DECOMMISSIONING PLAN

REVISION 1

(PREVIOUSLY IDENTIFIED FUSRAP AREAS EXCEPT DEBRIS PILES & SITE BROOK)

CE WINDSOR SITE
WINDSOR, CONNECTICUT

US NRC LICENSE NUMBER 06-00217-06
DOCKET NUMBER 030-03754

(DECEMBER 2008)

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Abbreviations and Acronyms

ACM	asbestos containing material
AEC	Atomic Energy Commission
ALARA	as low as reasonably achievable
bgs	below ground surface
CE	Combustion Engineering, Inc.
CFR	Code of Federal Regulations
CMU	concrete masonry unit
CP	Characterization Plan
CPM	counts per minute
CTDEP	Connecticut Department of Environmental Protection
D&D	decontamination and deconstruction
DAC	derived air concentration
DCGL	derived concentration guideline level
DP	Decommissioning Plan
dpm/100cm^2	disintegrations per minute per 100 square centimeters
ELI	evaporator line investigation
EM	electromagnetic
EMC	elevated measurement comparison
EU	enriched uranium
FSS	Final Status Survey
FUSRAP	Formerly Utilized Sites Remedial Action Program
HASP	Health and Safety Plan
HEPA	high efficiency particulate air filter
HEU	high enriched uranium
HP	Health Physics
HPGe	High Purity Germanium
HRR	Historical Review Report
HSA	Historical Site Assessment
KAPL	Knolls Atomic Power Laboratory
LEU	low enriched uranium
LLRW	low-level radioactive waste
MACTEC	MACTEC, Inc.
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDA	minimum detectable activity
MDC	Metropolitan District Commission
MSL	mean sea level
MW	monitoring well
NaI	sodium iodide
NFM	nuclear fuel manufacturing
NMM	nuclear materials manager
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
ORISE	Oak Ridge Institute for Science and Education
PCB	polychlorinated biphenyls
PM	project manager
PPE	personal protective equipment
PVC	polyvinyl chloride
QAP	Quality Assurance Plan
QC	quality control

Abbreviations and Acronyms

RCM	Radiological Controls Manager
RCRA	Resource Conservation and Recovery Act
RESRAD	Residual Radioactive Material Guidelines computer code
RSO	Radiation Safety Officer
RSR	Remediation Standard Regulation
RWP	radiation work permit
SAIC	Science Applications International Corporation
Site	CE Windsor Site
SNM	special nuclear material
TSCA	Toxic Substances Control Act
USACE	U.S. Army Corps of Engineers
USDOE	U.S. Department of Energy
USEPA	U.S. Environmental Protection Agency
VCA	Voluntary Corrective Action
WRS	Wilcoxen Rank Sum
WWTP	Wastewater Treatment Plant

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1.0 EXECUTIVE SUMMARY

From the mid-1950s, the Combustion Engineering (CE) Site in Windsor Connecticut has been involved in research, development, engineering, production, and servicing of nuclear fuels, systems, and services. This site-wide Decommissioning Plan (DP) provides the decommissioning information necessary to lead to license termination and unrestricted release in accordance with the requirements of the License Termination Rule at 10 CFR Part 20, Subpart E. This DP addresses all pertinent information as described in NUREG-1727, "NMSS Decommissioning Standard Review Plan" (NRC, 2000a), and as updated by NUREG-1757, "Consolidated Decommissioning Guidance" (NRC, 2006a; NRC, 2006b; NRC, 2003).

Certain buildings and areas on the Site were being addressed by the U.S. Army Corps of Engineers (USACE) under the Formally Utilized Sites Remedial Action Program (FUSRAP). Recently, an agreement was reached between the USACE and the U.S. Nuclear Regulatory Commission (NRC) that will allow ABB, Inc. (ABB) to complete decommissioning at the Site. Since there is extensive commingling of FUSRAP and NRC-licensed material, the USACE proposed to suspend FUSRAP activities at the Site in order to allow cleanup under NRC decommissioning (USACE, 2007). The NRC accepted this approach which allows ABB to supplement the existing DP, decommission the remainder of the Site pursuant to NRC regulations and complete license termination (NRC, 2007a). When the remediation effort is complete, ABB will demonstrate that NRC dose criteria are met for unrestricted license termination of the entire CE Site.

Remediation is planned under the DP for the remaining impacted areas. Remediation will include decontamination of buildings, demolition of structures to ground surface, removal of floor slabs and footings 3 (or 4) feet below ground surface, removal of underground utilities and any soils impacted above the derived concentration guideline levels (DCGLs), and final status survey. The south end of Building 3 (High Bay) will be unconditionally released and remain operational since it houses unique fossil fuel research facilities.

One additional area has been identified during previous decommissioning activities. The Burning Grounds was previously remediated and subsequently released for unrestricted use by the NRC (NRC, 1989). This area was associated with burning thorium (for disposal), and is included in this revision to the DP.

The purpose of Revision 1 to the DP is to include the entire Site for decommissioning under the jurisdiction of the NRC. Furthermore, it provides additional detail regarding remaining impacted portions of the Site that are similar to the prior remediated Commercial D&D areas. This will allow ABB to continue decommissioning in a timely manner while the more environmentally challenging portion of the Site (Site brook) undergoes a more detailed evaluation and review process. Therefore Revision 1 to the DP provides an update with respect to completed decommissioning activities (Commercial D&D areas) along with radiological status and remediation plans for Buildings 3 and 6, industrial waste lines, equipment storage yard, woods area, drum burial pit, clamshell pile and burning grounds.

A future revision to the DP will provide radiological status and remediation plans for Site brook and the adjacent debris piles. These areas are not included in Revision 1 since these areas are within wetlands and the anticipated remediation will require special permits in order to reduce the impact and plans for restoration.

Revising the October 15, 2003 DP to include the previously designated FUSRAP and Burning Grounds areas instead of developing an entirely new DP is possible due to the fact that the original DP addressed the entire Site, although it did not address remediation in the previously designated FUSRAP areas. However, the site-specific DCGLs were developed to be protective of uranium and byproduct materials that could be encountered across the entire Site, including the previously designated FUSRAP areas. Thus the remaining portions of the Site simply need to be added to the existing approved sitewide DP in order to allow decommissioning activities to proceed.

The following sections outline the CE Windsor Site Decommissioning Plan.

1.1 SITE AND LICENSEE INFORMATION

The CE Windsor Site is located in the Town of Windsor, Connecticut, eight miles north of Hartford, Connecticut. The entire Site consists of approximately 600 acres. The name and address of the licensee are:

ABB Inc.
2000 Day Hill Road
Windsor, CT 06095-0500

The address where licensed material will be used or possessed is, and where all correspondence concerning this license should be sent:

ABB Inc.
C/o John Conant
CEP 880-1911
2000 Day Hill Road
Windsor, CT 06095-0500

1.2 SUMMARY OF LICENSED ACTIVITIES

From the early 1960s to 2000, CE was involved in the research, development, engineering, production, and servicing of nuclear systems and fuel. Projects included nuclear research for commercial use. Nuclear fuel research and development was conducted in Buildings 2 and 5 and fuel was manufactured in Building 17. Liquid radiological waste was processed in Building 6A for a short period of time prior to 1961. Buildings 3 and 6 initially were designed and built for Naval nuclear fuel manufacturing at the Site.

Buildings 17 and 21 were built in 1967 and 1969, respectively. Building 17 was used for nuclear fuel manufacturing (NFM) from 1968 to 1993. Operations were moved off-site in 1993 and Building 21 was decommissioned. In 1998, Building 17 was renovated for use by Nuclear Field Operations.

1.3 NATURE AND EXTENT OF SITE CONTAMINATION

Residual contamination on facility structures, systems and components is the result of the deposition of uranium (primarily enriched UO_2) and byproduct materials. During operations, routine surveys were performed on process areas and waste streams to determine radionuclide distributions.

The primary radionuclides identified at the Site are U-234, U-235, U-238, and associated transformation products, Co-60 and Cs-137. The Final Historic Site Assessment (HSA) (Harding ESE, 2002) identified building and environmental locations where these radionuclides are known to be or potentially exist. These areas were targeted during characterization to determine the remediation and work control requirements.

As part of Commercial D&D, Building Complexes 2, 5, 6A and 17 have been decontaminated and dismantled and the below grade utilities have been removed. In addition, soil remediation to meet DCGLs has been completed, Final Status Surveys for these Complexes have been completed and accepted by the NRC (NRC, 2007b).

Building Complexes 3 and 6 were abandoned years ago, and remaining equipment, machinery, etc., and interior systems will be decontaminated and dismantled as necessary. Above grade structures are planned to be deconstructed including the removal of the building slabs and foundations. The south end of Building 3 (High Bay) will be released for unrestricted use using existing license criteria and will remain in use for fossil fuel research and development. Dismantlement is expected within 24 months of the approval of DP Revision 1.

The remaining radiologically impacted areas of the Site will be remediated as necessary. This will include removal of soil, piping, debris and other materials that are identified during decommissioning activities. Targeted areas for remediation are described in this revision with the exception of Site brook and the debris piles which will be provided in a future revision of the DP. Figure 4-11 shows the impacted areas at the CE Windsor Site, and identifies the current status for each.

1.4 DECOMMISSIONING OBJECTIVE

It is the objective of ABB to decommission the Site, including associated buried piping and adjacent grounds, such that the entire Site will meet the criteria for unrestricted use as specified by 10 CFR 20.1402, and to terminate NRC License No. 06-00217-06.

1.5 SITE SPECIFIC DCGLS

The objective of ABB is to decontaminate and decommission the CE Windsor Site facilities and lands in accordance with applicable Federal and State requirements and regulations such that the radioactive materials license held by ABB can be ultimately terminated.

Radiological contamination at the Site will be addressed following guidance in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NUREG-1575) (NRC, 2000b) through the preparation of a Historical Site Assessment and the development of the site-specific concentrations of licensed material in soil as DCGLs. The DCGLs are presented in the Derivation of Site-Specific Soil DCGLs report (ABB, 2003), which was submitted with DP Revision 0 and approved by the NRC (NRC, 2004). There are no changes to these site-specific DCGLs since they were specifically designed to address the various mixtures of uranium enrichments that are present at the Site, including previously designated FUSRAP materials.

The Burning Grounds were previously released by the NRC for unrestricted use (NRC, 1989) using Branch Technical Position release limits (NRC, 1981). Supplemental remediation will address cleanup criteria for two additional radionuclides identified during investigational sampling (Ra-226 and Th-232). Instead of calculating site-specific DCGLs, the NRC screening values (NRC, 1999) will be used in the Burning Grounds for the two additional radionuclides because small volumes are present.

Building DCGLs for the Site have also been developed since at least one potentially impacted building structure will remain at the time of license termination. The DCGLs are presented in the Development of Building DCGLs report (ABB, 2008). The building DCGLs are designed to address both byproduct and uranium residual materials that are present at the Site.

1.6 ALARA ANALYSIS

Based on the licensee's decision to remediate to unrestricted use criteria, and using appropriate dose modeling to relate concentrations to dose, the licensee can take advantage of the allowance given in Section 1.5, Appendix N of NUREG-1757, Volume 2 (NRC, 2006b) which states "In certain circumstances, the results of an ALARA analysis are known on a generic basis and an analysis is not necessary. For residual radioactivity in soil at sites that will have unrestricted release, generic analysis show that shipping soil to a low-level waste disposal facility is unlikely to be cost effective for unrestricted release, largely because of the high cost of waste disposal. Therefore shipping soil to a low level waste disposal facility generally does not have to be evaluated for unrestricted release." It goes on to state, "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." In this regard, the results of an as low as reasonably achievable (ALARA) analysis are "known on a generic basis and an analysis is not necessary."

However, to keep in the spirit of ALARA, simplified analyses for soils and buildings (cost vs. activity levels, possible benefits and costs relating to decommissioning, and a determination of residual radioactivity levels that are ALARA) have been developed. Each analysis was completed in the context of NUREG-1757,Volume 2 Appendix N. The analyses are based upon release criteria derived from site-specific dose modeling; i.e. the DCGLs referred to in Section 5 of this report. The ALARA analyses are presented in Section 7.0 of this report.

1.7 START AND END DATES

Remediation activities for the remaining areas of the Site will commence upon NRC approval of this revision to the DP. The majority of activities will continue for approximately 2 years, and decommissioning is planned to be complete within 3 years.

1.8 POST-REMEDIATION ACTIVITIES

No post-remediation activities have been identified beyond those discussed in this Decommissioning Plan.

1.9 AMENDMENT TO LICENSE TO INCORPORATE DP

This Decommissioning Plan revision submittal includes a request to amend NRC License No. 06-00217-06 to incorporate the Decommissioning Plan Revision 1.

2.0 FACILITY OPERATING HISTORY

2.1 LICENSE NUMBER, STATUS, AND AUTHORIZED ACTIVITIES

Current NRC License No. 06-00217-06 authorizes possession and use of licensed material at the CE Windsor Site.

The radionuclides and maximum activities and quantities and chemical form of radionuclides authorized under the current license are included in Table 2-1.

The license authorizes possession and use for those activities directly or indirectly related to decontamination, and decommissioning of buildings, systems, facilities and property at the CE Windsor Site. The licensee may under this license perform decontamination, monitoring, packaging, storage, and shipment of residual waste and receipt of licensed calibration standards without prior NRC approval.

License No. 06-00217-06 was renewed on April 27, 2001, and expires on April 30, 2011. Amendments issued since the last renewal include:

- Amendment No. 49 -RSO Amendment – Approved the change of the designated Site Radiation Safety Officer from Robert Clark to Robert Woodard.
- Amendment No. 50 - D&D Amendment – Approved the completion of specified activities for the decontamination and decommissioning of Building Complexes 2, 5, and 17.
- Amendment No. 51 – Administrative – Deleted License conditions 15 and 20 that appeared in Amendment No. 50. Also revised License condition 13 into the current NRC approved format.
- Amendment No. 52 – Administrative and Procedural – Deleted License condition 16 that appeared in Amendment No. 51. Also approved the request to possess and use radioactive material in the former "Health Works" complex at the facility in order to operate a counting laboratory.
- Amendment No. 53 – Approved the deconstruction and disposition of materials from the Building 2, 5 and 17 Complex areas, not including the removal of the building slabs.
- Amendment No. 54 – DP Amendment – Approved the Decommissioning Plan and DCGLs for decontamination and decommissioning of the Site.
- Amendment No. 55 – RSO Amendment – Approved the change of the designated Site Radiation Safety Officer from Robert Woodard to Robert Clark.
- Amendment No. 56 – Administrative – Reduction in possession limits.
- Amendment No. 57 – RSO Amendment – Approved the change of the designated Site Radiation Safety Officer from Robert Clark to Robert Woodard
- Amendment No. 58 – RSO Amendment – Approved the change of the designated Site Radiation Safety Officer from Robert Woodard to Heath Downey.

- Amendment No. 59 – Administrative – Changed the name of the licensee from ABB Prospects, Inc. to ABB Inc.
- Amendment No. 60 – Administrative – Incorporates areas that were previously considered FUSRAP into the license.

2.2 LICENSE HISTORY

From the early 1960's to the present, CE has been involved in the research, development, engineering, production, and servicing of nuclear and fossil fuel systems. These activities were performed under both commercial and federal contracts. Projects included nuclear and combustion research for commercial use, as well as large-scale boiler test facilities and coal gasification. Nuclear fuel research and development was conducted in Buildings 2 and 5 and nuclear fuel was manufactured in Buildings 3, 5 and 17. Liquid radiological waste was processed primarily in Building 6, and in Building 6A for a short period of time prior to 1961.

Buildings 17 and 21 were built in 1967 and 1969, respectively. Building 17 was used for nuclear fuel manufacturing (NFM) from 1968 to 1993. Operations were moved off-site in 1993 and Building 21 was released for unrestricted use by the NRC in 1997. Building 21 was then dismantled in 2001. In 1998, Building 17 was renovated for use by Nuclear Field Operations.

ABB's current materials license is License number 06-00217-06 and special nuclear materials License No. SNM-1067. Other licenses that have been in effect at the CE Windsor Site include License No. SNM-551 and License No. 06-30561-01 (Westinghouse). The maximum activities of radionuclides authorized under these licenses, the chemical forms of the radionuclides authorized, and a description of how the radionuclides were used are included in Tables 2-2 through 2-5. Recent research under FUSRAP has revealed the existence of various source licenses from the 1950's and 1960's. These licenses are briefly listed in Table 2-6.

2.3 PREVIOUS DECOMMISSIONING ACTIVITIES

2.3.1 Building 21

Building 21 was a warehouse for the Building 17 NFM Facility. The building is located directly west of Building 17. The NRC released Building 21 for unrestricted use in 1997.

The Building 21 footprint was approximately 10,000 ft^2. The building was a one-story metal structure constructed on a concrete slab.

Building 21 was constructed in 1969 as a warehouse for non-nuclear materials and parts, and for the receipt of UO_2. The warehouse was used to store supplies and sealed drums of UO_2 powder or pellets for Building 17 operations.

In 1993, all uranium manufacturing operations were consolidated to an offsite location. Building 21 was shut down in 1993, a decommissioning plan was submitted to the NRC in 1996, and D&D operations were completed. In 1997, the building was released by the NRC for unrestricted use.

2.3.2 Building Complexes 2, 5, 6A and 17

License Amendment No. 50 approved the decontamination and dismantlement of systems and components inside the buildings of Building Complexes 2, 5, and 17. This work is complete. License Amendment No. 53 approved the decontamination and deconstruction of the above-grade structures of the buildings of Building Complexes 2, 5, and 17 subject to NRC approval of building radiological survey report. Building 6A was decontaminated and deconstructed and the Building Complexes 2, 5, 6A and 17 building slabs, subsurface utilities and contaminated soil above DCGLs were removed under DP Revision 0. All decommissioning activities in Building Complexes 2, 5, 6A and 17 are complete, Final Status Surveys have been performed and these areas have been accepted by the NRC (NRC, 2007b).

2.3.3 Building Complexes 3 and 6

Building 3 was originally designed and constructed as a NFM Facility. When nuclear operations ceased in the early 1960s, the building was decontaminated and renovated for fossil fuel research and development. Building 6 has been used for liquid radiological waste handling since operations began at the Site. During routine operations, the tanks were periodically cleaned. When liquid radiological waste handling and processing activities ceased, the tanks were also emptied.

2.3.4 Outdoor Areas

Several outdoor areas were remediated by the licensee during the 1980s and 90s. This included removal of stored drums and equipment, excavation of buried drums and debris, and removal of soil. The former Burning Grounds area was remediated, had verification surveys performed by ORISE and was released for unrestricted use by the NRC (NRC, 1989).

2.4 SPILLS

There may have been a potential release of radiologically impacted wastewater from the systems that received wastewaters from Building Complex 5 and Building 17.

The subsurface soil between Buildings 6 and 5 may contain contamination along waste pipelines. The radiological waste pipeline between Buildings 6 and 6A is constructed of concrete and iron (dur-iron). The waste line received radiological wastes from Building 5 and Building 17, and it is likely that the pipeline leaked (Harding ESE, 2002). Use of these lines was terminated in 1998.

There are no other known spills associated with Building Complexes 2, 5, 6A or 17. These areas have been remediated and are complete as stated in Section 2.3.2 above.

There are similar concerns for radiological and industrial waste lines located in remaining areas of the Site. This includes lines connecting Buildings 3 and 6 in addition to the pipelines that were the primary liquid radiological waste system for the Site that run underground from Building 6 to Site brook (length of 3,700 feet). The original industrial waste pipeline was reported to have a blockage or leak, so a replacement line was installed in the 1970s.

3.0 FACILITY DESCRIPTION

3.1 SITE LOCATION AND DESCRIPTION

The CE Windsor Site is located in the Town of Windsor, Connecticut, eight miles north of Hartford, Connecticut (Figure 3-1). The Site consists of approximately 600 acres. Figure 3-2 shows the overall site layout and also delineates the building complex areas included in the Commercial Area.

The CE Windsor Site is industrially zoned by the Town of Windsor, and is located in a Mixed Land Use area of Hartford County. Nearby land uses are primarily commercial, agricultural, industrial, and residential. Much of the northern and western portions of the property are wooded.

The Site is bordered by Day Hill Road and agricultural and commercial land to the south; commercial development and a sand and gravel quarry to the west; the Windsor/Bloomfield Sanitary Landfill and Recycling Center (Landfill) to the north; and forested land as well as residential and commercial developments to the east. The northwest corner of the property is bordered by the Rainbow Reservoir portion of the Farmington River.

Surface water bodies on Site include: Great pond, located on the southwestern end of the property; Small Pond, located east of the Site buildings; Goodwin Pond; and the Site brook, both located on the northeastern portion of the property. The Site brook flows to the northwest from Goodwin Pond into the Farmington River at the northwest property boundary (Figure 3-2).

Approximately one-third of the property is developed with buildings, infrastructure, and maintained landscaping. The remaining two-thirds of the property is wooded, and may or may not have been disturbed by historic operations performed by either CE or previous owners. Parts of the wooded areas are known to have been excavated for fill, used to stage drums, and/or used as a historic disposal area.

Generally, the developed areas of the Site are mostly paved and/or landscaped and relatively flat. The wooded area along the northeastern portion of the property is less disturbed by Site development. There are several areas where historical operations have altered the land surface either by excavation and/or filling activities. The highest portion of the Site is approximately 210 feet above mean sea level (MSL). The Site topography drops to 98 feet above MSL along the banks of the Farmington River. Topographic contours are shown on Figure 3-2.

3.2 POPULATION DISTRIBUTION

The Site is located mostly within a commercial and agricultural area, however residential properties are present within 1/4 mile of the site boundaries (Figure 3-3). The regional socio-economic conditions for the area around the CE Windsor Site are discussed below.

The State of Connecticut's Hartford County is made up of 29 municipalities, including the Town of Windsor where the CE Windsor Site is located. According to the 2000 Census, the region population is about 23 percent minority. For this assessment, minority populations are identified as those communities within the region where the percent of the minority population exceeds the average for the region (USDOE, 1996). There are three minority population centers in the region: the Towns of Bloomfield and Windsor, and the City of Hartford. Of the three, the largest percentage of minorities is in the City of Hartford.

Low-income populations are identified as those communities within the region for which the percent of the population living in poverty exceeds 25 percent (USDOE, 1996). According to the 1990 Census, about eight percent of the regional population is at or below the poverty level. This proportion is consistent with the state average of eight percent, estimated by the 2000 Census.

The population distribution within a 50-mile radius of the Windsor Site is shown on Figure 3-4. Table 3-1 summarizes the population distribution. Table 3-2 presents socio-economic factors for Hartford County and the Town of Windsor based on recent 2000 Census data.

According to the Connecticut State Department of Economic and Community Development, employment in the area totals about 19,215 positions divided between manufacturing (20 percent) and non-manufacturing (80 percent). The majority of the manufacturing jobs involve fabricating metals, aircraft, and machinery. The majority of the non-manufacturing jobs involve agriculture, wholesale trade, retail, financial, insurance, real estate, services, and government services (CTDECD, 1999). The unemployment rate for the Town of Windsor was 3.1 percent in August 2001, slightly below Hartford County at 3.6 percent. (CTDOL, 2001). The Windsor Facility currently employs approximately 3,500 personnel (Flemming, 2001).

There are no known residences, schools, or day care centers within 200 feet of the CE Windsor Site. The nearest residences to the CE Windsor Site are located in Birchwood, north of the Farmington River (approximately 500 feet north of the Site). There are ten schools located within the Town of Windsor and one early childhood center. There are also several (15 or more) smaller parks and recreational areas within the Town of Windsor. Large recreational facilities include Pennwood State Park and Pine Hill Golf Club (both located within 2 miles of the CE Windsor Site).

Within this two-mile radius, approximately 600 people rely on public groundwater supply sources. Public water supply wells are located in East Granby Village (well numbers 1 and 2) and in Chelsea Commons (well numbers 1 and 2). These wells are located 1.8 miles and 1.9 miles, respectively, from the CE Windsor Site. Most of the people in the area are served by the Metropolitan District Commission (MDC) public water supply. Municipal water, however, is not available adjacent to the western portion of the Site on Beman Lane and on a portion of Tunxis Avenue from Beman Lane north to the Farmington River (Fuss & O’Neill, 1999). In addition, to the south and east of the Site, a few houses on Prospect Hill Road (near intersection of Silver Birch Lane and Day Hill Road), as well as all properties along Huckleberry Road are not served by the MDC and are inferred to use private wells (Fuss & O’Neill, 1999).

3.3 CURRENT AND FUTURE LAND USE

It is anticipated that future uses of the Site will be roughly consistent with its current use (commercial, light industrial uses). The current land use in the surrounding area is a mixture of commercial, light industrial, warehousing, office park, residential, municipal landfill, and commercial farming. The land use is trending toward commercial and industrial uses.

Commercial farming of both consumable produce and tobacco does occur in the near vicinity of the CE Site. Such commercial farms are characterized as large fields that are planted with a single commercially viable crop (such as tobacco, corn, cucumbers, etc.) and is harvested in bulk and trucked to a commercial wholesale buyer for subsequent distribution. These farms do not support residential habitation or subsistence. Locally such farms are known as “truck” farms.

Future residential use of the land is considered possible given the current community growth, planning, and development strategies of the local municipality. Therefore it is reasonable and credible to consider that the land might be used for locating residential dwellings in the future.

Subsistence farming is considered incredible at this Site because 1) the general population is moving away from subsistence farming, 2) the amount of land required to support subsistence farming is economically infeasible considering the value of the land, and 3) the population demographics are consistent with east coast urban/suburban uses.

3.4 GEOLOGY

The regional geology in Windsor is mapped within the Central Valley landscape of the Newark Terrain. The underlying bedrock is mapped as Portland Arkose and has been encountered at 90 and 120 feet below ground surface (bgs) at two locations at the Site. The overburden consists of quaternary deposits. The most pronounced feature is a dense till ridge, or drumlin, that trends north-south and is located in the middle of the Site. The southern end of the ridge is located near Building 6A, and the northern end of the ridge is located near the Greater Than 90 Day Storage Area. Weathered till outcrops at the ground surface, both behind Building 6A and in the woods near the Waste Pad area within the woods.

West of the ridge, the overburden consists of stratified sands and silts. These deposits become finer with depth and have been investigated to depths of approximately 120 feet bgs without encountering till or bedrock.

East of the ridge, the overburden consists of fine sands in silts to approximately 40 to 60 feet bgs. These deposits are underlain by ablation till that flanks the drumlin and pinches out to the east. In several borings, a coarse sand and gravel water-bearing zone was encountered at approximately 90 to 105 feet bgs. These coarse sands have been encountered within a soil unit that trends north-south and is located beneath Small Pond. This unit is not continuous to the east and west.

3.5 HYDROGEOLOGY

The hydrogeology at the CE Windsor Site has been characterized over the course of the Resource Conservation and Recovery Act (RCRA) Voluntary Corrective Action (VCA) program with the installation and sampling of over 130 groundwater monitoring wells. Construction details for all Site monitoring wells are presented in Table 3-3.

In 1999, the Connecticut Department of Environmental Protection (CTDEP) approved a change in the classification of the groundwater under most of the site from Class GA to Class GB. This change in classification considers that the groundwater beneath the northern portion of the Site is not a present or future source for GA quality use, and that nearby groundwater, e.g., at the neighboring landfill to the north, is presently classified as GB (Figure 3-5).

Class GA is groundwater within the area of existing private water supply wells or an area with the potential to provide water to public or private water supply wells. The Department presumes that groundwater in such areas is, at a minimum, suitable for drinking or other domestic uses without treatment.

Class GB is groundwater within a historically highly urbanized area or an area of intense industrial activity where public water supply service is available. Such groundwater may not be suitable for human consumption without treatment due to waste discharges, spills, or leaks of chemicals or land use impacts.

Depth to groundwater ranges from 0 ft below ground surface (bgs) at surface water boundaries to about 50 ft bgs in the central and northern portions of the Site. In the developed southern portion of the Site, groundwater depths measured in wells range from approximately 12 to 20 feet bgs. In the central and northern portion of the Site, groundwater depths range from 35 to 50 feet bgs, with the exception of the area near the Site brook and Farmington River, where the topography drops steeply and groundwater discharges to the surface water bodies.

Shallow groundwater is expected to flow generally toward the northwest and discharge to the Farmington River. However, on a smaller scale, local influences affect groundwater flow direction. These influences include the till ridge, surface topography, and surface water bodies. Groundwater flows from the southwestern portion of the Site and is diverted around the till ridge to the east and west. To the west of the ridge, groundwater flows to the northwest towards the Farmington River. To the east of the ridge, groundwater roughly follows the surface waters from Small Pond to Goodwin Pond, then follows the Site brook to the Farmington River.

Interpreted groundwater contours are shown on Figure 3-6.

Groundwater elevation data for Site-wide monitoring well sampling events in 2000 and 2001 are summarized in Table 3-4. Vertical gradients, determined from adjacent wells installed at different elevations within the aquifer, are presented in Table 3-5.

In the eastern portion of the Site, a deeper transmissive water bearing zone has been identified. Figure 3-7 presents the interpreted piezometric surface from wells established deeper within the water table and above the underlying dense till. This zone is hydraulically connected to the shallow water table groundwater; however, it is less influenced by surface water and local topography. As seen on Figure 3-7, flow in the area of Small Pond is towards the northeast. Regional topography suggests that deeper flow could continue towards the northeast to discharge into the Farmington River in its lower reach before its confluence with the Connecticut River. Shallow flow most likely follows the surficial drainage system and recharges Goodwin Pond and Site brook.

<u>Groundwater Flow Velocity</u>

During the RCRA VCA field programs, water level recovery (rising-head slug) tests were performed at a total of 22 selected wells. Resulting hydraulic conductivity (K) values varied widely, but could be placed into two general characterizations of aquifer materials. In tills, or silty sands just above tills, calculated K values ranged from about 0.3 to 3 feet per day (ft/d). In fine to courser sands, the K values ranged from about 3 to 40 ft/d. Results of hydraulic conductivity testing are presented in Table 3-6.

Typical estimated groundwater velocities were calculated using the locally determined hydraulic conductivity and hydraulic gradient, and an assumed porosity of 0.3. The assumed porosity of 0.3 is based on published estimates (Domenico & Schwartz, 1990), and is typical of fine sands. The estimated velocities ranged from 11 to 70 feet per year (ft/yr). The arithmetic average of the estimates calculated for the CE Windsor Site is 35 ft/yr, while the geometric mean (an estimate of the median of an assumed log-normally distributed velocity field) was 28 ft/yr. These results are somewhat higher than the limited previous data that indicated Site groundwater velocities ranging from 1 to 10 ft/yr.

It should be noted that all groundwater velocity and conductivities were calculated for horizontal flow within the water bearing zones. No measurements were conducted to estimate vertical flow properties, however, based on the horizontal laminations and overburden structures visible in the soil borings, it is reasonable to assume that the vertical conductivity and velocities will be at least one to two orders of magnitude lower.

3.6 SURFACE WATER HYDROLOGY

There are several surface water bodies on or adjacent to the CE Windsor Site. The Farmington River flows along the northern boundary of the property (Figure 3-1). Goodwin Pond and the Site brook (unnamed) are located within the northern portion of the property. The Site brook flows from Goodwin Pond, then northwest to the Farmington River. Small Pond located northeast of Building 3, drains into Goodwin Pond via a well-defined drainage way. Great Pond, a glacial kettle-hole pond with no outlet, is located just inside the southwestern property border. Silver Birch Pond lies approximately 0.5 miles east of the property (USGS, 1984a; USGS, 1984b). Small streams located south of the property may influence groundwater contour shapes along the Site's southern property line.

According to the Water Quality Classification of Connecticut map, surface waters in Great Pond and Silver Birch Pond are Class A, Goodwin Pond and the Site brook are Class B/A, and the Farmington River is Class BC (CTDEP, 1991). Class A surface waters are known or presumed to meet water quality criteria that support designated uses. Class A defined uses include potential drinking water supplies, fish and wildlife habitat, recreational use, and agricultural and industrial supplies. Class B surface waters are known or presumed to be suitable for fish and wildlife habitat, recreational use, and agricultural and industrial supplies. Class B/A surface waters may not be meeting water quality criteria for one or more Class A designated uses. Class BC surface waters are presumed to be suitable for supporting cold water fisheries (CTDEP, 1991).

With the exception of the southwestern section of the property, storm water runoff from the property flows into Small Pond, a small brook leading to Goodwin Pond, Goodwin Pond itself, and the Site brook. Surface drainage at the southwest end of the property flows toward Great Pond, where there is also an infiltration collection system that receives overflow from Great Pond during periods of high water. Great Pond has no other apparent outlet (USGS, 1984b), and likely maintains its level mainly through interaction with groundwater.

The Site brook flows into the Farmington River on the northwestern property boundary. The Farmington River subsequently flows into the Connecticut River approximately 10 river miles downstream from the CE Windsor Site (USGS, 1984a; USGS, 1984b).

The closest designated fishery to the property is the Farmington River, bordering the CE Windsor Site to the northwest, as listed in the 1992 State of Connecticut Angler's Guide (Weston, 1992). As part of the CTDEP effort to restore Atlantic salmon to the Connecticut River and selected tributaries, juvenile salmon are released into the Farmington River each spring (Weston, 1992).

3.7 ECOLOGY/ENDANGERED SPECIES

A request to identify endangered, threatened, and special concern species was submitted to the U.S. Fish and Wildlife Service (USFWS) and the CTDEP in August 2007.

The USFWS responded in a letter dated September 5, 2007 indicating that the dwarf wedgemussel (*Alasmidonta heterodon*) has been identified in the Farmington River upstream of the Site. No other federally-listed or proposed threatened or endangered species or critical habitat was known to occur in the project area.

The CTDEP responded in letters dated December 18, 2007 and December 31, 2007 indicating that the Eastern box turtle (*Terrapene carolina*) and the Eastern pond mussel (*Ligumia nasuta*) occur in the vicinity of project area.

Also, based on information obtained from the State of Connecticut Department of Environmental Protection, the Farmington River has been stocked with Atlantic salmon (*Salmo salar*) as part of the State and Federal Atlantic salmon restoration effort.

4.0 RADIOLOGICAL STATUS OF THE FACILITY

This section provides a description of the radiological status of the facility. The descriptions in this section are summarized from the same or equivalent information presented in the Final Historical Site Assessment (HSA) report previously prepared for the Site (Harding ESE, 2002). The HSA assessed the potential radiological impacts of historic operations at the Site. The purpose of the HSA is to provide a complete history of Site activities that may have resulted in the release of licensed material. The HSA was prepared in accordance with the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NUREG-1575) to address the following:

- Identify the potential, likely, and known sources of radioactive material and radioactive contamination based on existing or derived information;
- Differentiate impacted from non-impacted areas;
- Identify whether or not areas pose a threat to human health and the environment;
- Provide an assessment for the likelihood of contaminant migration; and
- Provide input to scoping and characterization surveys.

Under the MARSSIM, areas that have no reasonable potential for residual contamination do not need any level of survey coverage and are designated as non-impacted. Figure 4-11 identifies the impacted and non-impacted areas of the Site. A summary of the radiologically impacted areas of the Site is included in Table 4-6.

The following descriptions include summaries of the types and extent of radioactive material contamination in media at the Site: buildings, systems and equipment, surface and subsurface soil, and surface and groundwater.

4.1 CONTAMINATED STRUCTURES

The construction of the CE Windsor Site began in 1956 with CE's participation in contracts with the AEC. The AEC and US Navy were involved in a combined effort (Navy Nuclear Propulsion Program) to build a nuclear powered surface and submarine fleet.

The original buildings that were constructed to participate directly in the AEC contract activities include Buildings 1, 2, 3, 5, 6, and 6A. A number of other facilities were also constructed to support the operations performed in these buildings such as the WWTP, offices (Building 4) and the boiler house (Building 7).

From 1956 to 1961 the Site was used for AEC/US Navy fuel manufacturing, research and development and training activities. Between 1961 and 1993, CE was licensed by the AEC/NRC for commercial nuclear fuel manufacturing work.

The commercial work consisted of nuclear fuel manufacturing, reactor component testing and refurbishment, and research and development projects. The buildings involved in these activities included Buildings 1, 1A, 2, 2A, 5, 6, 16, 17, 18, and 21.

Remaining structures that have been or may be radiologically contaminated by the historic commercial operations at the CE Windsor Site are described in this section. The CE Windsor Site Commercial Area included Building 2 Complex (Buildings 1, 1A, 2, and 2A and the surrounding grounds), Building 5 Complex (Buildings 5, 15, 16 and 18 and the surrounding grounds), Building 6A Complex and Building 17 Complex. A complete list of structures is presented in Figure 3-2. The building complexes include all underground structures and the surrounding grounds, as shown in Figure 4-11.

Decontamination and dismantlement of interior systems and components in Building Complexes 2, 5, and Building 17 was authorized by License Amendment No. 50, and work is complete. License Amendment No. 53 authorized the decontamination and deconstruction of the above-grade structures of buildings in Building Complexes 2, 5, and Building 17 and this work is complete.

All other buildings existing on-site are considered non-impacted. Based on a review of available files, interviews with former employees, and historical records, no radiological materials were ever associated with Buildings 3B, 3C, 4, 7, 8, 9, 12, 13A, 14, 14A, 19, 20, 22, 23, and 24.

4.1.1 Building 2 Complex (Deleted – Revision 1)

4.1.2 Building 5 Complex (Deleted – Revision 1)

4.1.3 Building 17 Complex (Deleted – Revision 1)

4.1.4 Building 6A Complex (Deleted – Revision 1)

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DECOMMISSIONING COMPLETE

4.1.5 Building 3 Complex

The Building 3 Complex includes Buildings 3, 3A, 3B and 3C and is located in the southern portion of the Site (Figure 3-2). In 1956, CE entered into the nuclear fuel fabrication field and constructed Building 3 for the fabrication of nuclear fuel. Building 3 was used for US Navy nuclear fuel fabrication until 1961. After 1961, Building 3 was decontaminated and used for fossil fuel research and development. Building 3A was constructed in 1962 as office space to support fossil fuel research in the renovated Building 3. Radiological investigations and characterization of Building 3A were performed and no contamination was identified. Since Building 3A was not contaminated, it was dismantled down to the slab in 2002. Building 3B was built in 1995 for support of fossil fuel research. Building 3C was built in 1998 as a utility support facility for fossil fuel research. Buildings 3B and 3C are not contaminated.

Physical Description.

Building 3 is a 48,200 square feet (ft^2) one-story structure constructed of concrete block, concrete floors and steel framing with Transite siding, and a steel roof deck. The original building design consisted of a 12,200 ft^2 fuel fabrication "hot shop" at the north end of the building. The remainder of Building 3 consisted of the South or Cold Fabrication Shop which was adjacent to the hot shop and approximately 33,000 ft^2, which was divided into equipment and laboratory areas. Building 3 also housed an assembly area known as the Core Assembly Building, approximately 3,600 ft^2, with 50-ton and 10-ton overhead cranes. Layout of Building 3 is provided in Figure 4-12.

Historical Use.

From 1955 to 1961 multiple contracts were issued by the AEC for the design, development, fabrication, and rework of nuclear fuel components. In 1956, CE entered into the nuclear fuel fabrication field and constructed and operated Building 3. Nuclear fuel components fabricated in Building 3 included subassemblies, cores, and fuel plates. The nuclear operations ceased in 1961.

After 1961, Building 3 was used for fossil fuel research. There has been only limited radiological work in the building since 1961. From approximately 1970 to 1985, x-ray diffraction work was occasionally conducted in Building 3 on the components being tested in Building 5.

Radiological History and Potential Contaminants

The fuel manufacturing process consisted of a series of complex operations including melting, forging, rolling, shearing, vapor blasting, pickling and welding of uranium and its alloys. These operations were performed primarily in the north end of the building. These operations generated large quantities of radioactive waste containing uranium. Some of the operations, including forging and rolling of enriched uranium at an elevated temperature, included large quantities of liquids (salt baths) to provide lubrication and prevent oxidation, mineral oil for cutting, acids for pickling and wet vapor blasting with aluminum oxide (Harding ESE, 2002). The liquid waste was discharged to Building 6 for analysis and dilution prior to release via the industrial waste line to the Site brook. Radiological contamination was also released to the ventilation system and may have been discharged from the building through the ventilation system and via the doorways, which were reportedly left open for cooling the building (Harding ESE, 2002).

Nuclear fuel fabrication processes occurring in Building 3 began in the north end of the building within the hot shop, consequently resulting in the highest levels of contamination. After Building 3 began manufacturing operations, CE constructed a wall to keep the dust from the grit blasting and other “hot” work in the northern end of the building. During operations, uranium dust was present on ventilation units, beams, ceiling, and under the paint on the floors throughout the hot shop. The manufacturing area was also equipped with a ventilation system designed to ensure the proper direction of air flow and a slight negative pressure in the hot shop. Building 3 contained a water supply and drainage system for controlling process waters, industrial waste, sanitary waste and radiological waste.

The contamination in Building 3 further decreased at the South Fabrication Shop. The South Fabrication Shop contained several pits, one with a large pickling tank, and several smaller pickling pits. In 1996, a reboiler water leak resulted in a partial flooding of the southwest corner of Building 3 and the pit in the reboiler room. Water spread throughout the southwest corner of Building 3, including two posted contaminated areas, and out of the building under the south roll-up door. Samples were collected and the water was removed, all affected areas were released (Harding ESE, 2002).

The Core Assembly Building located on the south end of the building was intended to be maintained radiologically clean. However, there were times when final assemblies contained residual uranium, and had to be cleaned before being released from the building (Harding ESE, 2002). Products that left the Core Assembly Building went into the field (S1C, US Navy).

In 1961, nuclear operations in Building 3 ceased, and CE decontaminated Building 3. The highest levels of contamination existed in the north end of the building (hot shop) with decreasing levels trending southward in the Core Assembly Building and with further decreased levels of contamination in the Cold Fabrication Shop. Original decontamination efforts included the core assembly area, northern wall, and over-head areas.

Several additions were made to Building 3 after nuclear fuel manufacturing operations ceased. Several storage areas were added along the east side of the building. In addition, the footprint of the high bay was doubled in the early 1970s to support a fossil power plant safety valve testing program. These areas are indicated on Figure 4-13. The soil under this addition was surveyed prior to construction and was clean of radiological contamination from the plant operations.

Further decontamination has occurred in other areas within Building 3. The pits located at the south end of Building 3 have all been decontaminated; however, there is no available information on any subfloor sampling data relating to the pits. The ventilation system was removed from the hot shop in 1972. The roof of Building 3 was replaced in the late 1960s and again in the late 1970s.

A metal shear that is stored within Building 3 is known to have elevated levels of residual radioactivity. Additional materials may be identified during removal of fixed laboratory equipment (benches, hoods, etc.) or ventilation equipment.

The radiological history and known or potential radiological contaminants for each room or area associated with the building are summarized in Table 4-7.

Previous Radiological Investigations.

Various characterization surveys of Building 3 have been performed, including one by USACE (SAIC, 1999). Since Building 3 had been used for uranium and not byproduct materials, characterization surveys were only performed for alpha contamination. There are several localized areas of elevated residual radioactivity within the building. These include the locker room at the northeast end of the building, the records storage room (vault) adjacent to the locker room, tool storage room (north bay) and the materials lab in the northwest corner of the building. Fixed contamination levels ranging from 3,000 to 124,000 dpm/100 cm^2 alpha have been measured. Overhead surfaces in the north bay have elevated fixed contamination readings up to 9,000 dpm/100 cm^2 alpha. In addition, the north end of the building (hot shop) has contamination in the building materials (floor, walls (paint), overhead pipe insulation, structural beams, ceiling). Contamination levels of the walls (paint), overhead pipe insulation and structural beams were in the 14 to 1,423 pCi/g range. Concentrations of uranium in surface level concrete of the floor in Building 3 ranges from 0.7 to 1,152 pCi/g and the distribution is shown in Figure 4-14. Concrete cores of the slab have uranium concentrations ranging from 0.34 to 23 pCi/g as shown in Figure 4-15. Volumetric samples of the roof have also been collected with results from 0.07 to 2.29 pCi/g as displayed in Figure 4-16.

4.1.6 Building 6 Complex

Building 6 is located near the southern portion of the Site, immediately north of Building 3 (Figure 3-2). Building 6 was a liquid radiological waste dilution and pumping facility. Liquid radiological wastes containing all enrichment levels of uranium processed at the Site as well as byproduct materials were processed in Building 6 prior to being released to Site brook.

Physical Description.

Building 6 is a two level cast-in-place concrete structure with a steel roof deck . The Building 6 footprint is approximately 2,750 ft^2. The building houses ten 2,000-gallon steel storage tanks on the basement level (Figure 4-17), four 5,000-gallon steel dilution tanks on the ground level (Figure 4-18), and there is a shallow sump located in the southwest corner of the lower level.

The tank system is designed to use two of the 5,000-gallon dilution tanks with each set of five 2,000-gallon tanks. The two systems, although interconnected, are referred to as the A and B waste storage systems. The storage tanks are fitted with sampling pumps.

Building 6 is equipped with ventilation and radiological waste systems. The ventilation unit is mounted on the roof and provides ventilation for the 2,000-gallon storage tanks. The building was designed so that the radiological waste line enters the building and the industrial waste line exits the building. The radiological and industrial lines have been disconnected and sealed.

Historical Use.

The building was constructed as a liquid radiological waste collection and dilution facility for Building 3. In 1960 the liquid radiological waste from the Building 5 Complex was connected to Building 6. Building 3 was no longer generating liquid radiological waste after 1961 and eventually was disconnected from Building 6. In 1968 the liquid radiological waste from Building 17 was also connected to Building 6. The liquid radiological waste was sampled, diluted if necessary, and discharged to Site brook via the industrial waste lines.

Liquid discharges from Building 6 to the Site brook were terminated in the 1990s. After that, liquid effluent waste was filtered and processed via an evaporator located on the first floor of Building 6. The evaporator was removed from Building 6 in 2001. The radiological and industrial lines in Building 6 were disconnected and sealed. Additional information regarding the radiological and industrial lines is provided in Section 4.2.

Radiological History and Potential Contaminants

Building 6 functioned as a radioactive waste collection, monitoring, and dilution facility for fuel fabrication and laboratory operations from approximately 1956 until 1995.

Liquid waste was generated from the handling of uranium in conjunction with the fuel manufacturing programs and byproduct materials under both AEC and commercial nuclear power support work. Liquid waste included floor mop water, cleanup water, and effluents from sinks, toilets and showers in contaminated or potentially contaminated areas. Liquid waste was also generated from wet chemical analyses, and cleaning of glassware (Harding ESE, 2002).

The waste was then stored in ten 2,000-gallon tanks located in the building. The tanks were connected in series in the hot waste dilution vaults. Wastewater from the radiologically contaminated processes generally went directly to the hot waste tanks. After the waste was sampled and analyzed, if it met the release criteria, it was discharged to the Site brook. If the waste did not meet the release criteria, it was transferred to one of four 5,000-gallon dilution tanks located in Building 6 prior to being released into Site brook.

The storage tanks were periodically cleaned of sludge. The tanks were last emptied in 1993. The radiological history and known or potential radiological contaminants for each room or area associated with the building are summarized in Table 4-8.

Previous Radiological Investigations.

Various characterization surveys of Building 6 have been performed, including one by USACE (ENSR, 2001). Since Building 6 handled liquid waste from the entire Site, both uranium and byproduct materials, characterization surveys were performed for alpha and beta contamination. Surveys and volumetric sampling of the roof of Building 6 have not identified elevated levels of residual radioactive materials. Measurements of fixed and removable contamination are shown in Figure 4-19. No measurements of fixed contamination were greater than 5,000 dpm/100 cm^2 beta and no measurements of removable contamination exceed 1,000 dpm/100 cm^2 beta. Analysis of volumetric samples of the roofing material ranged from 0.55 to 1.1 pCi/g total uranium (Figure 4-20) with similar results for the concrete beneath the roofing material (Figure 4-21).

The ground floor level of Building 6 has fixed contamination measurements less than 5,000 dpm/100 cm^2 beta and removable contamination levels less than 1,000 dpm/100 cm^2 beta as displayed in Figure 4-22. Two concrete core samples of the floor were collected with total uranium results less than 1.3 pCi/g. Sediment samples were collected from two of the dilution tanks with maximum concentrations of 30.9 pCi/g Co-60 and 9,270 pCi/g total uranium as shown in Figure 4-23.

The basement level of Building 6 has fixed contamination measurements greater than 5,000 dpm/100 cm^2 beta on the floor in the vicinity of the storage tanks as displayed in Figure 4-24. Only one location had a removable contamination result greater than 1,000 dpm/cm^2 beta as shown in Figure 4-25. Surface samples of concrete from the floor and walls showed no significant concentrations of Co-60. Total uranium concentrations ranging from 9 to 439 pCi/g as displayed in Figure 4-26. Concrete core samples collected from the floor of the basement level also had no significant concentrations of Co-60 and total uranium concentrations ranging from 1.6 to 11.6 pCi/g as shown in Figure 4-27. Samples from paint on the exterior of the storage tanks indicated minimal concentrations of Co-60 and total uranium concentrations from 265 to 1239 pCi/g as shown in Figure 4-28. Sediment samples from inside the storage tanks have Co-60 concentrations from 1.9 to 56.3 pCi/g and total uranium concentrations ranging from 1,161 to 9,879 pCi/g as displayed in Figure 4-29. The sump in the southwest corner of the basement has sediment with 8.9 pCi/g of Co-60 and 5,483 pCi/g of total uranium.

4.2 CONTAMINATED SYSTEMS AND EQUIPMENT

This section presents an historical account and a summary of investigative activities for the systems associated with the CE Windsor Site. These systems were divided into the following categories:

- Industrial and radiological waste lines;
- sanitary waste lines;
- storm water lines; and
- underground utilities.

4.2.1 Industrial and Radiological Waste Lines

The industrial waste lines received both chemical and radiological waste from the facility buildings in the main campus area and discharged the waste to outfalls at Site brook. The Industrial Waste Lines immediately outside of the Building Complexes 2, 5, 6A, and 17, as shown on Figure 4-8, have been remediated under the Commercial D&D program. The waste lines associated with Building Complexes 3, 6, and the lines that run from Building 6 to the discharge at the Site brook will be remediated under Revision 1 of the DP. This is also depicted by area in Figure 4-11.

In the southern portion of the Site, the radiological wastes were disposed of through radiological or 'hot' waste lines and chemical wastes were disposed of through separate 'cold' lines. The 'hot' lines went to Buildings 6 and 6A for dilution, as needed, and discharged to the 'cold' line, located just north (downstream) of Building 6, and the liquid wastes then discharged to Site brook. After 1961 Building 6A was renovated for the maintenance department and all 'hot' waste went to Building 6 for dilution, and then to the Site brook. Figure 4-8 shows the schematic of the industrial waste lines for the CE Windsor Site.

4.2.1.1 Background

In the mid to late 1950s, during AEC operations and prior to licensed operations, radiological wastewater was generated at Buildings 3 and 5. All radiological wastewater from Building 5 flowed to dilution tanks located in the basement of Building 6A. Similarly, radiological wastewater from Building 3 flowed to similar tanks located in Building 6. Both of these lines were constructed of dur-iron pipe with the sections between manhole #6 and Building 6 constructed of concrete tile.

In 1960, the radiological waste system was re-designed and all radiological waste flowed to dilution tanks in Building 6. Building 6A was then renovated to house the Facilities Services and Engineering Department.

In December 1961, CE was no longer manufacturing nuclear fuel under AEC contracts, and an AEC license was requested to manufacture commercial fuel. Non-radiological wastewater or chemical wastewater (cold lines) was disposed of through separate industrial waste lines in the southern portion of the Site, which joined the discharge from Building 6, in which radioactive waste water had been diluted to appropriate concentrations. These formed the main industrial waste lines that ran from Building 6 to the Site brook.

In the early 1970s, it was noted that the small quantities of wastewater released to the industrial waste line were not discharging at the outfalls located in Site brook. Assuming that the acidity of the industrial wastes had potentially disintegrated a portion of the pipes, CE constructed a new line from Building 6 to the outfalls. The original line was abandoned in place, and the new line, constructed of polyvinyl chloride (PVC), was installed directly adjacent to the failed line. A new outfall was also installed that discharged to the Site brook approximately 100 feet upstream from the previous outfall. No investigation was performed to locate or characterize the potential leak.

The new industrial waste lines included two pH adjustment tanks that were located east of the WWTP (Figure 4-8). These tanks processed between 80,000 and 250,000 gallons of non-contact cooling water and process water per day. The industrial wastewater was continually monitored for pH. Acids or bases were added to neutralize the waste so that the pH was within the National Pollution Discharge Elimination System (NPDES) discharge permit limit (i.e. pH of 6 to 9). The neutralized solution was then discharged to the Site brook. No other analysis or treatment was performed on the waste stream.

4.2.1.2 Evaporator Line Investigation

After 1992, CE no longer held a NPDES permit for the discharge of wastewater to surface waters. The chemical industrial wastes were discharged to MDC and the radioactive wastewater was evaporated on-site to comply with these changes. CE installed an evaporator in Building 6 and used the former radiological waste lines that ran from Building 5 to Building 6 to transport the wastewater. In 1998, CE terminated use of this line. CE then initiated the Evaporator Line Investigation (ELI). Most of the radiological waste line associated with the evaporator has been removed during the prior decommissioning work of the Commercial D&D areas, A segment of this line remains (about 135 feet in length) that will be removed during decommissioning of the remaining impacted portions of the Site.

4.2.1.3 Additional Investigations

During a subsequent field program, CE installed monitoring wells adjacent to the manholes along the remaining portion of the industrial waste line (north of Building 6) and encircling Building 5. No soil or groundwater samples collected during this effort contained radiological materials above background. Subsequent sampling also indicated the lack of byproduct materials in the groundwater in wells adjacent to the industrial waste line.

Sections of the industrial waste line at the Site except for those around Buildings 3 and 6 and the pipeline that runs from Building 6 to Site Brook have been removed during previous decommissioning activities. Figure 4-30 shows the remaining waste lines associated with Buildings 3 and 6. These lines are associated with floor drains and laboratory drains in Building 3 and running from Building 3 to Building 6. Several modifications to these lines have occurred since original construction, and this figure represents a compilation of the best information available. There are tanks and piping associated with the industrial waste line inside Building 6 as previously described in Section 4.1.6.

A summary of the characterization data for the waste lines and surrounding soil is presented in Figure 4-31. Sediment samples from manholes with elevated levels of total uranium are located near Building 6, at the beginning of the pipeline. Concentrations of total uranium in these manholes range from 4,100 pCi/g to 97,000 pCi/g. Soil samples collected alongside the waste lines have not identified elevated concentrations of Co-60 or total uranium. Concentrations in the waste lines around Buildings 3 and 6 can be estimated from floor drain sediment samples collected in Building 3. There are no significant concentrations of Co-60 and total uranium ranges from 1 pCi/g in the south end of the building to 2,043 pCi/g at the north end as shown in Figure 4-32. Analysis of sediments inside the industrial and hot waste lines on the west side of Building 3 during maintenance activities in the 1980s, indicated total uranium concentrations of up to 67,000 pCi/g (CE, 1985).

4.2.2 Sanitary Waste Lines

The sanitary waste system includes the underground sanitary lines, the WWTP, the former leach fields, and the remaining leach field that received sanitary waste from the Health Works facility (or the former Process Development Unit (PDU) Building) (Figure 4-8). The WWTP operated from 1956 through 1992. On December 8, 1992 the Site wastewater was routed to the MDC.

<u>Sanitary Lines and WWTP</u>

The original sanitary lines and the WWTP were constructed in 1956 to support the Site operations. The WWTP was removed in 2001. The sanitary lines from near Building 6 to the former WWTP run parallel to the former and newer industrial waste lines (Figure 4-8).

The WWTP operated from 1956 through 1992. On December 8, 1992 the Site wastewater was routed to the MDC. The WWTP was demolished in 2001, although some underground structure and piping exists in the vicinity of the industrial waste line. These will be removed as part of decommissioning activities in the remaining impacted portions of the Site.

4.2.2.1 Previous Characterization Studies

During the decontamination of Building 17, radiological contamination was identified in the sanitary lines that ran from Building 17 to Manhole #4. This information prompted CE to investigate the potential presence of radiological contamination in all of the sanitary lines on-site. In 1994, CE conducted a radiological survey of the liquid and sludge within the sanitary line manholes. The results of this survey indicated the presence of low levels of uranium activity. After this discovery, daily liquid effluent samples were obtained from the last manhole (M-0) on ABB property before leaving Site from November 11, 1994 through April 5, 1995 and then weekly from April 5, 1995 through May 9, 1996. Additional liquid and sludge samples were also obtained by the MDC. These samples did not contain any uranium isotopes above the minimum detectable activity (MDA).

Concurrent with the weekly monitoring, CE evaluated Site buildings to determine if historic operations could still be contributing to the contamination. As a result of this evaluation, the following corrective actions were then taken to eliminate possible sources of radioactive contamination:

- termination of all water discharges (with the exception of fire protection) from Building 17;
- plugging the sanitary sewer lines from Building 17;
- abandoning all bathrooms in Building 3 that discharged to the main system (with the exception of the northeast bathroom where new lines were installed for the few fixtures to remain in service);
- installing a new sewer discharge line for Building 6A; and
- historical review and sampling for radiological materials of the Building 2, 3, and 5 complexes.

Based on the evaluation of radioactivity in the sludge from the sanitary manholes, uranium concentrations exceeded CE's protocol guidelines (of 30 pCi/g) at four manholes, S-15, S-16, S-17, and S-19. At least one sample from each of these manholes indicated uranium concentrations exceeded 40 pCi/g with a maximum concentration of 400 pCi/g. Each of these manholes were located in the area between Buildings 3, 5, 6, 6A and 17. All other sediment samples from the remaining manholes did not contain a measurable amount of uranium or concentrations were less than 18 pCi/g.

The four contaminated manholes were remediated in 1995. After the remediation, two of the four manholes (S-17 and S-19) were identified for post-remediation monitoring. The two other manholes (S-15 and S-16) did not require post-remediation monitoring because they no longer received wastewater, and were isolated from the rest of the system.

The two manhole locations that were included in the post-remediation monitoring program (S-17 and S-19) became re-contaminated due to additional sediments within the lines. These sediments were noted as inaccessible during the remediation. After the majority of the sediment was washed out, the manholes were decontaminated again and in 1996 all manhole samples were observed below the levels specified in the CE Protocol of 30 pCi/g.

In April 1995, CE calculated a conservative estimate of the total uranium content in the sanitary sewer system of 3,876 pCi. CE estimated that there is a total of 106 pCi of total uranium in the manholes and a conservative estimate of 3,770 pCi of total uranium in the sanitary sewer lines. This suggests that residual contamination may remain within the lines.

4.2.2.2 Source Evaluation

Based on the information gathered as part of this Sanitary Line remediation, the following potential uranium sources affecting the sanitary waste lines were identified:

- Building 2 complex (Buildings 1, 1A,and 2A) due to historical AEC fuel operation and subsequent nuclear power plant outage support operations;
- Building 5 complex (Buildings 5, 16, and 18) due to the historical commercial and AEC fuel development activities; and
- Building 17 complex due to the historical commercial fuel manufacturing operations and historic records of contamination found in the restrooms.
- Building 3 complex due to historical fuel manufacturing operations and historic records of contamination found in the restrooms.

Systems from Building Complexes 2, 5, 6A and 17 were removed as part of prior decommissioning activities associated with the Commercial D&D areas.

Leach Fields

As mentioned above, two leach fields were constructed at the CE Windsor Site. Both leach fields are shown on Figure 4-8. One leach field received sanitary waste from the Building 2 complex from 1956 through the mid-1970s. The original system discharged into a common septic tank and siphon chamber located to the west of the complex. Liquids from the siphon chamber were then discharged to a tile leach field approximately 100 yards northwest (see Figure 4-8). The leach field was located under the former zirconium and thorium-magnesium burning grounds. This area was also used as a storage area for 55-gallon drums containing radiologically contaminated soil and PPE. CE remediated the Burning Grounds in 1986 for radiological contamination. In November 1999, CE then remediated the leach field of chemical contamination. The leach field has been remediated in accordance with NRC, USEPA, and CTDEP regulations.

The second leach field that is still present on-site received sanitary waste from the former John B. Faucette Wellness Center also known as Health Works. This facility was also the former PDU, and was a test coal gasification plant. Recently this facility housed a radiological sample analysis laboratory during the prior decommissioning activities. Only small quantities of radioactive material in the form of sealed sources, wipes, volumetric debris or soil samples were brought into controlled portions of the building. No contamination was identified in this facility during use for radiological analysis or after it was vacated. Therefore, radiological contamination in the leach field is not expected to be present.

4.2.2.3 Additional Investigations

Sections of the original sanitary waste lines at the Site except for those around Buildings 3 and 6 and the pipeline to the former WWTP have been removed during previous decommissioning activities. Figure 4-30 shows the remaining sanitary lines associated with Buildings 3 and 6. These lines are associated with 2 locker rooms in Building 3.

A summary of the characterization data for the waste lines and surrounding soil is presented in Figure 4-31. Sediment samples from manholes showed no significant concentrations of Co-60 and total uranium concentrations were less than 3 pCi/g. Soil samples collected alongside the waste lines have not identified elevated concentrations of Co-60 or total uranium. No significant quantities of uranium or Co-60 were discovered during removal of the sanitary lines in the Commercial D&D areas and it is anticipated that similar results will occur during removal of the remaining lines.

4.2.3 Storm Water Lines

Several storm drainage lines with associated outfalls are located throughout the CE Windsor Site and are shown on Figure 4-9. The various separate drainage networks are identified as follows:

Storm drains servicing most of the industrial, developed southern part of the Site flow to a feeder line west of and parallel to East Main Street. This line turns eastward running to the north of Building 3 and discharges near the westernmost edge of Small Pond. This is the principal drainage network present at the Site.

The storm water lines at the Site currently only receive runoff during rain events. The storm drains are separate from the industrial waste lines that received industrial and radiological wastewater from many of the Site building (e.g., laboratory sink drains in Building 3, etc.).

Soil samples were also collected at ten outfall locations. Additional soil samples were later collected from all of the manholes designated by CE's radiological safety officer as having a potential for containing radiological materials. No soil samples collected from the storm water outfalls or manholes contained radiological materials above background.

CE has also completed the excavation and re-construction of two infiltration basins located north and west of Building 3. Samples were collected prior to the excavation to delineate the chemical characterization of each drainage basin. Soils were then excavated and confirmation samples were collected to confirm that all soils containing chemical constituent concentrations above the CTDEP RSR Criteria were removed. During these removal efforts, soil samples were screened by CE for radiological materials. No soils exceeded background activity.

Sections of the original storm water lines at the Site except for those around Buildings 3 and 6 have been removed during previous decommissioning activities. Figure 4-30 shows the remaining storm lines associated with Buildings 3 and 6.

4.2.4 Underground Utilities

The underground utilities at the Site provide a potential migration pathway for radiological or chemical contamination. Groundwater contamination has been found at concentrations above background levels at two locations: MW-1201 and WP-1403S. MW-1201 is located adjacent to manhole #5 of the industrial waste lines and WP-1403S is located just outside the original industrial waste outfall in Site brook. Both locations are in the vicinity of the Industrial Waste Lines, or its outfall, and these are the most likely source of the contamination. The Industrial Waste Lines and the impacted sediments in the Site brook are to be removed as part of the Site remediation.

4.3 SURFACE AND SUBSURFACE SOIL CONTAMINATION

CE has previously performed surveys to characterize the Site, including the Commercial D&D Areas. In addition, the areas previously designated as FUSRAP have been extensively investigated (ORISE, 1994; ORISE, 1996; SAIC, 1998; SAIC, 1999; ENSR, 2001). These surveys included the surface and subsurface soils in the subject areas. The results of the surveys indicated the presence of radiological contamination in the surface soils and the potential for contamination in some subsurface soils. The ongoing decommissioning effort includes further characterization of the soils. Soils will be remediated to the soil release criteria developed as described in Section 5 of this report.

4.3.1 Woods Area

The Woods Area is located west of East Main Street and the former Building 2 Complex (Figure 4-11) and straddles the access road that runs northwest from former Building 2. The area is approximately 7 acres in size. Two former storage areas, the RCRA Greater Than 90 Day Storage Area (GT-90) and the Waste Pad Area, are located alongside the access road and within the overall confines of the Woods Area. This area was used from 1956 through 1960 to dispose of miscellaneous waste material including piping, PPE and soils, mostly contained in 55-galllon drums.

In addition to historical knowledge, interviews conducted during the HSA (Harding ESE, 2002) confirmed that radiologically impacted soils and personal protective equipment (PPE) were stored in this area before 1961 and then again during the mid-1980s. Additionally, radiologically contaminated drums of PPE were stored in the late 1950s. Over time, rainwater infiltrated into the drums. The drums were then punctured to release the water and the PPE was burned to reduce the volume (Harding ESE, 2002). The waste burning was terminated by 1961, however this area was still used as a storage area for radiologically contaminated waste generated during several remediations at the Site.

Historical aerial photographs show drums staged on the storage pads as well as alongside the Woods Road where no fencing or visible secondary containment was apparent. Site inventory records from 1971 document the storage of 77 drums of radioactive materials at the Waste Pad area. It was historically used for temporary storage of sealed LLRW containers although other potential non-radiological COCs such as waste oils and solvents were also stored in that area. The GT-90 was used for storage of drummed chemical wastes. No LLRW was stored at the GT-90 (Harding ESE, 2002). However the soils beneath the asphalt pad may be radiologically contaminated from prior stored wastes. The GT-90 had a capacity to store approximately 150 drums.

Based on previous investigations, the surface soils on both sides of the access road contain radiological residuals above background levels. Recent data determined that uranium was present at activities above background at deeper locations. The recent data indicates elevated uranium activity to depths of approximately 3.5 feet at locations immediately adjacent to the road. This data is discussed further below.

Several investigations have been performed to characterize the extent of radiological contamination in the surface soils. CE conducted an extensive soil sampling effort in 1989. The highest activity levels observed were near the Waste Pad Area. In an investigation by ORISE, 30 areas were gridded around and within the woods area in the vicinity of the waste pad. Twelve of the blocks were then randomly selected and sampled (ORISE, 1994). A gamma walkover survey conducted by USACE identified elevated radioactivity in the same areas surrounding the waste storage pad, within the Woods Area (SAIC, 1998).

The most recent investigation was completed in 2000 by ENSR, under contract to USACE (ENSR, 2001). In this program the woods area was gridded off and samples collected for both gamma spectroscopy, with selected samples submitted for alpha spectroscopy. The resulting data supported the earlier sampling efforts. The areas of elevated uranium activity are located in areas adjacent to the road.

Detailed gamma walkover surveys were performed in this area and the results are shown in Figure 4-33. Most of this area had relatively low readings with elevated readings adjacent to the road and waste pad. Elevated readings with a maximum value of 819,754 CPM were found at a location adjacent to the west side of the waste pad. Soil data was compiled for this area with Co-60 results less than minimum detectable concentrations or just above and total uranium results from background to a maximum of 110,236 pCi/g. Locations of the samples and total uranium results are shown in Figure 4-34. Note that the soil and gamma walkover survey data have strong correlation for areas of elevated contamination.

4.3.2 Drum Burial Pit

The Drum Burial Pit is located west of the Woods Area in the northern portion of the Site (Figure 4-11). The area is approximately 1 acre in size. This area was used from 1956 through 1960 to dispose of miscellaneous waste material including piping, PPE and soils, mostly contained in 55-galllon drums that are now rusted and/or crushed. The drums eventually decayed and the adjacent soils were pushed over the waste subsequently burying the drums in place.

In the 1980s, CE employees noted the presence of partially buried drums visible near an access road. In 1990, CE excavated a small portion of the waste material. The excavated wastes contained uranium residues. This waste was repackaged at the waste pad (in the Woods Area). The material encountered in this excavation included: small piping, electrical wiring, plastics, considerable soils and sand, tools, vermiculate, fiberglass insulation, paint cans, conduit, chicken wire, PPE and asbestos. The repackaged wastes were eventually disposed.

In 1993, USDOE performed a designation survey that included the collection and analysis of soil samples from the drum burial pit (ORISE, 1994). In 2000, ENSR collected additional data from around and with in the drum burial pit (ENSR, 2001). Very few samples were collected directly from materials within and around the crushed drums, and consequently the data does not reflect the elevated levels of activity detected in earlier investigations.

During the 1990 excavation, gamma readings at the base of the excavation ranged from 0.1 to 2 milliRoentgen/hour (mR/hr). Surveys on the exterior surfaces of the excavated barrels ranged from 2,200 CPM to 400,000 CPM. Detailed gamma walkover surveys were performed in this area and the results are shown in Figure 4-35. Most of this area had relatively low readings with elevated readings adjacent to the road. Elevated readings had a maximum value of 30,670 CPM. Soil data was compiled for this area with Co-60 results less than minimum detectable concentrations or just above and total uranium results from background to a maximum of 16,000 pCi/g. Locations of the samples and total uranium results are shown in Figure 4-36.

4.3.3 Clamshell Pile

The Clamshell Waste Pile is located in a shallow swale approximately 600 feet north of the Site brook in the northwestern portion of the property (Figure 4-11). This area is approximately 15 feet wide, by 30 feet long and 6 feet deep, and fills a natural gully. The Clamshell Waste Pile was identified during a walkover of the northern property in March 1999.

In the late 1950s clamshells were used to buffer the pH concentration of the Site brook near the industrial waste outfalls. Because the Site brook received industrial wastewater including low level radioactive wastewater, the shells absorbed some amount of uranium and presently contain radioactive materials. No information is available as to when the shells were removed from the brook or why they were deposited in the woods.

CE conducted the first sampling of the Clamshell Pile material in September 1999. Results from that sampling confirmed the presence of radioactive materials. Additional samples were then collected to better characterize both the chemicals and radioactive materials present. In December 1999, a survey and sampling program of both the clamshells and the surrounding area was conducted.

The results of the sampling program indicated that the clamshells contained elevated concentrations of uranium, with a maximum concentration of 1,392 pCi/g. The results are shown in Figure 4-37.

4.3.4 Equipment Storage Yard

The Equipment Storage Yard is located on the western side of Small Pond, and northeast of Building 3 (Figure 4-11). The area is approximately 0.2 acre in size. This area was originally used in the mid to late 1950s as a disposal area for miscellaneous fill and construction debris. It has been referred to in historic memos as “the dump northeast of Building 3” (Harding ESE, 2002). Later in 1968, Building 20 was constructed in the area and used as a test facility for trash incineration. In 1969, Building 20 was turned over to the Facility and Engineering Services Department for use in their daily maintenance operations.

Building 20 was demolished in the fall of 2001. The underlying soils in the direct vicinity of the building have been identified as native materials, and are not associated with the filling operations in the Equipment Storage Yard. No soils associated with the demolition, nor the building materials contained radiological materials above background.

MDC topographic maps from 1961 show the area behind and east of the former Building 20 as being filled and not representative of the natural land surface. This is consistent with aerial photos from 1957, 1960, 1963, 1973 and 1985 that indicate this area was disturbed and built up with miscellaneous fill. The materials identified in several test pits support these findings. Two excavated soda bottles dated 1957 and 1959 were identified in test pits.

After the area was filled to the current grade, the yard was still used by the maintenance department to stock landscaping supplies (mulch, clean fill, and brush debris), non-hazardous debris, and miscellaneous material (e.g. a metal recycling roll-off container, snow plows, roadway signs, etc.) until mid-1999.

Investigations of the Equipment Storage Yard have identified three areas with radiological impacts. Two of the areas were associated with drums that were located on the southern edge of the yard, near the shoreline of Small Pond. One drum contained byproduct material and was removed by CE. The second drum contained uranium and remains in place. The third area was identified at a test pit. Gamma walkover surveys were performed in this area with a maximum reading of 40,000 CPM. Soil concentrations of total uranium have a maximum of 842 pCi/g. These locations and total uranium results are shown on Figure 4-38.

4.3.5 Buildings 3 and 6 Complexes

Buildings 3 and 6 are located in the southern portion of the Site and were constructed under the initial AEC contracts (Figure 4-11). The Building 3 Complex is approximately 5 acres in size and Building 6 Complex is about 1 acre in size. Nuclear fuel fabrication was conducted in Building 3 prior to 1961. Building 6 was used as a liquid radiological waste processing facility. These two buildings are grouped together in this investigation due to their geographical proximity, original use in the manufacturing of nuclear fuels, and the fact that the area located between the two buildings was used for storage of equipment and waste.

Various investigations have been performed around these buildings by both CE and USACE.

Detailed gamma walkover surveys were performed in this area and the results are shown in Figure 4-39. Most of this area had relatively low readings with a limited number of elevated readings adjacent to the buildings. Elevated readings had a maximum value of 127,363 CPM. Soil data was compiled for this area with Co-60 results less than minimum detectable concentrations or just above and total uranium results from background to a maximum of 3,700 pCi/g. Locations of the samples and total uranium results are shown in Figure 4-40. In addition, soil borings were collected through the slab of both buildings. Soil from below the slab in Building 6 did not have any significant concentrations of uranium or Co-60. However, Building 3 had one location with elevated levels of uranium at 270 pCi/g as shown in Figure 4-41.

4.3.6 Burning Grounds

The Burning Grounds are located north of the Woods Area, down a secondary access road that starts at the GT-90 (Figure 4-11). The area is approximately 2 acres in size. This area is the former zirconium and magnesium thorium burning grounds. The Burning Grounds were used from approximately 1956 to 1961. Zirconium tailings and turnings generated during fuel element assembly processes were transported in drums and burned at this location. By burning the tailings, the zirconium was stabilized and could then either remain in place or be transported off-site. After 1964, zirconium scrap was reportedly no longer burned on-site but was sent directly off-site to be reprocessed. However, some additional burning of miscellaneous materials may have continued through the late-1960s (Harding ESE, 2002).

The magnesium and thorium burning area was co-located with the zirconium burning ground. CE was licensed under the AEC to burn the magnesium and thorium wastes during the late 1950s. During this time, CE also accepted thorium wastes from off-site sources for burning. The burning area consisted of a bermed concrete pad. After burning activities ceased in the early 1960's, the area was used as a storage area for drums of radiological waste.

The burning area was remediated and cleaned of radioactive and hazardous materials in the 1980s in accordance with NRC regulations (ABB-ES, 1998). During this time the drums were repackaged at the Waste Pad (within the Woods Area) prior to off-site disposal. The NRC performed confirmatory surveys (ORISE) which identified areas in excess of the release criteria. The Branch Technical Position on "Disposal or Onsite Storage of Thorium and Uranium Wastes from Past Operations" Option 1 release criteria (NRC, 1981) was used for this area. The maximum concentration for unrestricted use of surface soils was 10 pCi/g for natural thorium. Additional remediation was performed followed by confirmatory surveys (ORISE). The NRC granted unrestricted release of this area in August, 1989 (NRC, 1989).

Data from the final surveys of this area in 1989 combined with some more recent data provide the current characterization data set. During gamma walkover surveys in 2006, several small elevated areas were identified in this area. Elevated readings had a maximum value of 11,000 CPM. Soil samples collected in the elevated areas identified thorium (Th-232) and radium (Ra-226) as the primary radionuclides. Locations with elevated levels of thorium and radium are shown in Figures 4-42 and 4-43. Maximum Th-232 concentration is 8 pCi/g and maximum Ra-226 concentration is 3 pCi/g.

4.3.7 Debris Piles

The Debris Piles are located north of the former WWTP and directly adjacent to Site brook on the south bank (Figure 4-11). The area is approximately 0.5 acre in size. The debris piles include brush; concrete rubble; partially buried drums; and other miscellaneous materials. The area is approximately 30 by 50 feet. Since the Debris Piles are adjacent to Site brook, they are located within the wetlands boundary. Since activities within wetlands require special permits and procedures in order to minimize impact to the ecosystem, additional information regarding this area will be provided in a future revision to the DP.

4.3.8 Site Brook

The Site brook is located in the northern portion of the Site and flows northwest from Goodwin Pond for approximately one half mile to the Farmington River (Figure 4-11). The area is approximately 4 acres in size. The Site brook has received industrial and diluted radiological waste waters and storm water runoff from the beginning of Site activities in 1955 though 1992. Since activities within wetlands require special permits and procedures in order to minimize impact to the ecosystem, additional information regarding this area will be provided in a future revision to the DP.

4.4 SURFACE WATER

There are five surface water bodies at the Site (Figure 3-2). They include the following:

- Great Pond: a kettle pond, located in the southwestern portion of the Site;
- Small Pond: formed by a dammed creek located in the southeastern portion of the Site:
- An unnamed tributary: that runs from the northern outlet of Small Pond and discharges to Goodwin Pond;
- Goodwin Pond: located just north of the former S1C facility; and
- Site brook: an unnamed brook that flows from Goodwin Pond and discharges to the Farmington River.

The Farmington River (northwest of the CE Windsor Site) is not considered part of the Site.

Of the five surface water bodies at the Site, only Great Pond receives stormwater from areas of commercial operations only. The other surface water bodies are either not impacted, or are impacted due to operation under the AEC contracts (Figure 4-11).

<u>Great Pond</u>

Great Pond is a glacial kettle pond that existed prior to the construction activities that began in the mid-1950s. There are no outlets or inlets to the pond, and with the exception of the northeast corner, Great Pond recharges groundwater radially.

In the early 1960s, CE would maintain the water elevation of Great Pond to support a stocked fish population. For a brief period (several years) CE would occasionally pump water from the Farmington River to maintain the pond elevation. Overflow from Great Pond would then drain to an outlet constructed in the southwest end of the Pond, and then discharge to an infiltration basin.

Great Pond is located north of Building 14 and west of several industrial buildings, including the former Building 5 complex, Building 6A, the Building 17 Complex, and a portion of the former industrial waste line that ran along the west side of Building 5 and 18. With the exception of Building 14 that houses the Site's cafeteria, post office, and offices, the areas east of Great Pond were impacted by radiological materials. Should any liquid release have occurred, the impacted water could have migrated down slope, and potentially impacted the eastern portion of Great Pond.

Due to the proximity to Great Pond of areas where radiological material may have been present, there was a potential for radiological material to have entered the pond via airborne emissions and/or surface runoff. Great Pond was cleared as part of Building Complexes 2, 5, 6A and 17 Final Status Surveys that were accepted by the NRC (NRC, 2007b).

<u>Small Pond</u>

Located on the east side of the Site, and received storm water runoff from the south and eastern portion of the Site, including areas around Buildings 3, 5, 6, and 7. Small Pond was initially formed by beaver dams located in the northern end of the pond in the early 1970s. The beaver dams were replaced by man-made dams in the mid 1970s. The pond likely receives groundwater discharge in its southern portions, near the inlet, but likely recharges groundwater at its northern extreme at the outlet to the tributary that flows to Goodwin Pond. The deepest point in the pond is approximately 5 feet during times of high water. During periods of low water the bottom sediments of the southern end of the pond are exposed.

Prior to the first beaver dam, Small Pond was a low lying wetland that drained to the north. In the late 1950s the area to the west was used as a disposal area for fill, construction materials, and crushed 55-gallon drums. Approximately 100 yards of soil and several crushed drums were removed under the VCA Program in 1999. However, several areas where crushed drums have been identified are still present. These areas are located along the western slope of Small Pond.

In 1999, CE conducted an electromagnetic (EM) survey across the western side of Small Pond and determined that there are several areas where metal objects were identified. No further action was taken to remove or sample the anomalies at that time, as Small Pond is currently undergoing characterization for Ecological Risk Assessment. Additional characterization activities will be performed in small pond to evaluate whether any of the metallic objects are radiologically impacted.

Tributary from Small Pond to Goodwin Pond

The small tributary from Small Pond to Goodwin Pond does not have a history of contamination. Samples collected from this tributary under the VCA Program did not contain radiological material above background. This tributary was not considered to be an area of concern under the VCA Program and is not considered to be radiologically impacted.

Goodwin Pond

Goodwin Pond is located in the northern portion of the Site. The western end of the pond is dammed and forms the headwaters for the Site brook. Goodwin Pond is located directly north of the former S1C Facility (Figure 3-2). Because of its proximity to the S1C Facility, Goodwin Pond was not investigated for chemical constituents under the Site's VCA Program. Instead, the S1C Facility and Goodwin Pond were investigated for chemicals by DOE under their own Corrective Action Program, under USEPA and CTDEP guidance.

Knolls Atomic Power Laboratory (KAPL) has conducted quarterly monitoring of both Goodwin Pond and the Site brook, and published results in annual Environmental Monitoring Reports. Based on their data, cobalt-60 and cesium-137 were detected in low concentrations in the sediments of the Site brook and its outfall to the Farmington River, but not in the sediments of Goodwin Pond. In addition, gamma emitting radionuclides were not detected in the surface water samples collected from Goodwin Pond. The lack of byproduct materials in Goodwin Pond sediments is most likely due to the location of KAPL's discharge of wastewater. The discharge was located below the Goodwin Pond dam in Site brook. The only discharge directly to the pond would have been from surface water runoff or infiltration from their leach field.

Site Brook

The Site brook flows west from Goodwin Pond for approximately one half mile to the Farmington River. The Site brook has received industrial and diluted radiological waste waters and storm water runoff from the beginning of Site activities in 1955 through 1992.

With the exception of the storm water outfall at the head of the brook, after 1992, all industrial and sanitary wastes were discharged to the MDC, radiological wastewater was evaporated on-site, and residues were disposed of in accordance with NRC regulations.

4.5 GROUNDWATER

Groundwater at the CE Windsor Site has been investigated for radiological constituents.

In June 2002, groundwater samples were collected from 72 groundwater monitoring wells and well points across the Site and were submitted for alpha and gamma spectroscopy. Groundwater samples were analyzed for cobalt-60 and cesium-137 by gamma spectroscopy using EPA Method 901.1, and uranium isotopes (U-234, U-235, and U-238) by alpha spectroscopy using DOE Method U-04 (modified). All samples from monitoring wells were collected using USEPA-approved low flow sampling techniques. Groundwater samples collected from well points, installed within the Site brook were filtered in the field using a 0.45 micron filter.

To better evaluate the groundwater quality, CE collected groundwater samples for both chemical and radiological analyses from five upgradient groundwater monitoring wells. These wells are located at the Site's eastern, southern, and western perimeters and represent the local background conditions or groundwater that has not been impacted by historic operations. The five background monitoring wells include MW-E01, ME-E03, MW-2401, MW-W01, and MW-S02. The locations of these wells and groundwater contours are shown on Figure 3-6. The radiological data from these wells are presented in Table 4-4.

Groundwater samples were collected from locations in and downgradient of areas where radiological materials were used and/or stored. The results were then compared to results from the background locations. This comparison indicated uranium at concentrations above background levels at two locations: MW-1201 and WP-1403S. Uranium concentrations were 11.48 pCi/L in MW-1201 and 18.566 pCi/L in WP-1403S. Sample results for the June 2002 sampling event are summarized in Table 4-5. The calculated total uranium results for these samples, however, are below the USEPA drinking water maximum contaminant level (MCL) for uranium of 30 μg/L.

Cobalt-60 was also detected in groundwater from monitoring location E-1. However, cobalt-60 was also detected in the method blank, and therefore the result was qualified as an estimated value. This location will be re-sampled in the future rounds to confirm/deny these results.

At this time, there is no reason to believe that groundwater remediation will be needed, and it is likely that the removal/remediation of waste pipes will remove the source of the minor contamination. It is unlikely that groundwater remediation due to radiological impacts will be required, with the exception of the collection and removal of groundwater that infiltrates the planned excavation of the industrial waste lines in the immediate vicinity of MW-1201. ABB implemented a radiological groundwater sampling program in 2002 and has completed 3 successive quarterly events at more than 70 monitoring locations. No results above maximum contaminant levels (MCLs) have been reported, and impacts above interpreted background are present at only two monitoring locations.

ABB expects to implement a post-remediation groundwater monitoring program for radiological parameters. This program would be conducted as required by the Connecticut Remediation Standard Regulation (RSR), but is not required under NRC regulations. This typically includes monitoring groundwater downgradient of areas where soil was remediated. Monitoring typically takes place quarterly for two to three years post-remediation, depending on the groundwater classification of the area being monitored. A post-remediation groundwater monitoring program will be developed after contaminated soil removal activities are completed. The number and location of wells to be sampled will be determined at that time.

Groundwater flow conditions vary across the Windsor Site. In general, groundwater associated with the proposed remediation areas in the southern portion of the Site flows to the east-northeast. Groundwater associated with the central and northern portions of the Site flows to the northeast towards Goodwin Pond and to the northwest towards the Farmington River.

Groundwater associated with the Woods Area, Drum Burial Pit, Clamshell Pile, Burning Grounds, Debris Piles, and Site Brook flows to the northwest towards the Farmington River. Other portions of the Windsor Site are located cross-gradient or upgradient of these remediation areas. Therefore, remediation of these areas of the Site will not impact groundwater conditions in other portions of the Windsor Site.

Groundwater associated with the Industrial and Radiological Waste Lines flows to the east and northeast towards Small Pond, Goodwin Pond, and the unnamed tributary that connects these two ponds. Other portions of the Windsor Site are downgradient of this remediation area, located approximately 600 to 1,000 feet beyond the proposed limits of remediation in this area.

Groundwater associated with the Equipment Storage Yard flows to the east and northeast towards Small Pond. Other portions of the Windsor Site are downgradient of this remediation area, located approximately 300 feet beyond the proposed limits of remediation in this area.

Groundwater associated with the Buildings 3 and 6 areas flows primarily to northeast towards Small Pond. Other portions of the Windsor Site are upgradient, cross-gradient, and downgradient of this remediation area. The closest downgradient location is approximately 1,000 feet beyond the proposed limits of remediation in this area.

Radiological groundwater monitoring is proposed for areas of the Site where remediation is proposed as interim monitoring until the post-remediation monitoring program begins. The proposed monitoring locations, frequency of monitoring, and constituents of concern that will be monitored for in each area of the Site are provided below.

Currently, five monitoring wells located downgradient of the previous remediation areas are being monitored for radiological constituents to meet the requirements of the CTDEP Remediation Standard Regulations (RSRs). This includes monitoring wells MW-0608, MW-0610R, MW-1203, MW-1507, and MW-1509 as shown in Figure 4-44. Groundwater monitoring at these locations for total uranium will continue until adequate data have been collected to meet the requirements specified in the CTDEP RSRs to allow groundwater monitoring to be discontinued.

Additional monitoring wells have been selected to evaluate groundwater conditions in support of the partial site release. The proposed monitoring locations are shown on Figure 4-44. The following provides a summary of the monitoring locations and rationale for the location.

MW-1603	Downgradient of Building Complexes 3 and 6
MW-1016	Downgradient of the Equipment Storage Yard
MW-E09DI	Downgradient of the Waste Lines
MW-E10DI	Downgradient of the Waste Lines
MW-13S	Downgradient of the Waste Lines
WW-2	Downgradient of the Drum Burial Pit
MW-0102	Downgradient of the Woods Area
MW-0103	Downgradient of the Burning Grounds

The above-listed monitoring wells are located downgradient of the respective remediation areas. Monitoring locations have not been identified for the Debris Piles, Site Brook, and the Clamshell Pile because groundwater associated with these remediation areas does not flow towards other portions of the Site.

Monitoring of the above-listed (additional) locations started in July 2008, and will be conducted on a semi-annual basis (at a minimum). Groundwater samples collected from MW-0103 (downgradient of the Burning Grounds) will be monitored for radium-226 and thorium-232. Groundwater samples collected from the other monitoring locations (MW-1603, MW-1016, MW-E09DI, MW-E10DI, MW-13S, WW-2, and MW-0102) will be monitored for total uranium. These are the radiological constituents that will be addressed during soil remediation in these areas of the Site.

The data collected will be used to establish "baseline" groundwater conditions associated with these areas of the Site prior to initiating remediation activities. Data collected during remediation activities will serve to assess possible effects of soil disturbances on groundwater.

5.0 DOSE MODELING

This section describes the development of DCGLs for the Site. A DCGL is a site-specific concentration determined to be protective of the health of individuals that might be exposed in the future to the residual radioactivity that might be left in place on the Site. The DCGLs were calculated to meet requirements set by the NRC.

To release this property from regulatory control and terminate the Site radioactive materials license, the risks to human health associated with potential exposure to radioactivity originating at the Site must be evaluated and demonstrated to be within acceptable limits. To comply with NRC criteria for site release, the residual radioactivity at the Site must not contribute an annual radiation dose in excess of the NRC criteria and must be reduced to concentrations that are ALARA taking into account existing socio, political, and economic factors. While the State of Connecticut does not have direct regulatory authority over the radiological constituents at the Site,[1] it is the expressed desire of both ABB and the NRC to consider the wishes of the State of Connecticut and local municipalities in arriving at decisions related to the release of the Site from radiological control. To this end, the State of Connecticut and the Town of Windsor have been consistently involved with ABB in discussions and decisions involving the acceptable basis for releasing the Site.

Radiological contamination at the Site is being addressed following MARSSIM guidance through the development of a Historical Site Assessment (Harding ESE, 2002) and the preparation of the DCGLs. The DCGLs are based on conservative assumptions and provide limits that are consistent with those required by the NRC.

5.1 SOIL DCGLS

The DCGLs are presented in the Derivation of Site-Specific Soil DCGL report (ABB, 2003), re-submitted to the NRC for review and approval on October 15, 2003. The development of the DCGLs was performed using a steering committee of stakeholders, including USACE, CTDEP, and the Town of Windsor to assess realistic future use scenarios and appropriate assumptions. In addition to the DCGLs developed for uranium and reactor byproduct, ABB intends to use NRC default values for the Burning Grounds area as discussed below.

It is anticipated that future uses of the Site will be roughly consistent with its current use (commercial, light industrial uses). The current land use in surrounding area is a mixture of commercial, light industrial, warehousing, office park, residential, municipal landfill, and commercial farming. The land use is trending toward commercial and industrial uses.

Commercial farming of both consumable produce and cigar tobacco does occur in the vicinity of the CE Site. Such commercial farms are characterized as large fields that are planted with a single commercially viable crop (such as tobacco, corn, cucumbers, etc.) and are harvested in bulk and trucked to a commercial wholesale buyer for subsequent distribution. They do not support residential habitation or subsistence. Locally such farms are known as “truck” farms.

Future residential use of the land is considered possible given the current community growth, planning, and development strategies of the local municipality. Therefore it is reasonable

[1] The NRC has sole regulatory authority in matters related to licensure and regulation of activities involving licensed radioactive materials.

and credible to consider that the land might be used for locating residential dwellings in the future.

ABB considers the future use of the Site for subsistence farming to be highly improbably. The likelihood that subsistence farming might occur at this Site is thought to be remote because: 1) the general population is moving away from subsistence farming; 2) the amount of land required to support subsistence farming is economically infeasible considering the value of the land; and 3) the population demographics in the Windsor area are consistent with east coast urban/suburban uses and trending away from farm/agricultural uses. In spite of the evidence supporting the conclusion that subsistence farming is highly improbable as a reasonable future use at the Site, a resident farming scenario has been included in this evaluation to meet CTDEP requirements.

From among the many scenarios and variants considered, a suite of six separate future use scenarios has emerged as candidates to be considered in the development of the site-specific soil DCGLs. The future use scenarios evaluated include:

- Workers exposed while working occupationally at the Site's facilities
- Construction Activities
- Users of Publicly Accessible Lands designated for recreational uses
- Commercial Crop "Truck" Farming
- Suburban Residential
- Resident Farming

The DCGLs for the CE Site were calculated using the RESRAD 6.0 modeling code. Each of the scenarios modeled results in a concentration corresponding to both the 19 mrem/y and 25 mrem/y dose limit. Considering the potential future land-use scenarios, the limiting scenario (the one that results in the smallest concentration yielding 19 mrem/y) is the Residential Farm scenario. Based on this scenario, a DCGL is proposed for each of total uranium in soil and reactor byproduct as Co-60 in soil.

Conservatism was built into the dose modeling, and thus the proposed DCGLs, by conscientiously selecting exposure factor values that err on the side of safety when confronted with uncertainty in the selection of input parameter values.

The DCGLs proposed have been derived using appropriate techniques in accordance with governing guidance, standards, and regulations. In addition, stakeholders including the Town of Windsor, CTDEP, and USACE have all provided input into the parameter selection and scenario derivation.

The dose evaluation described above will provide risk managers and decision makers with the substantive basis necessary to set and approve site-specific permissible concentration standards, the DCGLs, derived from the applicable regulatory limits for public dose. When approved, the proposed DCGLs will become the permissible soil and sediment concentration limits for unrestricted release of the CE Windsor Site from radiological controls.

The Burning Grounds area has residual radioactivity resulting from past operations involving the burning of zirconium and thorium metals for disposal. These materials were processed more than 30 years ago. As such, the residual materials are stable and localized in surficial soils as evidenced by the characterization data.

Furthermore, the Burning Grounds area is relatively small with respect to the remaining areas for remediation and the Site as a whole. The NRC screening values for surface soil contamination release levels were published in Federal Register on December 7, 1999 (64 FR 68395) in Table 3. Dose modeling for the NRC screening values utilized a residential exposure pathway that included ingestion of plants and animals grown on the property. This is similar to the exposure pathway for the approved site-specific DCGLs (residential farmer scenario). The default parameters for dose modeling of the NRC screening values have been established to be highly conservative. The combination of these factors indicates that it would be acceptable to use the NRC screening values as DCGLs for the Burning Grounds.

Due to the period of time since these materials were last processed (more than 30 years ago) and the short half-life of decay progeny for both Th-232 and Ra-226, it is assumed that decay progeny will have achieved secular equilibrium. Consequently the NRC screening values selected are those indicated in Table 3 with a "+C" which denotes that contributions from the complete chain of progeny in equilibrium with the parent radionuclide are included in the screening value.

Therefore the soil DCGLs for the Site include site-specific DCGLs, total uranium 557 pCi/g and Co-60 5pCi/g. In addition, the NRC screening values for Th-232 (+C) 1.1 pCi/g and Ra-226 (+C) 0.6 pCi/g will be used for the Burning Grounds area.

5.2 BUILDING DCGLS

Building DCGLs for the Site were calculated using RESRAD-BUILD Version 3.4 modeling code. The future use of the High Bay of Building 3 is anticipated to be the same as its current use, commercial occupancy. Another potential scenario for the High Bay is building renovation or demolition. Therefore these two scenarios were evaluated as exposure pathways as presented in the Development of Building DCGLs report (ABB, 2008). For each scenario, a DCGL is proposed for total uranium and reactor byproduct (Co-60) for building surfaces yielding 19 mrem per year. In addition, total uranium was further evaluated at low enrichment (3%) and high enrichment (90%).

The DCGLs proposed have been derived using appropriate techniques in accordance with governing guidance, standards, and regulations. Conservatism was built into the dose modeling by conscientiously selecting exposure factor values that err on the side of safety when confronted with uncertainty in the selection of input parameter values.

Results of the modeling found that 90% enriched uranium was a more potent dose producer in building exposure scenarios. For total uranium, building renovation or demolition is more conservative than building occupancy, and inhalation is the primary exposure pathway. On the other hand, reactor byproduct was more restrictive for building occupancy, and external dose is the primary exposure pathway. The proposed building surface DCGLs are 20,148 dpm/100 cm^2 for total uranium and 6,980 dpm/100 cm^2 for reactor byproduct (Co-60).

6.0 ALTERNATIVES CONSIDERED AND RATIONAL FOR CHOSEN ALTERNATIVE

It is the objective of ABB to decommission the CE Windsor Site, including associated buried piping and adjacent grounds, such that the entire Site will meet the criteria for unrestricted use as specified by 10 CFR 20.1402.

This is the most conservative decommissioning approach, and meets all regulatory requirements. No other alternatives were evaluated.

7.0 ALARA ANALYSIS

7.1 INTRODUCTION

This pre-remediation ALARA analysis has been conducted to demonstrate that the dose criteria in Subpart E of 10 CFR Part 20 has been met and concludes whether it is feasible to further reduce the levels of residual radioactivity to levels below those necessary to meet the dose criteria (i.e., to levels that are ALARA). This analysis encompasses the remaining impacted portions (approximately 30 acres of land out of 600-acres) and Building 3 High Bay located at the CE Site.

It is the intention of the licensee (ABB) to remediate the Site by removing and shipping contaminated soils and waste to a low-level waste disposal facility such that it meets the unrestricted use criteria presented in 10 CFR 20.1402 (TEDE to an average member of the critical group that does not exceed 25 mrem/y). Site-specific DCGLs derived for the CE Windsor Site as described in Section 5.0 were selected on the limiting scenario which met CTDEP requirements. CTDEP has determined that a potential future dose of 19 mrem/y is protective of human health and satisfies the requirements of its RSR regulatory framework. Therefore, ALARA is inherent since DCGLs are derived for 19 mrem/y.

Based on the licensee's decision to remediate to unrestricted use criteria, and using appropriate dose modeling to relate concentrations to dose, the licensee can apply the allowance given in Section 1.5, Appendix N of NUREG-1757, Volume 2 (NRC, 2006b) which states "In certain circumstances, the results of an ALARA analysis are known on a generic basis and an analysis is not necessary. For residual radioactivity in soil at sites that will have unrestricted release, generic analyses show that shipping soil to a low-level waste disposal facility is unlikely to be cost effective for unrestricted release, largely because of the high cost of waste disposal. Therefore shipping soil to a low level waste disposal facility generally does not have to be evaluated for unrestricted release. In addition, licensees that have remediated surface soil such that it meets the unrestricted use criteria in 10 CFR 20.1402 would not be required to demonstrate that these levels are ALARA."

In addition, Section 1.5, Appendix N of NUREG-1757, Volume 2 (NRC, 2006b) also addresses building ALARA levels by stating, "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".

With this in mind, the results of an ALARA analysis are "known on a generic basis and an analysis is not necessary." However, to keep in the spirit of ALARA, simplified analyses (possible benefits and costs relating to decommissioning, a determination of residual radioactivity levels that are ALARA, and cost vs. soil activity and building contamination levels) are presented below.

NUREG-1757, Volume 2 Appendix N provides information outlining a simplified method to estimate when a proposed remediation guideline is cost effective. Possible benefits, as well as possible costs, are derived and compared. If the desired beneficial effects (benefit) from the remediation action are greater than the undesirable effects (cost) of the action, the remediation action being evaluated is cost-effective and should be performed. Conversely, if the benefits are less than the cost, then the level of residual radioactivity is already ALARA without taking additional remediation action.

Table N1, Possible Benefits and Costs Related To Decommissioning (NUREG-1757, Volume 2)

Possible Benefits	Possible Costs
• Collective Dose Averted	• Remediation Costs
• Regulatory Costs Avoided	• Occupational Non-radiological Risk
• Changes in Land Values	• Additional Occupational/Public Dose
• Esthetics	• Transportation Direct Costs and Implied Risks
• Reduction in Public Opposition	• Environmental Impacts
	• Loss of Economic Use of Site/Facility

During these analyses, results from an appropriate dose modeling method were used to relate concentrations to dose. Information used for these analyses, regarding concentration to dose values, were obtained from the Derivation of the Site-Specific DCGLs (ABB, 2003) and Development of Building DCGLs (ABB, 2008).

7.2 DETERMINATION OF BENEFITS

7.2.1 Collective Dose Averted Benefit

Remediation of Site soils and building surfaces to levels that meet the unrestricted use criteria in 10 CFR 20.1402, using appropriate dose modeling to relate concentrations to dose, is known on a generic basis to demonstrate that these levels are ALARA. Therefore, calculation of the collective dose averted is not required in this analysis. However, collective dose costs are determined and evaluated in Section 7.3.

7.2.2 Regulatory Cost Avoided Benefit

Based on the decision to remediate the Site to unrestricted use criteria, costs associated with a decision to remediate the Site to a Restricted Release level (additional licensing fees, financial assurance related to both the decommissioning fund and the Site restriction, cost associated with public meetings or the community review committee and future liability release the Site) are avoided, and not taken into account in this analysis.

7.2.3 Changes in Land Value Benefit

Land and buildings released for unrestricted use from this Site would be primarily suited for industrial use due to its current status, geographical location, and proximity to other industrial type land uses (bordered by agricultural and commercial land to the south; tobacco fields and a sand and gravel quarry to the west; the Windsor/Bloomfield Sanitary Landfill and Recycling Center (Landfill) to the north; and forested land as well as some residential and commercial developments to the east).

Regardless of future land use, the Site will be remediated to unrestricted use criteria levels (established DCGLs) which allow for any and all land use scenarios, including the most restrictive - Residential Farm Scenario. In light of this, changes in land value can occur without adverse effect on the remediation activities planned for the Site, thus no additional land value benefit is gained with additional remediation activities.

7.2.4 Esthetics Benefit

The Site has minimal public visibility due to rolling hills and local vegetation/forest growth. The decommissioning plan calls for contaminated soil removal to meet the established DCGL soil activity values, followed by land reforestation and vegetation restoration. However, if a decision was made to remediate below the established DCGL value, an increasing quantity of previously undisturbed local forested land would be disrupted and removed. This additional remedial action would increase the overall environmental disturbance of the land, and prove to be a negative esthetics benefit overall.

7.2.5 Reduction in Public Opposition

ABB, in cooperation with the U.S. Army Corps of Engineers–New England District commissioned the formation of a steering group representing ABB, USACE, the State of Connecticut, and the Town of Windsor. The CE Site Project Steering Group was charged with representing stakeholder's interests in the process to derive a dose-based soil concentration guideline (DCGL) value specific to the CE Windsor Site. ABB, the USACE, and the NRC desire the cooperative input from the identified stakeholders and state regulators so that the final DCGLs are acceptable not only to NRC but also to the State of Connecticut and the impacted community.

7.3 DETERMINATION OF COSTS

The determination of costs excludes Environmental Impacts and Loss of Economic Use of Site/Facility costs. These two areas of cost are excluded in the analyses based on the decision of the licensee to remediate to unrestricted use criteria presented in 10 CFR 20.1402. This level of remediation ensures that the Site will be available for any future proposed activity, hence eliminating the loss of economic use. In addition, the Site decommissioning plan calls for environment restoration (reforestation and vegetation) of the Site restoration areas. These costs are unavoidable regardless of the remediation method and therefore excluded.

7.3.1 Determination of Total Costs

The total cost of a decommissioning alternative is determined in accordance with equation N-3 of NUREG-1757, Volume 2, Appendix N which states:

$$Cost_T = Cost_R + Cost_{WD} + Cost_{Acc} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} + Cost_{Other}$$

Where:
$Cost_T$ = Total cost of the remediation action;
$Cost_R$ = Monetary cost of the remediation action;
$Cost_{WD}$ = Monetary cost for transport and disposal of the waste generated by the action;
$Cost_{Acc}$ = Monetary cost of worker accidents during the remediation action;
$Cost_{TF}$ = Monetary cost of traffic fatalities during transporting of the waste;
$Cost_{WDose}$ = Monetary cost of dose received by workers performing the remediation action and transporting waste to the disposal facility;

$Cost_{PDose}$ = Monetary cost of the dose to the public from excavation, transport, and disposal of the waste;
$Cost_{Other}$ = Other costs as appropriate for the particular situation.

7.3.1.1 Soil Remedial Action – Removal of Contaminated Soil To A DCGL Value of 19 mrem per year.

The cost of the remedial action includes the cost of remediation, cost of waste disposal, and the cost due to traffic fatalities; but does not include land restoration costs. The total cost of the Soil Remedial Action is: $38,594,591.

Cost Function	Monetary Cost	Formula
$Cost_R$ + $Cost_{WD}$	$38,386,267	based on the cost of $5,737/m^3 x 6,691 m^3
$Cost_{Acc}$	$1,366	based on the movement of a total of 6,691 m^3 of materials, at a rate of 1.62 person-hours per m^3, an accident rate of 4.2 x 10^{-8} fatalities per person-hour, and a cost of $3,000,000 per fatality
$Cost_{TF}$	$206,959	based on 6,691 m^3 being shipped /13.6 m^3/shipment x 3690 km x 3.8 x 10^{-8} fatalities/km at a cost of $3,000,000 per fatality
$Cost_{WDose}$	NA	based on dose modeling performed, dose to an average construction worker is estimated to be 8 mrem/y at a soil concentration of 1500 pCi/g Total U or 14.5 mrem/y at a soil concentration of 12.6 pCi/g Co-60. At $2000 per rem person x 0.008 rem/y (0.014.5 rem/y) = $16 ($29) per year per construction worker. This dollar value is insignificant and will not add significant cost to the total cost of remediation and need not be evaluated for the different alternatives
$Cost_{PDose}$	NA	Dose to the public from excavation, transport, and disposal of the waste is negligible, hence monetary cost of the dose to the public from excavation, transport, and disposal of the waste is negligible, and will not add a significant cost to the total cost of remediation
$Cost_{Other}$	NA	Land restoration costs are not included in this analysis
$Cost_T$	$38,594,591	($Cost_R$ + $Cost_{WD}$) + $Cost_{Acc}$ + $Cost_{TF}$ + $Cost_{WDose}$ + $Cost_{PDose}$ + $Cost_{Other}$

NOTE: Monetary cost of the remediation action ($Cost_R$) and the monetary cost for transport and disposal of the waste generated by the action ($Cost_{WD}$) is combined into one value for this assessment.

7.3.1.2 Building Remedial Action – Removal of Contaminated Buildings To A DCGL Value of 19 mrem per year.

The cost of the remedial action includes the cost of remediation, cost of waste disposal, and the cost due to traffic fatalities; but does not include building restoration costs. The total cost of the Building Remedial Action is: $3,287,837.

Cost Function	Monetary Cost	Formula
$Cost_R$ + $Cost_{WD}$	$3,270,090	based on the cost of $5,737/$m^3$ x 570 m^3
$Cost_{Acc}$	$116	based on the movement of a total of 570 m^3 of materials, at a rate of 1.62 person-hours per m^3, an accident rate of 4.2 x 10^{-8} fatalities per person-hour, and a cost of $3,000,000 per fatality
$Cost_{TF}$	$17,631	based on 570 m^3 being shipped /13.6 m^3/shipment x 3690 km x 3.8 x 10^{-8} fatalities/km at a cost of $3,000,000 per fatality
$Cost_{WDose}$	NA	based on dose modeling performed, dose to an average construction worker is estimated to be 8 mrem/y at a building surface concentration of 8,500 dpm/100 cm^2 Total U. At $2000 per rem person x 0.008 rem/y = $16 per year per construction worker. This dollar value is insignificant and will not add significant cost to the total cost of remediation and need not be evaluated for the different alternatives
$Cost_{PDose}$	NA	Dose to the public from demolition, transport, and disposal of the waste is negligible, hence monetary cost of the dose to the public from demolition, transport, and disposal of the waste is negligible, and will not add a significant cost to the total cost of remediation
$Cost_{Other}$	NA	Building restoration costs are not included in this analysis
$Cost_T$	$3,287,837	($Cost_R$ + $Cost_{WD}$) + $Cost_{Acc}$ + $Cost_{TF}$ + $Cost_{WDose}$ + $Cost_{PDose}$ + $Cost_{Other}$

NOTE: Monetary cost of the remediation action ($Cost_R$) and the monetary cost for transport and disposal of the waste generated by the action ($Cost_{WD}$) is combined into one value for this assessment.

7.4 DETERMINATION OF RESIDUAL RADIOACTIVITY LEVELS THAT ARE ALARA

7.4.1 ALARA Analysis

The purpose of this section is to determine whether the DCGLs selected for remediation action are ALARA. Since the intent of the calculation is to determine whether additional soil or building surfaces should be remediated in order to lower the radiological dose, only the cost associated with the additional remediation is used as input for the this section.

Soil or building surface concentrations that are ALARA are determined in accordance with Equation N-8 of NUREG-1757, Volume 2 Appendix N, where:

$$\frac{\mathrm{Conc}}{\mathrm{DCGL_W}} = \frac{\mathrm{Cost_T}}{\$2000 \times \mathrm{P_D} \times 0.025 \times \mathrm{F} \times \mathrm{A}} \times \frac{r+\lambda}{1-e^{-(r+\lambda)N}}$$

This calculation allows the licensee to estimate a concentration at which a remediation action will be cost-effective prior to starting remediation. Results less than one (1) indicate the remediation action is warranted to meet the ALARA requirement. Results greater than one (1) indicate the remediation action is not warranted to meet the ALARA requirement.

7.4.1.1 ALARA Analysis for Soil Remedial Action

Inputs and Assumptions:

Cost Function	Monetary Cost	Formula
$\mathrm{Cost_R}$ + $\mathrm{Cost_{WD}}$	\$3,838,053	based on the cost of \$5,737/m^3 x 669 m^3 (10% of total volume for the remedial action)
$\mathrm{Cost_{Acc}}$	\$137	based on the movement of a total of 669 m^3 of materials, at a rate of 1.62 person-hours per m^3, an accident rate of 4.2 x 10^{-8} fatalities per person-hour, and a cost of \$3,000,000 per fatality
$\mathrm{Cost_{TF}}$	\$20,693	based on 669 m^3 being shipped / 13.6 m^3/shipment x 3690 km x 3.8 x 10^{-8} fatalities/km at a cost of \$3,000,000 per fatality
$\mathrm{Cost_{WDose}}$	NA	based on dose modeling performed during the generation of the DCGL Report, dose to an average construction worker is estimated to be 8 mrem/y at a soil concentration of 1500 pCi/g Total U or 14.5 mrem/y at a soil concentration of 12.6 pCi/g Co-60. At \$2000 per rem person x 0.008 rem/y (0.014.5 rem/y) = \$16 (\$29) per year per construction worker. This dollar value is insignificant and will not add significant cost to the total cost of remediation and need not be evaluated for the different alternatives
$\mathrm{Cost_{PDose}}$	NA	Dose to the public from excavation, transport, and disposal of the waste is negligible, hence monetary cost of the dose to the public from excavation, transport, and disposal of the waste is negligible, and will not add a significant cost to the total cost of remediation
$\mathrm{Cost_{Other}}$	NA	Land restoration costs are not included in this analysis
$\mathrm{Cost_T}$	\$3,858,882	($\mathrm{Cost_R}$ + $\mathrm{Cost_{WD}}$) + $\mathrm{Cost_{Acc}}$ + $\mathrm{Cost_{TF}}$ + $\mathrm{Cost_{WDose}}$ + $\mathrm{Cost_{PDose}}$ + $\mathrm{Cost_{Other}}$

Symbol	Definition or Value
Conc	The average concentration of residual radioactivity in the area that will be considered ALARA
$DCGL_w$	Derived Concentration Guideline Level
r	0.03
N	1000 years
A	121,410 m^2
λ	0
F	0.5 (assuming 50% of the source term is removed during remediation activities)

$$\frac{\text{Conc}}{\text{DCGL}_W} = \frac{\$3,858,882}{\$2000 \times \left(4 \times 10^{-4}\right) \times 0.025 \times 0.5 \times 121,410} \times \frac{0.03 + 0}{1 - e^{-(0.03+0)1000}}$$

$$\frac{\text{Conc}}{\text{DCGL}_W} = 95$$

Since this value is greater than one (1), it is determined that the Remedial Action soil DCGL is ALARA, and no additional remediation action is warranted.

7.4.1.2 ALARA Analysis for Building Remedial Action
Inputs and Assumptions:

Cost Function	Monetary Cost	Formula
$Cost_R$ + $Cost_{WD}$	\$4,150	based on the cost of \$37,730/m^3 x 0.11 m^3 (decontamination of an area 360 square feet at a depth of 1/8 inch)
$Cost_{Acc}$	NA	based on the movement of a total of 0.11 m^3 of materials, at a rate of 1.62 person-hours per m^3, an accident rate of 4.2 x 10^{-8} fatalities per person-hour, and a cost of \$3,000,000 per fatality (less than \$1) This dollar value is insignificant and will not add significant cost to the total cost of remediation and need not be evaluated for the different alternatives
$Cost_{TF}$	\$3	based on 0.11 m^3 being shipped / 13.6 m^3/shipment x 3690 km x 3.8 x 10^{-8} fatalities/km at a cost of \$3,000,000 per fatality

Cost Function	Monetary Cost	Formula
$Cost_{WDose}$	NA	based on dose modeling performed, dose to an average construction worker is estimated to be 8 mrem/y at a building surface concentration of 8,500 dpm/100 cm^2 Total U. Decontamination of 360 square feet will take one day, which is equivalent to 0.03 mrem per worker. At $2000 per rem person x 0.00003 rem/y = $0.06 per year per construction worker. This dollar value is insignificant and will not add significant cost to the total cost of remediation and need not be evaluated for the different alternatives
$Cost_{PDose}$	NA	Dose to the public from decontamination, transport, and disposal of the waste is negligible, hence monetary cost of the dose to the public from decontamination, transport, and disposal of the waste is negligible, and will not add a significant cost to the total cost of remediation
$Cost_{Other}$	NA	Building restoration costs are not included in this analysis
$Cost_T$	$4,153	$(Cost_R + Cost_{WD}) + Cost_{Acc} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} + Cost_{Other}$

Symbol	Definition or Value
Conc	The average concentration of residual radioactivity in the area that will be considered ALARA
$DCGL_w$	Derived Concentration Guideline Level
r	0.09
N	70 years
A	34 m^2
λ	0
F	0.5 (assuming 50% of the source term is removed during remediation activities)

$$\frac{Conc}{DCGL_w} = \frac{\$4,153}{\$2000 \times (0.09) \times 0.025 \times 0.5 \times 34} \times \frac{0.07 + 0}{1 - e^{-(0.07+0)70}}$$

$$\frac{Conc}{DCGL_W} = 3.8$$

Since this value is greater than one (1), it is determined that the Remedial Action building DCGL is ALARA, and no additional remediation action is warranted.

7.5 CONCLUSION

These pre-remediation ALARA analyses demonstrate that the dose criteria in Subpart E of 10 CFR Part 20 will be met using the Remedial Action soil and building DCGLs and demonstrates that the Remedial Action soil and building DCGLs are ALARA.

As presented in Section 7.2, Determination of Benefits, all possible benefits are realized with no comparison against cost for the benefit. This is primarily due to the decision of the licensee to remediate to unrestricted use values, incurring those costs regardless of the outcome of the analysis.

As presented in Section 7.3, Determination of Cost, costs were determined for the Remedial Actions.

As presented in Section 7.4.1, ALARA Analysis, an ALARA analysis for both soil and building Remedial Actions were performed. The ratio of Conc to $DCGL_W$ was calculated to be 95 for soils and 3.8 for buildings. Since these values are greater than one, it is determined that both soil and building $DCGL_W$ for the Remedial Actions for the unrestricted release of soil and buildings are ALARA, and no additional remediation is justified.

8.0 PLANNED DECOMMISSIONING ACTIVITIES

The scope of decommissioning activities includes the decontamination and deconstruction of structures in Building Complexes 3 and 6, including concrete foundations, the removal of buried utilities, remediation of impacted soil areas, removal of impacted subsurface utilities (industrial and radiological waste lines, sanitary lines, storm water lines), and the transportation and disposal of radioactive and mixed waste. Conduct of Final Status Surveys for impacted areas of the Site will be conducted as appropriate.

Decommissioning activities for Building Complexes 2, 5, 6A, and 17 are complete. For Buildings 3 and 6, the sequence of decontamination and deconstruction of above grade structures will generally follow the outline shown below (which was applied to Building Complexes 2, 5, 6A and 17):

- Identification of equipment by type/piping, tanks or ductwork system
- Radiological Characterization;
- Asbestos removal/ Interior Transite Removal;
- Hazardous material removal;
- Radiological decontamination;
- Equipment/systems dismantlement;
- Exterior transite and asbestos roofing removal;
- Concrete Masonry Removal;
- Structural Demolition;

Once the above ground structures have been removed, the D&D of the below grade structures will begin. This process will generally follow the outline shown below:

- Pavement, foundations, and below grade utilities;
- Removal of radiologically impacted soil;
- Waste Disposal; and
- Site restoration
- Final Status Surveys/Sampling

As described in Section 4, there is some volumetric contamination of building materials in Buildings 3 and 6. An evaluation will be performed to determine if it is safe and effective to remove these materials prior to dismantlement of the buildings. If removal prior to dismantlement is safe and effective, then this will be the preferred approach, and the buildings will be unconditionally released prior to deconstruction using existing license surface release criteria. If not, then the buildings will be dismantled with limited decontamination (similar to Building 17 in the Commercial D&D Areas) and the building materials will be carefully screened and segregated for disposal.

In addition, the High Bay of Building 3 (south end) will not be dismantled, as this portion of the building houses a unique fossil fuel research facility. The original portion of the High

Bay (Figure 4-13) was used for final fuel assembly, packaging and shipping and does not have residual radioactivity that will require remediation. A series of Final Status Surveys will be performed and documented to demonstrate that this portion of the building is acceptable for unrestricted release.

8.1 CONTAMINATED STRUCTURES

8.1.1 Building Demolition

The above-grade structures in Building Complexes 2, 5, 6A, and 17 have been dismantled, including concrete foundations, the removal of buried utilities.

Buildings 3 and 6 will be dismantled as part of decommissioning of the Site as indicated in the schedule provided as part of Section 8.5. This scope includes the separation and segregation of controlled materials and the minimization of hazardous and mixed waste generated. Please note that the slabs and foundations deeper than four feet also may remain in place if shown to meet appropriate release criteria.

8.1.2 Radiation Protection Methods

Radiation protection methods are described in Section 10.

8.1.3 Structural Steel Removal

Mechanical means of cutting and removing the structural steel shall be employed to the largest extent possible.

The non-ACM roof deck and roofing material, panels and concrete floor decking shall be demolished with the structure whenever possible.

Steel beams, joists, purlins, etc. will be sheared as close to the joints (cross members, plates, decking, etc.) as practical to create long, accessible (straight) metal pieces. Structural Steel shall be sized and segregated for disposal.

8.1.3.1 Sequence for Building 2 and 5 Complexes (Deleted – Revision 1)

8.1.3.2 Sequence for Building 17 Complex (Deleted – Revision 1)

8.1.3.3 Sequence for Building 6A (Deleted – Revision 1)

8.1.3.4 Sequence for Building 3 Complex

The structural components will be cut using a shear/grappler starting at one side and working to the other side of a bay. Structural steel will then be pulled and set aside for sizing and removal. Other building components will be removed in a similar manner.

The steel components will be sheared to size, and whenever possible, loaded directly into containers. Smaller components will be shuttled to containers using skid-steer loaders.

8.1.3.5 Sequence for Building 6 Complex

Building 6 is primarily concrete construction, and will be removed as described in Section 8.1.4.1 below.

8.1.4 Concrete, Masonry and Pavement Removal

8.1.4.1 Concrete Removal Procedure

Manual jackhammers, equipment mounted jackhammers (hoe ram), skid-steer loader or shears will be used to remove/dismantle and size reduce concrete, or CMU (Concrete Masonry Unit) structures. CMU walls may also be brought down using pushover techniques.

High density concrete structures will be removed/dismantled using shear mounted pneumatic hammers or a shear mounted pulverizer.

Steel reinforcement bars will be torch-cut, sheared, or saw-cut as required for dismantlement, leveling, or size reduction purposes.

All concrete foundations with some exceptions will be removed. Deep Basements and pits in the complexes will be removed to a depth of four (4) feet below grade. Holes will be punched in slab floors to release precipitation.

8.1.4.2 Removal of Slab and Foundation

The slab will be broken by utilizing a track mounted pneumatic hammer, or excavator bucket. All material will be size reduced as required for containerization.

Following demolition and size reduction concrete debris will be containerized for disposal.

8.1.5 Procedures

Contaminated structures in the remaining impacted portions will be decommissioned following procedures and activities established in DP Revision 0 (Building Complexes 2, 5, 6A and 17):

- Decontamination of above grade and basement structures.
- Removal of interior systems, components, walls, floors, piping, wiring, conduit, etc.
- Use of invasive means such as scabblers, and CO_2 or hydro blasting of surfaces to remove surface contaminants.
- Cutting of pipes and equipment using powered saws and torches to reduce the volume of radioactive waste materials and to allow for easier storage of waste materials in containers.
- Abatement of asbestos using stripping methods approved by the State of Connecticut Department of Public Health. A Connecticut licensed asbestos abatement contractor will be used to perform this work. Abatement contract personnel are provided radiological safety training to allow access to areas where the asbestos is radiologically contaminated.
- Abatement of lead and/or PCB based paints using stripping methods approved by the State of Connecticut Department of Health. A Connecticut licensed lead abatement contractor will be used to perform this work. Abatement contract personnel are provided radiological safety training to allow access to areas where the lead is radiologically contaminated.

- Use of pneumatic and electric driven chippers and jackhammers to remove volumetrically contaminated concrete surfaces.
- Storage, packaging and shipping of waste materials.
- Deconstruction of above-grade structures.

8.1.6 Decontamination Techniques

The methods for decontamination may include, but are not limited to:

- HEPA Vacuuming – HEPA vacuuming can be used to remove large amounts of loose dust/debris from surfaces.
- Wet Wiping – Small areas or items that require re-cleaning after the survey, or areas inaccessible to vacuuming will be decontaminated by use of wet wipes (water dampened disposable wipes) or wipes used with a cleaner/detergent.
- Hand Scabbling with HEPA Exhaust – May be employed to remove contamination from concrete surfaces.
- Floor Scabbling with HEPA Exhaust – May be employed to remove contamination from concrete surfaces.
- Roto-Peen with HEPA Exhaust – May be employed to remove contamination from concrete surfaces.
- Hydroblasting – May be employed to remove contamination from concrete or metal surfaces.

8.1.7 Request for Additional Information

Additional information regarding building demolition was provided as a response to NRC request for additional information (ABB, 2004) and will apply for the remaining impacted portions:

> *Building demolition will be accomplished by pushing the building structural components onto the concrete flooring and support footers of the building footprint. It is expected that minimal debris will fall to the soil areas surrounding the buildings. Additionally, the surface soil contamination levels exterior to these buildings are in most areas indistinguishable from background concentrations, and where there are isolated areas of elevated surface soil concentrations, these levels are less than 30 pCi/g. Because of the minimal soil contamination present, even if debris makes contact with the soil there is little chance of the debris being contaminated in excess of Regulatory Guide 1.86 release levels (the current approved unrestricted release levels for this project). All Construction and Demolition waste will be placed in roll-offs and those roll-offs surveyed prior to transport from the Site to a disposal/recycling facility.*

8.2 CONTAMINATED SYSTEMS AND EQUIPMENT

8.2.1 Remediation Tasks

Decommissioning activities related to systems and equipment include the removal of interior systems, components, walls, floors, piping, wiring, conduit, etc. from buildings. This scope includes the separation and segregation of controlled materials and the minimization of hazardous and mixed waste generated.

In addition, the remaining industrial and hot waste lines associated with licensed activities will be removed. Impacted segments of sanitary and storm waste lines will also be removed, similar to activities performed in the Commercial D&D Areas. Process and controls for removal of contaminated systems and pipelines will follow those established in the approved DP.

8.2.2 Decontamination Techniques

The methods for decontamination may include, but are not limited to those techniques described in Section 8.1.6.

8.2.3 Radiation Protection Methods

Radiation protection methods are described in detail in Section 10.

8.2.4 Equipment to be Removed

The scope of decommissioning activities as related to systems and equipment includes the removal of interior systems, components, walls, floors, piping, wiring, conduit, etc. from buildings.

8.2.5 Procedures

Contaminated systems and equipment in the remaining impacted portions will be decommissioned following procedures and activities established in DP Revision 0 (Building Complexes 2, 5, 6A and 17).

8.2.6 Use of Procedures

Decommissioning activities involving licensed material will be conducted in accordance with approved, written procedures or Radiation Work Permits (RWP), as described in Section 7.5 of License No. 06-00217-06.

8.3 SOIL AND SUBSURFACE UTILITIES

8.3.1 Removal/Remediation Tasks

Contaminated soil will be removed to below approved DCGLs. In addition, impacted buried piping and equipment in the Building Complexes and remaining impacted areas will be removed. The excavated areas will be backfilled and seeded for cover vegetation. Areas of remediation include contaminated surface areas identified during the characterization as well as areas under existing buildings, including buried pipelines and utility trenches.

8.3.2 Techniques Employed to Remove or Remediate Surface and Subsurface Soil

Trenches will be excavated beside the pipeline to be excavated to prevent striking the pipeline prior to removal. Soil and debris removed during excavation will be set aside for characterization and containerization.

8.3.3 Radiation Protection Methods

Radiation protection methods are described in detail in Section 10.

8.3.4 Procedures

Contaminated soil and subsurface utilities in the remaining impacted portions will be decommissioned following procedures and activities established in DP Revision 0 (Building Complexes 2, 5, 6A and 17):

8.3.5 Use of Procedures

Decommissioning activities involving licensed material will be conducted in accordance with approved, written procedures or RWPs, as described in Section 7.5 of License No. 06-00217-06.

8.3.6 Request for Additional Information

Additional information regarding characterization of waste lines, in process surveys and authorizing backfill was provided as a response to NRC request for additional information (ABB, 2004) and will apply for the remaining impacted portions:

> *Waste lines within the scope of the Decommissioning Plan will be characterized by scanning surveys of the industrial waste and hot waste lines prior to removal and the collection and analysis of at least two samples approximately every twelve feet where historical information or surveys indicate a likely potential for contamination exceeding the DCGLs to be present in sediments in the lines. It is planned that all hot waste lines will be processed as contaminated such that the pipe will be disposed of as radioactive waste along with its contents.*
>
> *For industrial system waste lines, if the sampling indicates that there is no residual contamination, the pipes will be opened to the extent necessary to perform a surface scan using nominally 100 cm^2 scintillation or gas proportional detectors and a wipe survey of every square meter of surface area. 100% surveys of the interior bottom of the line will be performed and if contamination is found then 100% surveys of the top interior portion will be conducted. Exterior pipe surveys will be conducted if contaminated soil is detected during the initial soil scanning of the trench. The scanning and removable surface contamination levels will be compared with the building surface release levels from Regulatory Guide 1.86 and if less than those values the piping will be disposed of as normal construction debris waste. For sanitary sewer lines and storm drains the same survey methodology will be used except scanning will be performed for 10% of the interior surface area with biased surveys performed at locations where material build-up appears. No scans or surveys of the remaining utilities (MDC Water Lines, Fire*

Protection water, Lines, etc.) will be performed unless elevated soil concentrations are detected during excavation of the surrounding soils.

If the scanning shows levels greater than building surface release levels, a determination will be made as to the cost-effectiveness of decontaminating pipe versus disposing of the pipe as radioactive waste.

The Decommissioning and Decontamination project scope calls for the removal of all underground utilities within the Buildings 2, 5, and 17 Complexes. Utility components that are located underground and will need excavation prior to removal include the following:

- *Potable water lines and piping*
- *High Temperature & Chill Water lines*
- *Fire protection System Water piping and hydrants*
- *Electrical feed lines to the building complexes*
- *Storm drains and associated effluent piping*
- *Sanitary Sewer piping and associated manholes*
- *Industrial Waste Piping and Associated manholes*
- *Radioactive Waste Piping and associated manholes and clean-outs*

The Site processes and use history show that the soils surrounding these utilities do not contain radioactive contaminants significantly in excess of background soil concentrations. It is expected that at certain junctures near manholes and piping joints for the Industrial and Hot (radioactive) waste lines, leakage of high concentrations of radioactive materials into the surrounding soil may have occurred. Consequently excavation of the Industrial and Hot Waste lines and associated manholes will be performed in a manner that identifies these points where leakage may have occurred and removes any soil where the resulting contamination may exceeds the designated DCGL.

The remaining utilities such as storm sewers and the sanitary sewer system piping would not have contained sufficient radioactive material such that due to a leak, the surrounding soil should not significantly exceed background soil concentrations for the D&D project. Surveys will still be performed for these excavations with the primary purpose of the surveys being worker protection and detection of any concentration anomalies. Interior surveys of these pipes will be conducted as described in the above RAI question concerning characterization of waste underground waste lines.

For excavations involving the Industrial and Hot Waste lines, soil will be monitored using a NaI detector (2" x 2" or 1" x 3") during the excavation of the soil from ground level to the top of the piping. It is not expected that any contaminants or elevated soil concentrations will be located in these upper surface soils. If elevated activity is detected at count rates that exceed a value indicating 50% or greater of the designated DCGL, then a soil sample will be collected and analyzed with the on-site gamma spectroscopy system.

Pipe joints and areas where leakage appears to have occurred due to soil discoloration, odor, etc. will be marked for further evaluation once the piping is removed. After pipe removal, scanning of the bottom of the trench will be performed to identify areas where activity in excess of 50% of the soil DCGL exists. When found those areas will either have the soil removed and disposed of as contaminated radioactive waste or further sampling and analysis with the on-site spectroscopy system will be made. Scanning, while intended to cover the entire bottom of the trench, will emphasize locations where the pipe joint and leakage markers were placed during the initial excavation activity. In addition, at least one sample every 100 linear feet of trench will be collected at a location where the highest elevated count rate can be found in the trench and in the associated excavated soil spoil pile.

It is expected that additional, independent surveys by regulatory agencies and the CE/ABB oversight contractor will be made during this process. The trench will be back-filled with the original soil in the spoil pile once all soils exceeding the DCGLs have been removed from the trench and associated spoil piles.

For excavations that involve utilities other than the industrial and hot waste lines, scanning will be performed periodically during excavation to verify that there are no soil concentration anomalies. Where count rate values exceed twice the background count rates for that area and geometry, additional evaluations will be made to ensure soil DCGLs will not be exceeded.

8.3.7 Site Brook

Since activities within wetlands require special permits and procedures in order to minimize impact to the ecosystem, additional information regarding decommissioning activities in Site brook and adjacent Debris Piles will be provided in a future revision to the DP.

8.4 SURFACE AND GROUNDWATER

8.4.1 Remediation Tasks Planned for Ground and Surface Water

Uranium was detected in groundwater at two locations at the Site (MW-1201 and WP-1403S). However, at both of these locations the levels or activities were below the USEPA MCLs. Decommissioning activities include the removal of the source of contamination, and it is anticipated that the radiological concentration will decrease due to natural attenuation; therefore, no active groundwater remediation is anticipated.

8.5 SCHEDULES

The projected schedule for decommissioning, in Figure 8-1, identifies the principal decommissioning tasks included in the DP and an estimated time required to complete each task. The tasks are organized according to the planned work sequence. The dates are referenced to NRC approval of the DP Revision 1 and separate approvals of other submittals to the NRC (e.g., future DP Revision). If it is determined that decommissioning cannot be completed as outlined in the schedule, an updated schedule will be submitted to the NRC for approval.

9.0 PROJECT MANAGEMENT AND ORGANIZATION

ABB has contracted with a decommissioning contractor to perform the decontamination and deconstruction (D&D) of the remaining impacted portions at their CE Windsor Site. The decommissioning contractor has the overall responsibility for safe completion of all D&D activities.

9.1 DECOMMISSIONING MANAGEMENT ORGANIZATION

The functional organization for the completion of D&D activities includes the Project Manager, Site Manager, Radiological Controls Manager, Quality Assurance Manager, Radiation Safety Officer and support personnel, as required. Figure 9-1 shows the functional organization and its relationship to ABB and the corporate interface. Note that one person may hold more than one management position as staffing will fluctuate according to need during the project.

9.2 DECOMMISSIONING MANAGEMENT POSITIONS AND QUALIFICATIONS

9.2.1 Project Manager

The Project Manager (PM) has overall responsibility for the safe conduct of the CE Windsor Decontamination and Decommissioning Project. This individual provides the senior project leadership for implementation and execution of a project specific Quality Assurance program, a project specific radiological health and safety program, and for compliance with all local, state, and federal regulations. The Project Manager has further assigned these responsibilities to the Site Manager, the Radiological Controls Manager, and the Quality Assurance Manager.

9.2.1.1 Experience and Qualifications

The Project Manager will hold a degree in science or engineering or equivalent job-related experience and training as deemed appropriate by ABB, and shall have a minimum 15 years of experience, at least five years of which shall be in a project management role.

9.2.2 Site Manager

The Site Manager is responsible for operational aspects of decommissioning project performance. This individual is responsible for implementation and oversight of the decontamination and dismantlement activities including waste management and engineering support. The Site Manager is responsible for ensuring that Work Plans are approved and current for each specific work activity, and the labor workforce possesses and maintains the project, and site-specific training required to complete the project safely.

9.2.2.1 Qualifications and Experience

The Site manager will hold a degree in science or engineering or equivalent job-related experience and training as deemed appropriate by ABB and have at least ten years supervisory or management experience, with at least five years decontamination and decommissioning experience.

9.2.3 Radiological Controls Manager

The Radiological Controls Manager (RCM) is principally responsible for radiological health and safety, and regulatory compliance. This individual's duties also include implementation of a site-specific Health and Safety Plan (HASP). The RCM shall perform periodic safety inspections to ensure compliance with all regulatory requirements, review and validate analytical and air monitoring data, and perform required self-assessments. In addition, the RCM will develop, implement, and maintain a training matrix for personnel assigned to the project and ensure that required qualifications are maintained current through timely notification of training requirements and scheduled training to the Site Manager. The RCM will keep the RSO informed of issues related to the duties assigned to be performed on behalf of the RSO.

9.2.3.1 Experience and Qualifications

The RCM shall hold a degree in science or engineering or equivalent job-related experience and training as deemed appropriate by ABB, and have at least five years experience in areas such as radiation safety, radiation monitoring, emergency preparedness, industrial safety, and personnel exposure evaluation. The RCM shall have demonstrated a proficiency to conduct specified radiation safety programs, recognize potential radiation and chemical safety problem areas in operations and advise operation supervision on radiation protection matters. The RCM shall be capable of directing the surveillance activities of the Health Physics Technicians.

The duties and responsibilities of the RCM include, but are not limited to:

- Surveillance of overall activities involving radioactive material, including monitoring and surveys of all areas in which radioactive material is used.
- Determine compliance with rules and regulations, and license conditions.
- Monitor and maintain absolute and other special filter systems associated with the use, storage, and disposal of radioactive material.
- Provide necessary information on all aspects of radiation protection to personnel at all levels of responsibility, pursuant to 10 CFR 19, and 10 CFR 20.
- Proper delivery, receipt, and conduct of radiation surveys of all shipments of radioactive material arriving at or leaving the Site within the scope of this license, including proper packaging and labeling of that radioactive material.
- Distribute and process personnel monitoring equipment, determine the need for evaluation of bioassays, monitor personnel exposure and bioassay records for trends and high exposures, and notify individuals and their supervisors of exposures approaching maximum permissible amounts and recommend appropriate remedial action.
- Conduct or provide oversight of training programs and otherwise instruct personnel in the proper procedures for the use of radioactive material prior to use, at periodic intervals (refresher training) and as required by changes in procedures, equipment and regulations, etc.
- Supervise and coordinate the radioactive waste disposal program, including effluent monitoring and maintenance of waste storage and disposal records.
- Store radioactive materials not in current use, including wastes.
- Perform or arrange for calibration of radiation survey instruments.
- Maintain, in conjunction with the Site RSO, an inventory of all radioisotopes on-site and limit the quantity of radionuclides on-site to the amounts authorized by the license.

- Immediately terminate any activity that could pose a threat to public, workers or the environment.
- Supervise decontamination, renovation, material control, remediation, and decommissioning operations.
- Maintain other records not specifically designated above, e.g., receipt, transfer, and survey records as required by 10 CFR 30.51, “Records,” and 10 CFR Part 20, Subpart L, “Records” (guidance is provided in NUREG-1460, dated November 1992, “Guide to Reporting and Record Keeping Requirements”).
- Periodic meetings with and reports to project management, ABB management and the RSO.
- Designate and maintain a list of qualified supervisors and users of licensed materials. Qualified individuals will be identified through evaluation of previous job experiences, education, and/or site-specific training programs.
- Develop and maintain training programs in accordance with 10 CFR Part 19.12.
- Develop and maintain operational Radiation Protection procedures to ensure program implementation and compliance with regulatory requirements.
- Maintain records of licensed material accumulation and transfer as required to support inventory and accountability.

9.2.4 Quality Assurance Manager

The Quality Assurance Manager is responsibility for auditing the implementation and execution of Quality Assurance and Control procedures and to evaluate conformance with procedures and other requirements. The Quality Assurance Manager will also be responsible for supervising the project’s Document Control procedures. This individual is responsible for preparation, implementation, and oversight of the Self-Assessment and Audits procedures, including identification of deficiencies and improvements, corrective actions, and feedback.

9.2.4.1 Qualifications and Experience

The Quality Assurance Manager shall hold a degree in science of engineering or equivalent job-related experience and training as deemed appropriate by ABB and have a minimum of five years experience in management, with a minimum of two years experience in oversight and responsibility for quality assurance and quality control issues.

9.2.5 Radiation Safety Officer

The duties, responsibilities, qualification and experience requirements for the Radiation Safety Officer are defined in License No. 06-00217-06.

9.2.6 Nuclear Materials Manager

The Nuclear Materials Manager (NMM) has overall responsibility for the oversight and accountability of special nuclear material (SNM). The NMM processes transaction and balance reports and maintains a physical inventory of SNM. An inventory control protocol will be instituted for material to be removed during excavation that will ensure the licensing limit for U-235 is maintained and that no accumulated, excavated materials exceed this total value. Prior to collection and storage of additional U-235 material, written approval from the Nuclear Materials Manager or his designee must be obtained.

9.2.6.1 Qualifications and Experience

The Nuclear Materials Manager will hold a degree in science or engineering or equivalent job-related experience and training as deemed appropriate by ABB and have at least two years decontamination and decommissioning experience.

9.3 DECOMMISSIONING TASK MANAGEMENT

Activities involving licensed material shall be conducted in accordance with approved, written procedures as specified in section 7.5 of the license application, and/or radiation work permits (RWP). Decommissioning tasks shall be conducted through procedures or RWPs.

RWPs are managed in accordance with a written procedure. The procedure addresses request, initiation, development, issuance, and termination of an RWP. The RWP may be requested by the supervisor of a particular activity. The request will include a description of the activity to be performed and authorized users of the RWP. The RCM or RSO will initiate the RWP and provide a description on the RWP of existing and/or anticipated radiological conditions.

After initiation, the RCM or RSO shall develop the RWP. The development effort includes specific identification of the radiological conditions and radiological protection requirements (e.g. clothing, respiratory protection, dosimetry, monitoring, training). Also, hold points and special instruction may be described on the RWP. The development effort also includes creating a sign-in/out sheet for use by the authorized users. The development effort ends with approval of the RWP by the RSO or RCM. Following development, the RWP is issued. The RWP form may contain items such as the job description, location, known radiological conditions, protective clothing requirements, respiratory protection, dosimetry, training, HP monitoring requirements, and any other special instructions. Issuance includes a review of RWP with the authorized users, as required. A pre-job meeting may be prerequisite to issuance of the RWP. During use, a copy of the RWP is maintained at the worksite, and authorized users may be required to sign-in/out when participating in the subject activity, indicating their understanding of the requirements of the RWP. The RWP is terminated upon completion of the subject activity. Termination is identified by signature on the RWP and completion of a checklist indicating reason for termination and confirmation of final radiological survey of the activity or area. Upon termination of the RWP, a package is completed and filed. The package generally contains the RWP, RWP request form, sign-in sheets, applicable radiological surveys, and any other documents pertinent to the job. If radiological conditions or requirements change, appropriate changes to the RWP may be made by the RSO or RCM. Alternatively, a new RWP may be issued.

9.4 TRAINING

Training will be performed in accordance with Section 7.8 of License No. 06-00217-06.

10.0 RADIATION SAFETY AND HEALTH PROGRAM DURING DECOMMISSIONING

Occupational dose will be kept as low as is reasonably achievable. To this end, a radiation safety program has been established commensurate with the scope and extent of licensed activities at the Site. The following sections provide a description of the primary elements used to realize this commitment.

10.1 RADIATION SAFETY CONTROLS AND MONITORING FOR WORKERS

This program and associated operating procedures are the primary means used to administratively establish safe radiation work practices and ensure compliance with the requirements of the NRC.

10.1.1 Workplace Air Sampling Program

10.1.1.1 Collection

Concentrations of radioactive material in air will be determined, as needed, by sampling the air. Air sampling shall be conducted in accordance with or equivalent to the guidance provided in NRC Regulatory Guide 8.25, “Air Sampling in the Workplace”, July 1992. Breathing zone air samples (belt mounted pump with sample head affixed to worker’s lapel) will be the primary method of monitoring the worker’s intake of radioactive material. The samples will be collected under known physical conditions (e.g. filter, sample time, flow rate). The flow meters of air samplers shall be calibrated at least annually. Calibration shall also be performed after repair or modification of the flow meter.

Air samples will also be collected from general and localized areas when and/or where there is potential for generation of airborne radioactive material. These samples will be used to verify that the confinement of radioactive material is effective, and provide warning of elevated concentrations for planning or response actions. In each case, the sampling point will be located in the airflow pathway near the known or suspected release point(s). As necessary, more than one air sample location may be used in order to provide a reasonable estimate of the general concentration of radioactive material in air.

The RSO shall apply professional judgment and experience to identify air sampling appropriate for the specific situation. Such judgment will be based on historical air sampling and characterization results, quantity of contamination of the material being handled, potential for release of contaminants based on physical form and activity, type of confinement or containment, and other factors specific to the activity.

Air sampling of the workplace will also be conducted under the following two conditions:

1. Areas with removable contamination greater than 1,000 dpm/100cm^2 and the worker is actively working in the area for greater than one hour during that workday; or
2. Areas with total contamination greater than 1,000 dpm/100 cm^2 and the work involves invasive activities such as drilling, scabbling, digging, or otherwise causing the release of contaminants or contaminated material into the air.

As familiarity with work activities increases, the RSO may modify the aforementioned conditions. Any modification will be explained and justified in writing by the RSO.

10.1.1.2 Action Level and Limit

An administrative action level shall be established for breathing zone air samples of one derived air concentration (DAC); air sample results greater than this administrative action level shall be reported to the RSO. An administrative limit shall be established for breathing zone air samples of 12 DAC-hours per week; individual exposure greater than this action level shall require the individual to be restricted from work involving potential exposure to airborne radioactive material unless approved by the RSO.

10.1.2 Respiratory Protection Program

The respiratory protection program (program) provides guidance and instruction regarding protection of workers from occupational injury and illness due to exposure to airborne radioactive material. The program is implemented by written procedures. The program and implementing procedures are the primary means used to administratively establish safe respiratory protection practices and compliance with requirements of the NRC.

The program covers routine use of respiratory protection equipment. The functional areas of the program include medical evaluation, fit testing, selection, issue, inspection, use, cleaning, maintenance, storage, and training.

10.1.2.1 Medical Evaluation

Prior to the initial fit test, and at least every 12 months thereafter, an evaluation will be made of each worker required to wear respiratory protection equipment as part of the worker's duties as to whether or not the worker can wear the required respirator without physical risk. A worker will not be allowed to wear a particular type of respirator if, in the opinion of a physician, the worker might suffer physical harm due to wearing the respirator. A worker shall not be allowed to use a respirator without a current medical evaluation.

10.1.2.2 Fit Test

All workers required to wear respiratory protection equipment shall be required to successfully complete a fit test prior to initial use of the equipment. The fit test shall be repeated at least annually. A worker shall not be allowed to wear a respirator without a current successful fit test.

10.1.2.3 Selection

Respirators shall be selected from those approved by the National Institute for Occupational Safety and Health for the contaminant or situation to which the worker may be exposed. Health Physics shall select the respirator type. Selection shall be based on the physical, chemical, and physiological properties of the contaminant, the contaminant concentration likely to be encountered, and the likely physical conditions of the workplace environment in which the respirator will be used.

10.1.2.4 Issue

Workers may be assigned respirators for their exclusive use or they shall otherwise be issued by Health Physics. Respirators shall only be assigned or issued to workers qualified, with respect to the program, to use respiratory protection equipment. The type of respirator selected shall be documented on the Radiation Work Permit.

10.1.2.5 Inspection

All respirators shall be inspected with regard to operability before, and routinely after, each use, and after cleaning.

10.1.2.6 Cleaning

Respiratory protection equipment that is used routinely shall be cleaned after each use. Respiratory protection equipment that is used by more than one worker shall be cleaned and disinfected after each use. The need for cleaning shall also be based on contamination surveys of the work area and of the respiratory protection equipment.

10.1.2.7 Maintenance

Respiratory protection equipment shall be maintained to retain its original effectiveness. Replacement or repair shall be done only by experienced persons, with parts designed for the respirator. No attempt shall be made to replace components or to make adjustments or repairs beyond the manufacturer's recommendations. Reducing valves or admission valves on regulators shall be returned to the manufacturer or equivalent for repair.

10.1.2.8 Storage

Respirators shall be stored to protect against dust, sunlight, heat, extreme cold, excessive moisture, or damaging chemicals. Respirators shall be stored in dedicated carrying cases or cartons that protect from dirt and damage.

10.1.2.9 Training

All workers required to use respiratory protection equipment shall be instructed in the content and applicability of the program and implementing procedures, and especially in the proper use of the equipment and its limitations. Refresher training shall be conducted annually. A worker shall not be allowed to use a respirator without current successful completion of training.

10.1.3 External Exposure Determination

Individual monitoring devices shall be provided to workers who require monitoring for external exposure pursuant to 10 CFR 20.1502(a). External monitoring shall be conducted in accordance with or equivalent to NRC Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses", July 1992.

External exposure monitoring, when required, generally is accomplished using thermoluminescent dosimeters worn on the front of the upper torso. For work areas where the external radiation field is non-uniform and external monitoring is required, extremity dosimetry may also be issued to the worker. Radiological surveys may be performed to supplement personnel monitoring when work is being performed where workers are required to be monitored.

Dosimeters shall be processed at least quarterly by a vendor accredited by NVLAP.

Work restriction shall be implemented for any worker reaching 50% of the annual limits of 10 CFR 20.

10.1.4 Internal Exposure Determination

Individual monitoring shall be provided for workers who require monitoring of the intake of radioactive material pursuant to 10CFR 20.1502(b). Monitoring of intake shall normally be conducted by use of air samples, particularly of the breathing zone. Internal dose shall be determined by converting airborne concentrations to intakes in accordance with NRC Regulatory Guide 8.34 "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses", July 1992.

When a potential or actual condition exists where the worker(s) could have received an unmonitored intake of radioactive material, and cannot otherwise be estimated, the intake shall be determined by measurements of quantities of radionuclides excreted from or retained in the body. These measurements shall be made consistent with the guidance provided in NRC Regulatory Guide 8.9 "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program", July 1993.

Determination of radiation dose to the embryo/fetus shall be performed in accordance with NRC Regulatory Guide 8.36 "Radiation Dose to the Embryo/Fetus", July 1992.

Work restrictions shall be implemented for any worker with an intake in excess of 50% of the applicable limit in 10 CFR 20. Work restrictions shall be implemented for any worker with an intake in excess of 50% of the chemical toxicity limit for soluble uranium.

10.1.5 Summation of External and Internal Exposures

Results of internal and external monitoring shall be used to calculate total organ dose equivalent and total effective dose equivalent to workers for which monitoring is required. Summation of internal and external doses shall be performed in accordance with NRC Regulatory Guide 8.34 "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses", July 1992.

10.1.6 Contamination Control Program

Contamination control shall be managed for exposure control and monitored by radiation surveys in accordance with approved procedures.

10.1.6.1 Exposure Control

Personnel exposure to radioactive material will be controlled by application of engineering, administrative, and personnel protection provisions. The priority of application will be descending with respect to their order of description below.

A. Engineering - Engineering controls will be used, as practicable, to minimize or prevent the presence of uncontained radioactive material. Engineering controls will predominantly be comprised of containment, isolation, ventilation, and decontamination.

B. Administrative - Administrative controls will be used to control work conditions and work practices. Administrative controls will predominantly be comprised of the following:

i. Access control: Routine access to work areas will be limited to personnel necessary to accomplish tasks or activities. Access will also be controlled with respect to training and use of specified personnel protection equipment.

ii. Postings and barriers: Postings will be used to inform personnel of relevant hazards or conditions and associated access requirements. Barriers may be used to prevent unauthorized access.

iii. Procedures: Written procedures may be used to describe specific radiation protection requirements necessary for tasks that involve radioactive material.

iv. Radiation Work Permits: The requirements for Radiation Work Permits (RWP) are described in Section 9.3. RWPs will be used to describe specific or special worker protection requirements for activities involving radioactive material and not covered by a procedure. RWPs may also be used in conjunction with a procedure.

v. Contamination Control: Action levels and limits for radiation surveys, described later in this section, will be used to control the levels of radioactivity on equipment and in areas.

C. Personal Protective Equipment - Personal protective equipment will be used to control personnel exposure to radioactive material when administrative controls are not sufficient and engineering controls are not practicable. Personal protective equipment may include head covering, eye protection, respiratory protection, impervious outerwear, gloves, and/or protective shoes or shoe covers.

10.1.6.2 Radiation Surveys

Radiation surveys will be performed to describe the radiation types and levels in an area or during a task, to identify or quantify radioactive material, and to evaluate potential and known radiological hazards.

The types of radiation surveys and their frequency are described in the following subsections.

A. Contamination Measurements - Measurements will be made of removable alpha, beta, and beta-gamma radiation, as applicable. The measurements will be made by wiping an area with cloth, paper, or tape. The radiation levels will be measured on the wipe. Contamination surveys shall be performed at the end of each workday where invasive demolition of contaminated surfaces was performed.

B. Radiation - Exposure rate measurements will be performed using an ion chamber or equivalent. Measurements will be made at approximately 30 centimeters. Measurements may also be made at contact.

C. Personnel - Personnel will be frisked prior to leaving access controlled areas.

D. Action Levels - Action levels are established to inform facility personnel when a situation needs to be evaluated so that corrective actions can be taken. Action levels are set so that corrective actions can be made before a regulatory limit is exceeded.

Exceedance of action levels requires investigation including evaluation of preventative and/or corrective action. The investigation, and documentation of such, is completed commensurate with the significance of the condition.

Radiation levels exceeding the values described in the following subsections will be reduced below the respective levels as soon as practicable.

i. Removable: The action level for removable alpha or beta-gamma radiation on a surface is 1,000 dpm/100cm^2.

ii. Exposure Rate: The action level for exposure rate is two millirem per hour at 30 centimeters.

iii. Personnel: The action level for personnel is 100 counts per minute above background.

E. Limits - Limits, as release criteria, are described in Section 14.1. The limits are administered such that when exceeded, action must be taken to reduce the levels or additional controls must be applied.

Items or areas will not be released for unrestricted use until the relevant limits are satisfied.

All accessible surfaces and areas that exceed the respective limits will be decontaminated on a timely basis. In no case will the delay to initiate control exceed one normal workday. In the case of personnel contamination, there will be no delay to initiate decontamination.

10.1.7 Instrumentation Program

Instrumentation shall be maintained that is capable of performing the radiation surveys and measurements of radioactive material required by regulation, license, and procedures. The types and management of radiation detection instrumentation is described in the following sections.

10.1.7.1 Type and Use

Examples of the types of instrumentation available and their intended use are described in Table 10-1.

10.1.7.2 Calibration

Calibration, maintenance, repair, and efficiency determination shall be performed according to written procedures, instructions, or other guidance documents reviewed and approved by the RSO, or by a commercial calibration service.

A. Frequency - Instruments shall be calibrated at least annually or following maintenance, repair, or adjustment likely to affect the primary calibration.

B. Radiation Energy - Calibration shall be performed using a source (s) providing radiation fields similar to those in which the instrument will be used.

C. Label - Each instrument shall be labeled or marked with the following information as applicable:

i. Unique identification (e.g. serial number),
ii. Initials or specific identifying mark of individual completing the calibration,
iii. Energy correction factors,
iv. Graph or table of calibration factors for each type of radiation for which the instrument may be used,
v. Instrument response to an identified check source,
vi. Unusual or special use conditions or limitations, and
vii. Date by which calibration is again required.

D. Standards - Calibration shall be performed using standard sources traceable to NIST. Gamma spectrometry system(s) measurements may be performed using high purity germanium radiation detectors that have been specifically characterized by the vendor to enable a sourceless efficiency calibration methodology. When this method is selected, the vendor's computer software performs a mathematical efficiency calibration without the use of sources.

10.1.7.3 Verification

Instruments in use shall be verified (checked) daily when in use to ensure that the instrument is in proper working condition. An instrument shall be removed from service if the source check is not within ±20 percent of the initial post-calibration value. Laboratory instruments used for radioactivity measurements are evaluated daily before use via check sources and efficiency checks. Maintenance or repair shall be performed if the daily source or background checks are not within prescribed ranges.

10.1.7.4 Sensitivity

Radiation detection systems shall be capable of detecting radiation of radioactivity significantly less than the respective limits. Measurement sensitivity shall be determined using the guidance of NUREG-1507 "Minimum Detectable Concentrations with Typical Radiation Safety Instruments for Various Contaminants and Field Conditions", 1997.

10.2 NUCLEAR CRITICALITY SAFETY

Based on the License limit of 325 grams Uranium 235, there are no nuclear criticality safety concerns within the scope of the D&D. An inventory control protocol will be instituted for material to be removed during excavation that will ensure the licensing limit for U-235 is maintained and that no accumulated, excavated materials exceed this total value. In addition, for fissile exempt materials, a ratio of fissile to non-fissile material mass based criteria, will be utilized to handle, store and transport large amounts of low concentration SNM contaminated materials with no nuclear criticality safety concerns or mass limits.

10.2.1 Request for Additional Information

Additional information regarding nuclear criticality safety was provided as a response to NRC request for additional information (ABB, 2004) and will apply for the remaining impacted portions:

A criticality safety protocol will be instituted for material to be removed during excavation to ensure that license possession and control limits for U-235 are not violated. The safety protocol will provide essential control measures, summarized below, which ensure license possession and control limits are not exceeded for U-235, and that criticality safety is assured. By controlling license possession limits, there is no potential for criticality. Under these conditions, U-235 mass values will remain well below those required for criticality, even for ideal geometry and moderation conditions.

- *The fundamental principle of the safety protocol is control of U-235 mass. The NRC, through a study by Oak Ridge National Laboratories, has determined that there is no potential for criticality at U-235 volumetric concentrations of less than 1,900 pCi/g and less than 10% enrichment. Such material, once verified, will need only to meet license possession limits for U-235.*
- *Suspect U-235 high activity areas will be examined prior to significant material disturbance to determine whether significant suspect material is likely to be present. (In this context, "significant" refers to material that might contain U-235 concentration greater than 1,900 pCi/g of U-235 and that might exceed 350 grams of U-235.)*
- *Concentration, enrichment, and U-235 gram weight determinations for specific areas will be made prior to significant disturbance of suspect material.*
- *An independent verification of the concentration, enrichment, and mass value calculations will be made prior to determining the U-235 gram weight of the material to be moved.*
- *Collection and containerizing of enriched material results in adding the U-235 gram weight to license inventory records. Collected U-235 mass is accounted for within the possession authority granted by NRC license. Prior to collection and storage of additional U-235 material, written approval from the Nuclear Materials Manager or his designee must be obtained.*
- *For materials that exceed the 1,900 pCi/g or 10% enrichment, movement of the material will be done in volume increments where the U-235 gram weight is less than 350 grams.*

10.2.2 Fissile Exempt Material

In accordance with license conditions for fissile exempt materials, once materials have been packaged for transportation and disposal they may be treated as fissile exempt. The fissile exempt ratio of at least 2,000 grams of solid non-fissile material for every gram of fissile material has been demonstrated by the NRC to be safe for handling, storage, transportation and disposal without any additional nuclear criticality safety controls, security, or mass based limits. The fissile exempt ratio is equivalent to a concentration 1,080 pCi/g of U-235.

10.3 HEALTH PHYSICS AUDITS, INSPECTIONS AND RECORD-KEEPING PROGRAM

The radiation safety program shall be subject to an annual audit and periodic inspections. Each are performed to determine if radiological operations are being conducted in accordance with regulations, license conditions, and written procedures.

An audit of the program shall be conducted annually. The audit shall be conducted by individual(s) not responsible for the design and/or implementation of the program. The audit will consider the basic functional areas of the program; e.g. Radiation Work Permits, Radiation Protection Procedures, radiological surveys and air monitoring, ALARA program, individual and area monitoring results, access controls, respiratory protection program, training, etc.

The audit shall be conducted in accordance with a specific audit plan developed by the auditor. A written report shall be generated upon completion of the audit describing the results. The report shall be distributed to Site management. As necessary, a written corrective action plan shall be prepared to address non-compliance issues. All corrective actions shall be tracked to completion. Once corrective actions have been completed, a written closure report shall be distributed to management documenting the completion of corrective actions.

The periodic inspections shall be conducted by the Health Physics staff. These inspections shall be routine reviews performed of operations and activities. The inspections shall normally be completed against a pre-established checklist. Checklists may be developed independently for differing periods; e.g. daily, weekly, monthly, etc. The checklist items shall usually be comprised of routine procedural requirements. Any findings discovered during the routine inspection shall be recorded on a tracking log. The log shall be maintained by the RSO. The log shall include a description of planned corrective action and date of completion of corrective action.

11.0 ENVIRONMENTAL MONITORING AND CONTROL PROGRAM

Decontamination and deconstruction activities will be conducted in a manner that protects the health and safety of the public and employees, and the environment. This includes development of programs and procedures that provide for monitoring and detection, and control of releases of radioactive material into the environment as a result of decontamination and deconstruction activities.

11.1 ENVIRONMENTAL ALARA EVALUATION

This section describes the requirements for establishing and maintaining releases of radioactive materials ALARA in effluents. The ALARA effort includes the following elements:

- Management commitment to ALARA, including goals;
- Procedures, process controls, and engineering controls;
- ALARA reviews and reports.

11.1.1 Management Commitment

ABB establishes management commitment to the environmental ALARA program, establishing an ALARA policy and implementation of ALARA goals.

Policy

It is policy of ABB to protect the public and the environment by maintaining releases of radioactive material to the environment ALARA.

The Nuclear Regulatory Commission's radiation protection regulations require that licensees use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve releases to the environment that are ALARA. This is a concept of public and environmental protection for which exposure to radiation, radioactivity, and releases of radioactive material are managed and controlled to levels below regulatory limits. ALARA practices involve the balancing of costs and benefits, not dose minimization. ALARA is a commitment to go beyond regulatory limits to the lowest practicable exposure or contamination level, taking into account social, technical, and economic considerations.

ALARA Goals

The project ALARA goals for effluents are 20% of the respective value in 10 CFR 20 Appendix B. The goals are not intended to set precedent or to be applied as a limit. The goals may be adjusted on the basis of review with regard to what may be ALARA for the particular circumstance.

Investigation Levels

The investigation levels for effluents are 50% of the respective value in 10 CFR 20 Appendix B. If exceeded, an investigation will be initiated. The results of the investigation will include identification of appropriate corrective actions.

Responsibility

The Radiation Safety Officer will be responsible for setting and periodically reviewing the ALARA goals. The RSO shall also be responsible for conducting investigations initiated due to exceeding an investigation level.

11.1.2 Procedures, Process Controls, and Engineering Controls

A description of the procedures, process controls, and engineering controls to maintain concentrations of radioactive material in effluents ALARA is provided Section 11.3.

11.1.3 ALARA Reviews

The effectiveness of the ALARA emphasis for the environmental monitoring and control of effluents is evaluated through the use of surveillances and audit(s).

Surveillances

The environmental monitoring and control program will be periodically subjected to surveillance by the RSO. The surveillances are intended to verify that the objectives of the program are being satisfied and that the program is generally effective. The surveillances may be limited in scope.

Audit

Program audits are scheduled and conducted under the Quality Assurance Program described in Section 13 of this Decommissioning Plan.

11.2 EFFLUENT MONITORING PROGRAM

Effluent monitoring is described in Section 10.1 of the current license. Effluent monitoring is conducted in accordance with a written procedure. The following sections describe any monitoring of discharges to the environment from local area of D&D activity (e.g. Building Complex 3) that will be provided with respect to this DP.

11.2.1 Baseline Concentrations

Baseline concentrations of radionuclides have been established by historical results of the Site environmental monitoring program.

11.2.2 Expected Concentrations

Concentrations of radionuclides in Site effluents are not expected to change as a result of the activities conducted under this DP. The effluent controls described in Section 11.3 are intended to realize this condition.

11.2.3 Physical and Chemical Characteristics

The physical and chemical characteristics of radionuclides in discharges from the activities conducted under this DP are not expected to be different than those of the current effluents.

11.2.4 Discharge Locations

Each area of D&D activity will have a known location(s) of discharge of storm water from the area. The location(s) will be identified in written procedures or work plans.

Emissions to air may occur from specific locations of activity and therefore will vary with the progress of the project. The emissions are expected to be ground-level or from interior of a structure. No discharges from stacks are planned.

11.2.5 Sample Collection and Analysis

Storm water

Storm water samples will be collected from at least one location at or near the boundary of the area of D&D activity. Collection of samples will be made more often (1) at the beginning of D&D activities until a predictable radioactivity composition is established, (2) whenever there is a significant unexplained increase in gross radioactivity, or (3) whenever a circumstance might cause a significant variation in radioactivity composition. The samples will be grab samples.

Storm water samples will be analyzed for gross alpha and gross beta activity. Samples may also be analyzed for gross gamma activity at the discretion of the RSO. The lower limit of detection for theses analyses will be not more than ten percent of the concentration limit listed in Table II of Appendix B to 10 CFR Part 20 for total uranium as gross alpha activity.

Air

Air samples will be collected from at least three locations at or near the boundary of the area of D&D activity. The sample locations will be chosen with consideration of meteorological conditions and D&D activity in order to sample the estimated maximum concentration. Air samples will be collected continuously for the duration of the activity being monitored.

Air samples will be analyzed for gross alpha and gross beta activity. Samples may also be analyzed for gross gamma activity at the discretion of the RSO. The lower limit of detection for these analyses will be not more than ten percent of the concentration limits listed in Table II of Appendix B to 10 CFR Part 20 for total uranium as gross alpha activity.

11.3 EFFLUENT CONTROL PROGRAM

This section describes the effluent control program and identifies the controls and actions necessary to meet the objectives of the program.

11.3.1 Procedures, Process Controls, and Engineering Controls

Available process options will be considered to control the concentration of radioactive material in effluents to the environment. Examples of process controls include recycling, leakage reduction, and modification of facilities, operations, and/or procedures. If further reduction in effluent concentration is necessary, available engineering options will be considered. Examples of available engineering options include filtration, adsorption, containment, and storage.

Process and engineering options will be implemented unless a review indicates that a substantial reduction in effluent concentration would not result or costs are considered unreasonable. A determination of reasonableness may be based on a qualitative review requiring the exercise of professional judgment for factors difficult to quantify. These factors could include nonradiological social or environmental impacts, availability and practicality of alternative technologies, and potential for unnecessarily increasing occupational exposures.

Effluent controls will be described in a written procedure or incorporated into operating procedures. The primary effluent controls used are expected to be dust suppression and erosion control.

Dust Suppression

Procedural controls, such as use of less aggressive decontamination or demolition techniques, will be used to minimize generation of fugitive emissions. Engineering controls, such as water spray or filtration, will also be utilized to control fugitive emissions and minimize visible dust.

Erosion and Sediment Control

Erosion and sediment controls may be temporary or permanent, depending on the duration of the activity and any specific objectives. Controls will be provided in accordance with best management practices, regulatory guidance, manufacturer's specifications, and good engineering practices. Temporary controls serve to minimize erosion and restrict the transport of sediment within the project area. Permanent controls serve to stabilize the Site with durable erosion control features to control sediment discharge, and protect nearby surface waters. Descriptions of erosion and sedimentation control practices that will be used during the project include:

Stabilization practices include the following:

- Minimizing disturbance areas;
- Minimizing and controlling dust;
- Stabilizing surfaces after final grading; and
- Permanent vegetative cover for disturbed areas not intended for other cover.

Structural features to control erosion and sedimentation include:

- Barriers to isolate areas of erosion and minimize sediment transport;
- Check dams in swale areas to minimize sediment transport;
- Erosion control blankets to minimize erosion due to concentrated flow prior to establishing vegetation;
- Construction of stabilized construction entrances to minimize the transport of sediment from project areas; and
- Stockpiles will be surrounded by sediment barriers.

Storm Water Management practices include:

- Maintaining runoff flow patterns and discharge locations similar to existing conditions; and
- Maximizing overland flow through vegetated areas.

11.3.2 Action Levels

The action levels for implementation or revision of effluent controls are those described previously in Section 11.1.1.

11.3.3 Releases to Sanitary Sewer

Releases to the sanitary sewer will only be made upon authorization of the RSO and based on review of applicable sample results.

The project may use self-contained sanitary systems serviced by a qualified vendor; releases will not be made to these systems.

11.3.4 Estimates of Doses to the Public

The doses to the public from radioactive material in effluents is estimated to be less than 10% of the applicable limits described in 10 CFR 20.

12.0 RADIOACTIVE WASTE MANAGEMENT PROGRAM

12.1 PROGRAM DESCRIPTION

The waste streams anticipated to be generated during decommissioning of remaining impacted portions of the Site will include, but are not limited to Resource Conservation and Recovery Act (RCRA) or Toxic Substances Control Act (TSCA) wastes, low-level radioactive waste (LLRW), mixed waste, sanitary waste, and demolition and construction debris. Waste treatment activities are not anticipated.

Metal items removed during demolition including beams, doors, framing, etc., will be radiologically surveyed and disposed of accordingly. Plans for radiological/chemical decontamination of metal items to be handled as radiologically-contaminated waste will be developed. Concrete will be broken to manageable size chunks and containerized. All demolition debris will be containerized, characterized, and disposed. For the remaining impacted portions, estimates of waste materials and volumes include 26,880 cubic feet of impacted building rubble, debris and D&D waste, plus 208,850 cubic feet of impacted soil for removal.

12.2 WASTE SEGREGATION

The waste streams generated as a result of CE Windsor Site D&D Project decommissioning activities will be segregated into like waste streams. The wastes will be packaged in accordance with the appropriate procedures and temporarily staged in designated areas of the CE Windsor Site. An effort will be taken to temporarily stage the wastes in the vicinity of the structure/component being dismantled. The material will be staged until placed in the appropriate shipping container. Wastes will be staged at the contractors staging and storage area, then moved to the proper disposal facility.

Co-mingling will be strictly controlled through labeling, containerization and physical segregation. Co-mingling will be prevented to the extent possible through the use of tarps, discrete barriers and containerization.

Access will be controlled to the segregated wastes by delineating specific entry and exit points to the waste staging/storage areas. In the event significant co-mingling occurs, the waste stream will be sampled in a manner that assures the analytical results are representative of that waste stream.

Waste segregation will be based on radiation surveys and analysis of bulk samples. Six basic techniques will be employed:

- Surface scans;
- Direct static measurements;
- In-situ gamma measurements;
- Material sampling;
- Paint sampling; and
- Removable contamination.

Prior to demolition, surface scans will be performed on buildings, equipment, and materials with portable radiation detection instrumentation (e.g. gas proportional detector, alpha/beta scintillation detector, gamma detectors). Direct static measurements will also be performed on surfaces of buildings, equipment, and materials with portable radiation detection instrumentation (e.g. gas proportional detector, alpha/beta scintillation detector, gamma detectors). In-situ gamma spectroscopy and spectrometry measurements may be performed on building surfaces and materials with NaI or HPGe detectors to identify facility-related gamma emitters. Samples of materials will be collected to identify and quantify radionuclide concentrations. Paint samples will be collected to quantify the amount of radioactivity in the paint. Removable contamination surveys will be performed on surfaces using standard smear techniques (e.g. filter paper for large areas or cotton swab for holes).

Both random and biased surveys and sampling will be performed. Biased sampling will be based on results of surface scans, walkdowns, historical use of the item or area, and professional judgment.

12.2.1 Management of Mixed and LLRW

Waste staging areas will be inspected at least every seven days and results documented on an inspection checklist. Inspection and documentation points shall include whether rainwater has accumulated in staging area, container condition, housekeeping, posted signs, container dates, contents, evidence of leaks, compatibility with wastes stored in them, and whether containers are closed (except when adding or removing wastes).

12.2.1.1 Mixed Wastes

Mixed wastes (LLRW/RCRA) or toxic substances wastes (LLRW/TSCA) will be managed in an area that meets the requirements of a LLRW Staging Area/RCRA Accumulation Area (90-day or Satellite) or LLRW Staging Area/PCB Storage Area according to waste characterization. Boundary markings and signs will identify these areas.

The types of solid mixed wastes expected to be generated during decommissioning operations included soil and/or sediments associated with removal and remediation of industrial waste lines in the vicinity of Building 6. Other solid mixed wastes could potentially be identified during removal of equipment and systems from the buildings. No liquid mixed wastes are expected to be generated. Approximately 800 cubic feet of solid mixed waste is expected to be generated in the remaining impacted portions. Radionuclides in the mixed waste are expected to range from less than 100 pCi/g to 3,000 pCi/g total uranium and less than 300 pCi/g of Co-60.

The mixed wastes will all be in a solid form with minimal residual liquids. Consequently, the waste will be contained in 55-gallon drums or similar DOT approved shipping containers. Packaging and shipping will be the same as that for other contaminated soils. At present ABB intends to send mixed wastes for disposal at EnergySolutions in Clive, Utah.

Mixed waste at the CE Windsor Site is primarily regulated by two federal agencies and one state agency: the Environmental Protection Agency and the NRC, and by the CTDEP. The CTDEP and USEPA have regulations that govern the generation, storage, and disposal of mixed waste. Because ABB is a large quantity waste generator, wastes can be stored for 90 days prior to shipment. For mixed wastes, if there is no viable treatment facility available, or if arrangements for disposal take longer than usual due to the radiological material contained with the waste, then it is permissible to store the wastes for longer periods of time provided written approval is obtained from the State of Connecticut DEP.

No permit is required for hazardous or mixed waste generators. The only requirement is for the generator to have a EPA Identification number. The ABB EPA Identification number is CTD001159557.

12.2.1.2 Low Level Radioactive Waste

The types of LLRW expected to be generated during decommissioning of the remaining areas includes materials and equipment from the buildings, soil, debris and subsurface utility piping. Soil contaminant levels are expected to range from less than 100 pCi/g to 110,000 pCi/g of total Uranium. Until actual excavation and analysis of excavated material occurs, it is not realistic to provide a more accurate quantification of the type and concentration of radionuclides. LLRW will be segregated from uncontaminated wastes to minimize the amount of LLRW generated. The staging area will have boundary marking and signs will be posted identifying the area.

LLRW liquids will be stored in an area that provides secondary containment of such size so as to contain 10 percent of the volume of all containers or the volume of the largest container (whichever is greater). Liquid wastes will not be true liquids but solid soils that contain water and are not expected to exceed more than 50 cubic feet. It is the intent of the project to try and blend these wastes where possible to limit the moisture content and provide additives such as WaterWorks to absorb and solidify any excess water in the waste containers. It is expected that liquid waste will be contaminated at concentrations less than 100 pCi/L total Uranium, and less than 5 pCi/L Co-60.

The storage, packaging, and disposal will be the similar to that of other wastes generated during the project.

12.2.2 Intentional Mixing of LLRW

During decommissioning, it may be necessary to intentionally mix waste in order to meet waste acceptance criteria (WAC) at off-site disposal facilities. The NRC has indicated that this is an acceptable practice as described in Commission Paper SECY-04-0035, *Result of the License Termination Rule Analysis of the Use of Intentional Mixing of Contaminated Soil*. This process will follow the guidance provided by the NRC in NUREG-1757 Volume 1 Revision 2, Section 15.13. Since uranium is not listed in Table 1 or 2 of 10 CFR 61.55, it is classified as Class A waste. Therefore intentional mixing of uranium wastes will not change waste classification since it will always be Class A waste. The following describes the process, materials, equipment, radiological surveys and sampling in order to meet waste acceptance criteria for off-site disposal.

Intentional mixing of soil or soil-like material will be conducted utilizing the simplest approach possible. The waste streams are not homogenous with respect to uranium concentrations and intentional mixing will not attempt to achieve homogeneity. Intentional mixing will be done to the level necessary in order to meet the fissile exempt criteria and WAC for off-site disposal. In this regard, material that has been intentionally mixed will not be significantly different than other wastes generated during decommissioning operations.

Heterogeneity of waste will be addressed by additional radiological surveys and sampling in order to demonstrate that fissile exempt and WAC have been met.

There are several options for proposed methods of intentional mixing of soil or soil-like material. The first is mixing elevated materials during excavation / removal operations with adjacent low concentration material. This option does not require additional equipment and is simply mixing of material as it is excavated and placed into a container. Elevated and low concentration materials can be mixed by excavating small amounts of each and then combining them with the excavator bucket prior to placing into a container. In addition, placement of materials into the container can be utilized to further mix them by alternating higher concentration and lower concentration materials. The second is to mix contents from one container (higher concentration) with material from another (lower concentration) into an additional container. This option involves placement of material from the higher and lower concentration materials alternatively into the additional container. These materials can also be mixed together inside the container prior to additional placement of materials from the other containers.

In some areas there may be small pieces of debris mixed with soil or soil-like material which could require intentional mixing. In this case, debris may be too small to be readily separable from soil, so representative surveys and sampling of the debris and soil mixture will be evaluated prior to mixing in order to verify that intentional mixing will meet fissile exempt and WAC. Similar to soil or soil-like materials, proposed methods for intentional mixing include mixing during excavation / removal operations with adjacent material or mixing contents from one container with another utilizing mechanical or hand tools. Heterogeneity of waste will be addressed by additional radiological surveys and sampling in order to demonstrate that fissile exempt and WAC have been met.

In other areas, there may be larger pieces of debris which could require intentional mixing with soil or soil-like material. Here the debris is large and may contain significantly elevated concentrations of uranium, such as an industrial waste line pipe section. In this case, the debris would be broken apart in order to facilitate mixing with soil or soil-like material. Proposed methods for intentional mixing include mixing contents from one container of soil or soil-like material into another containing debris utilizing mechanical or hand tools. Heterogeneity of waste will be addressed by additional radiological surveys and sampling in order to demonstrate that fissile exempt and WAC have been met.

Radiological surveys and sampling will be performed in conjunction with intentional mixing. Preremediation surveys and sampling will be performed to verify characterization data as well as to support nuclear criticality safety and SNM accountability. Areas will be identified that require intentional mixing. These areas will be then be evaluated for intentional mixing options. The preferred option is mixing with deeper soils or adjacent areas. If mixing with deeper or immediately adjacent areas will not meet fissile exempt and WAC, then mixing with lower concentration soil from other portions of the Site will be necessary. The appropriate low concentration LLRW will be determined from the available waste inventory.

12.3 WASTE PACKAGING

To the extent practical, the number of waste packages and the number of waste shipments will be minimized. Waste will be packaged in a manner that provides containment and protection for the duration of the anticipated storage period and until disposal is achieved, or until the waste is removed from the packaging. Waste packages will be marked such that their contents can be identified.

A commonly used packaging for shipment of radioactive material will be steel or plastic, open head or closed head, drum (e.g. 55-gallon). Steel boxes (e.g. B-25 type) will be used as a packaging for dry, bulk radioactive materials. Two other packagings that may be used for dry, bulk materials are intermodal containers and bulk polyethylene bags (e.g. supersacks).

All packagings will be inspected prior to use to ensure suitability for intended use.

12.4 DISPOSAL FACILITY

Radioactive waste generated during the decontamination and decommissioning will be transferred to a recipient who is properly licensed to receive such waste as specified in Section 11.3 of the License Application.

12.5 LICENSED MATERIAL INVENTORY AND ACCOUNTABILITY

Inventory and accountability of Licensed material is accomplished by keeping track of receipts and outgoing shipments of material in logs. Records will be maintained of Licensed material content for waste material accumulated and shipped.

13.0 QUALITY ASSURANCE PROGRAM

Decommissioning and decontamination activities will be performed under the provisions of the Decommissioning Contractor's, Quality Assurance Plan (QAP) and in the Characterization Plan (CP). The requirements and guidance contained in the QAP/CP are based on the principle that work shall be planned, documented, performed under controlled conditions, and periodically assessed to established work item quality and process effectiveness and promote improvement. The requirements described in the QAP/CP reflect the responsibilities assigned to management and personnel of all departments and their responsibility for planning, achieving, verifying, and assessing quality and promoting continuous improvement. The QAP/CP further delineate the quality contributions of all personnel and encourages their active participation in accomplishing the quality objectives. Key elements of the QAP/CP are described in the following sections.

13.1 D&D CONTRACTOR ORGANIZATION

The Decommissioning Contractor's President has ultimate responsibility for the implementation of the Quality Assurance Plan, but has delegated the day-to-day implementation of the Plan to the Project Manager. Appropriately trained and qualified personnel assigned to specific activities perform technical work, including quality-related work.

The Quality Assurance Manager is organizationally independent of cost and schedule to guarantee objectivity, and reports directly to the President. The Quality Assurance Manager is responsible for designing and implementing quality assurance procedures; for monitoring, auditing and inspecting to confirm that quality is being achieved; and for verifying that corrective action, when required, has been effective.

The President, Project Manager, Quality Assurance Manager, Site Manager, Radiological Controls Manager, Radiation Safety Officer, and field personnel have the authority to stop work for cause, real or suspected. Any employee can make work stoppage recommendations to any of those positions, and when there is an imminent danger all employees have the authority to stop work. Work will not resume until the concern is addressed and appropriately resolved.

Personnel and organizations not directly responsible for managing or performing the work verify the achievement of quality. Figure 9-1 shows the independence of the Quality Assurance Manager.

13.2 QUALITY ASSURANCE PROGRAM

Methods of quality assurance are used during the planning, design, procurement, installation, operation, maintenance, remediation and decontamination/decommissioning of structures, systems, components and project under the management of the Decommissioning Contractor. The Decommissioning Contractor's QAP/CP is documented and maintained in accordance with applicable standards and sound management practices. Activities affecting quality are identified and are made subject to the QAP/CP in the documentation.

13.3 DOCUMENT CONTROL

Documents that specify quality-related requirements and instructions are identified, reviewed, approved, issued, distributed, and maintained as controlled documents in accordance with written procedures. A listing of the type of documents to be maintained as controlled documents is shown in an appendix to the QAP. The controlled document list will be updated as needed, to ensure it is comprehensive, current, and complete.

Changes to controlled documents are reviewed and approved by the same organization that reviewed and approved the documents originally, or by other designated qualified organizations. Disposition of superseded and modified documents is controlled in accordance with written procedures. A master list of controlled documents is maintained to identify the current revision number of instructions, procedures, specifications, drawings, and procurement documents. The list is distributed periodically to those individuals or organizations responsible for maintaining the applicable controlled documents, to prevent the use of outdated or obsolete documents.

Appropriate controlled documents are available in work areas before initiation of and during the performance of activities affecting quality. This availability is verified periodically by Quality Assurance. Changes or revisions to controlled documents are verbally communicated to affected individuals and a required reading program assures awareness of the change.

13.3.1.1 Instructions, Procedures, and Drawings

Quality-related activities shall be prescribed by and accomplished in accordance with documented and approved instructions, procedures, or drawings. These instructions, procedures and drawings shall contain the necessary detail required by the activity and include or reference appropriate acceptance criteria.

13.3.1.2 Responsibility

Quality Assurance is responsible for verifying implementation of all quality-related work outlined in controlled documents, procedures, or drawings.

Employees and subcontractors are responsible for implementing the Quality Assurance Plan and applicable instructions, procedures, and drawings.

The cognizant managers and appropriate supervisors are responsible for developing and implementing all quality-related technical documents or procedures.

Technical supervisors are responsible for developing, securing approval for, conducting, and reporting on elements of the work program affecting quality in accordance with the Quality Assurance Plan and applicable instructions, procedures and drawings. The cognizant managers are responsible for ensuring development of all technical documents dealing with quality-related work under their cognizance, and for ensuring compliance with those documents.

13.3.1.3 Implementation

A written plan, procedures, and instructions governing implementation of the Quality Assurance Program shall be developed. These documents address requirements concerning scope and purpose, applicability, responsibilities, and records. They are issued and maintained as controlled documents. Approval of the Cognizant Manager and concurrence with Quality Assurance are required before these documents may be issued or revised.

Activities affecting quality are controlled and authorized by documents (e.g. procedures, instructions, drawings). These documents are reviewed as necessary by authorized personnel having appropriate technical, quality, and administrative expertise to ensure adequacy and completeness. Written procedures clearly outline the actions to be accomplished in the preparation, review, approval, and control of procedures, instructions and drawings.

Drawings initiated by subcontractors are controlled in accordance with the QAP. Project Management is responsible for the control of drawings developed by subcontractors. Quality Assurance verifies compliance through normal audit procedures.

Any errors or deficiencies in instructions, procedures, and drawings are corrected upon discovery. Revisions or changes are made, reviewed, approved, and documented in accordance with written procedures before the revision or change is implemented.

13.4 CONTROL OF MEASURING AND TEST EQUIPMENT

The Decommissioning Contractor's technical staff will use appropriate procedures to ensure adequate control of measuring and test equipment that affect site characterization and the quality of design, construction, or operation. The procedures describe calibration technique, frequency, maintenance, and control of measuring and test equipment.

Measuring and test equipment is labeled, tagged, or otherwise identified and documented to indicate the next calibration due date, as well as to provide traceability to calibration test data. Before measuring and test equipment is used, it is checked by the user to have a current calibration. Equipment is calibrated at specific intervals based on manufacturer's recommendations or on required accuracy and equipment history of drifting, precision, purpose, or any other characteristics that could affect accuracy. If a piece of equipment is found to be out of calibration, evaluations are made to determine the validity and acceptability of any measurements performed subsequent to the last calibration. If items are measured with equipment found to be out of calibration, the items are re-inspected.

Standards for calibration are determined with appropriate reference to nationally accepted standards, manufacturers' instructions, intended uses, and other factors. If national standards do not exist, the basis for calibration is documented. Calibrations are performed immediately prior to use when such action is necessary to maintain or ensure accurate measurements and tests.

Documented calibration records are maintained as Quality Assurance records, in accordance with applicable procedures. Calibration instructions are maintained as controlled documents.

13.5 CORRECTIVE ACTION

Corrective actions are accommodated through written procedures that implement an audit tracking system. Conditions adverse to quality are evaluated via the audit tracking system, and if found to be significant, are investigated to determine root causes, to decide on immediate corrective actions, to project preventive actions, and to define follow-up needs. The evaluations are documented within the audit tracking system.

Follow-up verification by the Quality Assurance Manager or designee ensures that the audit tracking system actions have been implemented in a timely manner and are effective. The Quality Assurance Manager monitors progress and closes audit tracking system actions in a timely manner.

The Quality Assurance Manager reports on audit tracking system actions pending and closed, and on trends related to Nonconformance Reports and Corrective Action Reports, at each Management Review meeting.

Documentation will be maintained of Corrective Action Reports, actions taken to resolve the condition, and any follow-up audits or actions.

13.6 RECORDS

A records management system for items with quality assurance requirements includes, in part, the following: operating logs, results of reviews, inspections, tests, audits, monitoring of work performance, and material analyses. Records also include closely related data such as qualifications of personnel, training, procedures, equipment records (including calibrations), evaluations and analyses of a quality-related nature.

The types and locations of quality assurance records are identified in a subject-oriented records list. Individual records are classified, designated, validated, and stored in accordance with written procedures. Quality assurance documents are traceable to relevant items and activities, and are identifiable and retrievable. Record retention is in accordance with applicable regulatory requirements.

13.7 AUDITS AND SURVEILLANCES

Audits and surveillances are planned and scheduled according to the type and status of work being performed. Unannounced audits and surveillances are performed as necessary.

The results of audits and surveillances shall be documented. Quality Assurance is responsible for ensuring that audit findings and observations are monitored and closed out in a timely manner. Audit results are documented and reviewed by D&D Contractor management personnel who are responsible for the audited area.

Management personnel take appropriate action to identify root causes, correct deficiencies, prevent recurrences, and determine impacts of audit findings in their area of responsibility. Follow-up actions are performed as necessary to ensure that appropriate corrective actions have been implemented in a timely manner and are effective.

13.8 PROGRAM CHANGES

Changes to the key elements of the Quality Assurance Program presented in this Decommissioning Plan will be submitted to the U.S. Nuclear Regulatory Commission for review and approval prior to implementation.

The NRC will be notified of any changes to the organizational elements within 30 days after the announcement of the change is made.

Editorial changes or personnel reassignments of a nonsubstantive nature do not require NRC notification.

14.0 FACILITY RADIATION SURVEYS

Radiological characterization survey results will be collected to provide sufficient information to permit planning for the CE Windsor Site remaining impacted areas remediation that will be effective and will not endanger remediation workers, to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected, and to provide information that will be used to design the final status survey.

14.1 RELEASE CRITERIA

The radiological criteria for release of materials for unrestricted use are described in Section 10.0 of the License Application, which is Table 1 of NRC Regulatory Guide 1.86. The unrestricted release criteria for buildings, soils and sediments that remain on-site will be in accordance with the DCGLs or NRC screening values as discussed in Section 5.

The release criteria for materials (e.g. equipment, components, etc.) will be evaluated by removable and/or scanning radiation surveys. The coverage of the release survey will be based on professional judgment including any applicable history or process knowledge but at least 10 percent of the surface area of the survey unit. The radionuclides present will be identified to determine the applicable release criteria. If survey results are greater than the release criteria the item will not be released, or a 100% scan will be completed of the survey unit. The location exceeding the release criteria will be decontaminated and resurveyed, or will not be released. Surveys for soils will be conducted in accordance with Section 14.4 of this Decommissioning Plan.

Release surveys will be performed and documented in accordance with written procedures.

14.2 CHARACTERIZATION SURVEYS

A building characterization plan was developed and implemented at the Site. The plan provides details of the characterization methods including measurement and sampling techniques, and measurement system quality issues. Measurement and sample collection, handling, and analyses were performed in accordance with written procedures to ensure precise, accurate, reproducible, complete, and comprehensive results. Characterization surveys for the remaining impacted areas are also complete. In the event that additional characterization surveys are needed, the existing plan may be revised.

The characterization surveys were conducted using the following guidance:

- NUREG-1575 “Multi-Agency Radiation Survey and Site Investigation Manual”, Revision 1, 2000;
- NUREG-1505 “A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys”, 1997; and
- NUREG-1507 “Minimal Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions”, 1997.

14.3 REMEDIAL ACTION SUPPORT SURVEYS

14.3.1 General

Radiation surveys and measurements (surveys) will be conducted to 1) support remediation activities, 2) determine when a survey unit is ready for the final status survey, and 3) provide updated estimates of parameters used for planning the final status survey. These surveys may include routine operational surveys conducted to support remediation activities. These surveys may be performed of building and equipment surfaces, and bulk materials. Areas or items that are expected to satisfy the DCGLs on the basis of remedial action support surveys will be identified as ready for final status survey.

14.3.2 Survey Design

Random and biased surveys will be performed. Biased sampling will be based on results of historical surveys, walkdowns, historical use of the item or area, and professional judgment.

14.3.3 Conducting Surveys

Remedial action support surveys will include one or more of the following: surface scans, direct static measurements, in-situ gamma measurements, material sampling, paint sampling, and removable sampling.

Surface scans and direct static measurements will be performed with portable radiation detection instrumentation (e.g. gas proportional detector, alpha/beta scintillation detector, gamma detectors). In-situ gamma measurements using NaI or HPGe detectors may be performed to identify radionuclides. Samples of material and paint may be collected to identify and quantify radionuclide concentrations. Measurements of removable radioactivity will be performed on surfaces using standard smear techniques.

14.3.4 Evaluating Survey Results

Survey data (e.g. surface activity levels and radionuclide concentrations in media) will be converted to standard units and compared to the DCGLs. If results of these surveys indicate that remediation has been successful in meeting the DCGLs, decontamination efforts will cease. Otherwise, additional remediation will be performed.

14.4 FINAL STATUS SURVEY DESIGN

14.4.1 Overview

The NRC regulation, 10 CFR 30.36(g)(4)(iv) requires a description of the final radiation survey. The final status survey design will be submitted at the completion of remediation, or when the design of the final radiological survey has been completed. The final status survey report will be submitted after the final radiological survey has been completed. For license termination, the final design will encompass consideration for the entire Site. The final status survey for the CE Windsor Site will be designed using the guidance contained in NUREG-1575 “Multi-Agency Radiation Survey and Site Investigation Manual” (MARSSIM). The surveys will demonstrate that the residual radioactivity in each survey unit satisfies the applicable criteria for unrestricted release, as described in Section 5. The surveys will provide data to demonstrate that radiological parameters do not exceed the established DCGLs.

<u>Survey Design</u>

The survey designs will begin with the development of data quality objectives (DQOs). The DQOs will be developed using guidance provided on the DQO Process in Appendix D of

MARSSIM. On the basis of these objectives and the known or anticipated radiological conditions at the Site, the numbers and locations of measurement and sampling points used to demonstrate compliance with the release criterion will be determined. Finally, survey techniques appropriate for development of adequate data will be selected and implemented.

Radionuclides of Concern

The radiological contaminants at the CE Windsor Site are the uranium isotopes (U-234, U-235, U-238) and commercial power plant byproduct materials (e.g. Cs, Co and Mn). The uranium is enriched in U-234 and U-235 above naturally occurring levels.

Uranium

The suite of radionuclides found in enriched uranium (EU) is fixed by the physical and chemical processes used to produce the EU and by the laws of physics describing radioactive decay. The same physical laws govern the relative concentrations of these radionuclides, making their proportions at a given U-235 mass enrichment known with a high degree of certainty. Isotopically, EU does not vary substantially by batch for a given percent enrichment, assuming that the EU was produced using the same enrichment (isotope separation) process.

The CE Windsor Site did not receive or process UF_6 at any time, and it is unlikely that any contamination from the Gaseous Diffusion Plants (Tc-99, Np-237 and Pu-239) would have been carried over to the Site. Due to the physical characteristics of the gaseous diffusion process, any trace amounts of the heavier elements (Np-237 and Pu-239) would have been diminished during the enrichment process, and would have tended to have been relegated to the depleted uranium tails. The lighter elements, such as Tc-99, which may have been carried over into the enriched uranium would have dropped out during the fuel fabrication process when the UF_6 was converted into uranium dioxide for the fabrication of fuel pellets. Since no UF_6 was processed on-site, Tc-99 contamination is unlikely.

The uranium enrichment percentage does impact the relative concentrations of the uranium isotopes in the fuel mixture. In gaseous diffusion, the smaller U-234 atoms are more readily enriched than either U-235 or U-238 atoms. Likewise, U-235 atoms are more readily enriched than U-238 atoms. Uranium 234 having a radioactive half-life more than 3 orders of magnitude shorter than U-235, dominates the total uranium radioactivity for enrichments greater than naturally occurring levels.

The HSA (Harding ESE, 2002) indicates that some depleted and natural uranium materials were present and used for research and development on-site. However, quantities of these materials used on-site are overwhelmed by the amount of EU used. Typical commercial grade LEU fuel stock was supplied to CE at about 3.5% enrichment. HEU with enrichments greater than 90% was used to manufacture special nuclear fuels for the Federal government. Thus, a range of potential uranium isotopic ratios might occur on the Site, and could vary from one location to another, depending upon the deposition source.

Byproduct Materials

Radionuclides produced in the operation of a nuclear reactor are classified as byproduct materials, as they are the “byproduct” of a nuclear reaction. There are two subcategories of isotopes collectively classed as byproduct materials. They are described by their production mechanisms: 1) fission products, and 2) activation products. The nuclear fuel services work performed by CE and later by Westinghouse at the Site involved the repair, maintenance and testing of reactor plant components. Since nuclear fuel itself is clad, or jacketed, to prevent a significant release of fission products, the principle radionuclides associated with plant components handled at the Site are activation products. For purposes of derivation of the DCGL it is unimportant to further distinguish between these, but it does serve to understand the byproduct material isotopic mixture.

Isotopes found in byproduct materials are generally characterized by short half-lives and beta decay mechanisms. The shortest-lived isotopes rapidly decay away and are essentially gone before components can be removed from a reactor plant for service. After one year only a small number of the longest-lived radionuclides remain in potentially significant quantities. Therefore, by the time any soil cleanup will begin, no new byproduct radioactive materials will have been introduced at the Site.

The isotopes in the byproduct radionuclide mixture are best characterized by the waste stream profile associated with the nuclear facility support work performed at the Site. The waste stream profile identifies 5 isotopes that contribute 1% or more to the total byproduct activity. These include:

Isotope	% of Total Activity
Co-60	73.2%
Cs-137	1.1%
Fe-55	2.9%
Mn-54	2.7%
Sb-125	5.0%
Others	15.1%

The clear dominance of cobalt-60 with respect to percentage of contribution to total activity is amplified by the fact that Co-60 is by far the most potent dose producer among the byproduct nuclides present.

Derived Concentration Guideline Levels (DCGL)

Section 5.0 of this report provides the DCGLs used to design the surveys. For the purpose of the final status surveys, the DCGLs of Section 5.0 represent contamination conditions that are approximately uniform across the survey unit and will be specifically referred to as $DCGL_W$.

A separate DCGL will be derived for small areas of elevated activity and will be specifically referred to as $DCGL_{EMC}$ (elevated measurement comparison). The method for determining the values for the $DCGL_{EMC}$ will be to modify the $DCGL_W$ by a correction factor that accounts for the difference in area and the resulting change in dose. The area factor is the magnitude by which the concentration within a small area of elevated activity can exceed the $DCGL_W$ while maintaining compliance with the release criterion. (If the $DCGL_W$ is multiplied by the area factor, the resulting concentration distributed over the specified smaller area delivers the same calculated dose.)

Classification of Areas based on Contamination

All areas of the Site do not have the same potential for contamination and, accordingly, do not need the same level of survey coverage to demonstrate that residual radioactivity in the area satisfies the applicable criteria. The surveys were designed so that areas with higher potential for contamination receive a higher degree of survey effort.

The survey designs fall into one of two categories, non-impacted and impacted. Areas that have no reasonable potential for residual contamination are designated as non-impacted areas and are not provided any level of survey coverage. Areas that have some potential for containing contaminated material are designated as impacted areas. Impacted areas are subdivided into three classes according to known or suspected levels of contamination with regard to the classification guidance of MARSSIM. Specific and thorough consideration was given to Site operating history and/or known contamination based on Site characterization efforts:

- Class 1 areas: These areas have, or had prior to remediation, a potential for radioactive contamination (based on Site operating history) or known contamination (based on radiological surveys). Areas that are suspected of containing contamination in excess of the DCGLs shall be classified as Class 1.
- Class 2 areas: These areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the DCGLs.
- Class 3 areas: Areas that are potentially impacted but not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the DCGL, based on Site operating history and previous radiological surveys. These are areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification.

Class 1 areas have the greatest potential for contamination and, therefore, receive the highest degree of survey effort, followed by Class 2, and then Class 3 areas. Areas may be further subdivided into units in accordance with the guidance in MARSSIM or to better facilitate assessment of the area.

In some cases, there is not enough information to differentiate what class is appropriate for areas identified as impacted. In these areas, additional characterization data will be needed to better determine the extent of radiological contamination. These areas and classifications include the structures, systems, and open lands discussed in the Historical Site Assessment Report.

14.4.2 Investigation Levels

Radionuclide-specific investigation levels will be used to indicate when additional investigations may be necessary. The investigation levels will also serve as a quality control check for the measurement process. The investigation levels to be used at the Site are provided in Table 14-1.

14.4.3 Instruments and Methods

The instruments to be used, calibration, and operational checks to be used during the FSS are the same as those described in Section 10.1.7.

Measurement methods used to generate data during the surveys can be classified into three categories commonly known as scanning surveys, direct measurements, and sampling. These techniques will be combined in an integrated survey design.

Scanning Surveys

Scanning will be performed to identify areas of elevated activity that may not be detected by other measurement methods. Scanning will be performed of structure surfaces and land areas. Structure surfaces will be scanned for both alpha and beta/gamma radiations. Land areas will be scanned for gross gamma radiation. The types of instruments used for scanning listed in Section 10.1.7.

Direct and Removable Measurements

Direct and removable measurements will be made to determine average activity in a survey area or unit. Direct and removable measurements will only be made of structural surfaces. Direct and removable measurements will be limited to alpha and beta/gamma measurements. The types of instruments used for direct and removable measurements are listed in Section 10.1.7.

Sampling

Sampling will be limited to land areas. Samples of soil will be collected and analyzed for the radionuclides of concern, as applicable. The analysis technique and typical detection limit for each radionuclide of concern is provided in Section 10.1.7.

14.4.4 Reference Areas

The reference areas used for the conduct of the final status surveys for land areas will be as described in the Site characterization report. The reference area for structural surfaces may be determined at the time of the survey.

14.4.5 Reference Coordinate System

Reference coordinate systems will be used to facilitate selection of measurement and sampling locations, and to provide a mechanism for relocating a survey point. Land area scanning surveys and soil sample locations will be referenced to the Connecticut State Plane Coordinate System.

14.4.6 Summary of Statistical Tests

Measurements from a survey unit will be compared to equivalent measurements from the reference areas. In general, the comparison will be whether the survey unit exceeds the reference area by more than the DCGL. The Wilcoxen Rank Sum (WRS) statistical test will be used to evaluate the data from the final status surveys. Alternatively, if background (reference areas) concentrations are appreciably lower than the DCGL then the Sign Test will be the statistical test for compliance evaluation.

In addition, an elevated measurement comparison (EMC) will be performed against each measurement in a Class 1 unit to ensure that the measurement result does not exceed the specified investigation level; i.e. the $DCGL_{EMC}$. If any measurement exceeds the $DCGL_{EMC}$, then additional investigation will be completed regardless of the outcome of the WRS test.

14.4.7 Control and Handling of Samples for Laboratory Analysis

Sample collection will be conducted in accordance with a written procedure. Laboratory analyses will be conducted in accordance with a written procedure. A written chain-of-custody procedure will be used to ensure integrity of samples and data from sample collection through data reporting.

15.0 FINANCIAL ASSURANCE

A Decommissioning Funding Plan was submitted by letter dated March 30, 2001. The Decommissioning Funding Plan includes a site-specific cost estimate for decommissioning, and a description of the means for adjusting the cost estimate periodically over the life of the facility. Revision 1 of the Decommissioning Funding Plan was submitted by letter dated November 12, 2003. This addressed the Commercial Areas D&D activities, which are complete.

A Surety Bond and Standby Trust Agreement, as permitted by 10CFR30.35(f)(2) was submitted by letter to NRC Region I Document Control Desk on April 19, 2000. A revision of the Decommissioning Funding Plan to update the decommissioning cost estimate for license termination that will allow unrestricted release for the entire CE Windsor Site was submitted by letter dated September 19, 2008. Decommissioning Funding Plan Revision 2 was accepted by the NRC on letter dated October 14, 2008. ABB is revising financial assurance instruments following the guidance in NUREG-1757 Volume 3.