO will document the basis for that conclusion and include the documentation in the individual's dosimetry record. The RSO will also estimate the internal dose to the individual via other means and include the estimate in the individual's exposure history. The RSO will identify the route of entry (i.e., inhalation, ingestion, etc.), as the most likely route based upon current knowledge of exposure conditions. The lung clearance class for intake by inhalation will be selected based upon current knowledge of the chemical form and/or particle size.

The CEDE (stochastic) incurred by workers will be estimated by:

$$CEDE\ (millirem) = \frac{Intake}{ALI_S} \times 5{,}000$$

where Intake = the activity taken into the body as determined from bioassay measurements, and ALIS = the stochastic Annual Limit on Intake for the radionuclide of interest. Committed dose equivalents to particular organs or tissues of interest (CDE) will be estimated by:

$$CDE_T\ (millirem) = \frac{Intake}{ALI_{NS}} \times 50{,}000$$

where Intake = the activity taken into the body as determined from bioassay measurements or air monitoring results, and the ALINS = the non-stochastic

Annual Limit on Intake for the radionuclide of interest. To determine the contribution of CDE to the CEDE, the CDE is multiplied by the appropriate organ dose weighting factor specified in 10 CFR 20.1003.

In general, minors will be excluded from work with the potential for intakes of radioactive material. Internal exposure determinations for declared pregnant workers will be based on air monitoring results unless the RSO (or designee) determines that special bioassay sampling is warranted. These intakes will be converted into a dose to the embryo/fetus based on methodologies discussed in Regulatory Guide 8.36.

8.1.4 External Exposure Determination

Individual monitoring devices, at a minimum, will consist of a whole body thermoluminescent dosimeter (TLD) or equivalent (e.g., optical dosimeter, etc.). The TLDs will be ordered from a vendor that has been approved in advance by the Decommissioning Contractor, and whose program has met the requirements of ANSI N13.11. In addition, the vendor must demonstrate accreditation by National Voluntary Laboratory Accreditation Program (NVLAP), which includes range, sensitivity and accuracy performance criteria. TLDs will be processed at the end of the decommissioning and individual doses will be tracked.

8.1.5 Summation of Internal and External Exposures

Internal and external radiation exposures will be assessed for the project. The total organ dose equivalent (TODE) is computed by summing the deep dose equivalent (DDE) from external sources, as determined from external radiation monitoring, and the CDE, as determined from internal radiation monitoring. The TEDE is determined by summing the CEDE from sources internal to the body and the HD.

8.2 Contamination Control Program

The procedures for accessing contaminated areas will address the responsibilities of all personnel permitted access, contamination limits, posting, labeling and tagging requirements, protective clothing requirements of each level of contamination encountered, entry and exit requirements, measurement methodologies, decontamination of personnel and training requirements, as described in the RPP procedures. Routine surveys will be performed throughout the decommissioning activities to assess contamination control methodologies.

The initial level of protection for the intrusive tasks of the decommissioning operations (i.e., where residual radioactivity may be encountered) will be hard hats, Tyvek coveralls, safety glasses with side shields, steel-toed boots, and gloves. Upgrading or downgrading of the level of protection will be specified by the RWP.

To assure radioactive materials remain under the control, each person performing invasive activities frisk using calibrated, handheld instrument prior to leaving the contaminated work area. Equipment, people and materials will be frisked and decontaminated prior to exiting the controlled area. Records of release surveys will be maintained on standardized forms and maps. Release criteria will be consistent with those shown in the Final Status Survey Plan (Appendix B).

8.3 Instrumentation Program

Radiation survey equipment and instrumentation suitable for detecting and quantifying the radiological hazards to workers and the public will be used. The selection of equipment and instrumentation to be utilized will be based upon knowledge of the radiological contaminants and concentrations that were identified during the characterization survey.

All instruments will be calibrated and maintained according to applicable RPP Procedure. Instruments will be calibrated using radiation sources which are National Institute of Standards and Technology (NIST) traceable. The methods used to estimate uncertainty bounds for each type of instrumental measurement will be as specified in ANSI N323.

Each instrument will be response-checked using a reference source and have pre-operational checks performed daily. Pre-operational checks will include battery function, response to reference source, reset button function, audible response function if applicable, physical condition, current calibration and response to background radiation. These results will be documented and any instruments failing any of the pre-operational checks will be tagged and taken out of service.

8.4 Health Physics Audits, Inspections and Recordkeeping

During the implementation of this CP, at least one assessment of the effectiveness of the RPP will be performed per calendar year by an individual who is certified by the American Board of Health Physics in the comprehensive practice of health physics and has demonstrated experience performing assessments of radiation safety programs for decommissioning projects. The results of this assessment will be reported to the Project Manager and the RSO.

Informal assessments and inspections will be completed by the RSO (or designee) on a daily basis, with unexpected, non-conforming, or unusual items and situations documented, along with their resolution. These assessments and inspections will include performance and documentation of radiological surveys, radiological work practices, posting and labeling, contamination control, and internal and external dosimetry. Due to the frequency of these informal assessments and inspections, they serve as routine, unannounced inspections by the RSO (or designee).

Any findings identified during formal assessments or informal assessments and inspections will be evaluated by the RSO for compliance with license commitments or NRC requirements. Documentation of such evaluations will be maintained and available for inspection, including corrective actions taken to prevent recurrence and any follow-up to verify effectiveness of corrective actions. Records of RSO audits will include the dates of the audit, the name of the auditor, persons contacted by the auditor, areas audited, audit findings, corrective actions, and any follow-up required.

9.0 ENVIRONMENTAL MONITORING AND CONTROL PROGRAM

9.1 Environmental ALARA Evaluation

We are committed to reducing exposures to radioactive materials to levels that are ALARA. Exposures should be reduced to ALARA to employees and members of the public living near the site. Potential pathways for exposure exist during the removal of contaminated building surfaces. Engineering and administrative controls will be implemented when performing invasive activities which could generate radioactive particulate inside the buildings. No invasive activities are planed exterior of the building.

High-Efficiency Air Particulate (HEPA) filtered negative pressure air scrubbers will be used in the general work area. These will be combined with the use of HEPA filtered vacuums at the point of operation to control fugitive emissions. Fixative agents will be used to further reduce the levels of removable contamination on surfaces designated for removal.

10.0 RADIOACTIVE WASTE MANAGEMENT PROGRAM

10.1 Solid Radioactive Waste

It is estimated that approximately 35 cubic yards of radioactive waste will be generated from the removal of flooring from the decontamination of building surfaces. An additional 5 cubic yards of radioactive waste will be generated from plastic sheeting, PPE, used HEPA filters, etc. from materials that come into direct contact with contamination while performing the scope of work. Radioactive waste management is key to controlling the total cost for the decommissioning of this site. The majority of waste will be composed of wood flooring that will have a low density; therefore, packaging of the material to get the highest density per cubic foot will optimize budget for waste disposal.

These wastes should be packaged into the largest conveyance possible to maximize packaging efficiency and minimize the total number of shipments. These waste containers will be staged on-site and loaded individually within the containment. Containers may be lined with a roll-off liner that can be tied of and sealed. A tarp may be used to cover each container prior to transport. When each container has met the best possible density, it will be sealed and covered in the methods mentioned above and staged in an area outside the containment for future transport. It is our recommendation that no more than three containers be staged for a limited time outside the containment. Coordination of waste pickup and transport on a timely basis will ensure security of staged waste containers.

The radioactive waste will be shipped direct to US Ecology in Wayne, Michigan or the US Ecology facility in Grandview, Idaho. Both of these US Ecology facilities are permitted to accept the remediation waste generated under this scope.

There is the potential to generate radioactive waste that contains lead based paint from remediation activities at the site. These wastes will be separated from non-;lead containing wastes. They will be tested by an offsite laboratory using the Toxicity

Characteristic Leaching procedure (TCLP) to determine if they area hazardous for lead. These wastes are acceptable for disposal at the US Ecology facilities previous listed. A separate waste acceptance profile will be completed and waste stabilization will occur at the designated facility prior to disposal.

11.0 QUALITY ASSURANCE PROGRAM

11.1 Quality Assurance Project Plan

DDES shall develop a Quality Assurance Project Plan (QAPP) utilizing the guidelines of MARSSIM Section 9 for the Final Status Survey. The QAPP will incorporate at a minimum, the following:

- Description of the Quality Assurance and Quality control goals, Data Quality Objectives (DQO), procedures, and plans to be implemented for all D&D activities.
- Description of the methodology to ensure that all radiological survey data meet the 95% confidence level.
- The QAPP will be developed and organized with emphasis given to maximizing worker safety; minimizing/eliminating off-site releases and minimizing overall project costs. The quality control program will control all quality documents during the performance of D&D operations. Quality documents include, but are not limited to:
    - Training Records
    - Survey Records
    - Instrumentation Records
    - Shipping Records
    - Work Procedures and Plans

12.0 Release Surveys

12.1 Materials and Equipment Release Criteria During Decommissioning

Release surveys for materials and equipment that may become surface-contaminated during decommissioning will be performed using portable radiation survey instruments. Table 12-1 describes the criteria for materials and equipment release as they are taken from Table 1 of U. S. Nuclear Regulatory Commission (NRC) Regulatory Guide No. 1.86 ("Termination of Operating Licenses for Nuclear Reactors", June, 1974), which is consistent with the guidance found in NUREG-1757, Vol. 1, Section 15.11.1.1:

Table 12-1 Material and Equipment Release Criteria

Radionuclide(1)	Acceptable Surface Contamination Limits (dpm/100 cm^2)		
	Removable(2), (5)	Average(2), (3)	Maximum(2), (4)
Th-Natural, Th-232, Sr-90, Ra-223, Ra-224, U- 232, I-126, I-131, I-133	200	1,000	3,000
U-Natural, U-235, U-238, and associated decay products	1,000	5,000	15,000

Notes	
(1)	Where surface contamination by both alpha- and beta-gamma emitting nuclides exists, the limits established for alpha-and beta-gamma emitting nuclides are to be applied separately.
(2)	As used in this table, disintegrations per minute (dpm) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.
(3)	Measurements of average contaminant will not be averaged over more than one (1) square meter. For objects of less surface area, an average will be derived for each object.
(4)	The maximum contamination level applies to an area not more than 100 square centimeters (cm^2).
(5)	The amount of removable radioactive material per 100 cm^2 of surface area will be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels will be reduced proportionately and the entire surface will be wiped

12.1.1 Remedial Action Support Surveys

DDES will conduct remedial action surveys to guide clean-up actions and provide updated estimates of site-specific parameters used for final status survey planning. These surveys may consist of surface scans, volumetric screening, composite sampling and offsite analysis.

12.1.2 Final Status Survey Design

At the conclusion of decommissioning activities, DDES will submit to the NRC a Final Status Survey Report that is compliant with the content requirements specified in the Plan.

13.0 FINANCIAL ASSURANCE

13.1 Cost Estimate

The cost estimates for the decommissioning actions described in this CP were developed using a variety of cost-estimating data, including vendor-provided information, conventional cost-estimating guides, prior experience, and prior similar estimates as modified by site-specific information. Site-cost experience and good engineering judgments were also used to identify those items that will control the estimates. In addition, the following were also assumed:

- The estimated inventories of radioactive materials are representative of the quantities that are present at the time decommissioning activities begin.
- The decommissioning effort will begin upon NRC approval of this DP.
- Unit costs presented in the cost estimates represent materials, labor, equipment, and overhead and profit (O&P) costs. For cost data sources that did not include O&P, a value equal to 25% of the combined materials, labor and equipment cost was used to represent O&P.
- In accordance with NRC guidance, a 25 percent contingency has been added to both the capital costs and the long-term surveillance and monitoring costs of all alternatives.

The estimated present cost for the decommissioning the Benrus Building is $112,000 and would be completed in two weeks.

13.2 Certification Statement

Cherry Hill Partners LP will be responsible for funding the CP and Final Status Survey of the Former Benrus Clock Factory Site.

Appendix A
Radiological Characterization Report

Appendix B

Final Status Survey Plan