

DECOMMISSIONING PLAN

**CE WINDSOR SITE
WINDSOR, CONNECTICUT**

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

(OCTOBER 15, 2003)

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Abbreviations and Acronyms

| | |
|------------------------|---|
| ACE | Advanced Critical Experiment |
| ACM | asbestos containing material |
| AEC | Atomic Energy Commission |
| ALARA | as low as reasonably achievable |
| AOC | Area of Concern |
| bgs | below ground surface |
| BNFL | British Nuclear Fuel, Ltd. |
| BONUS | Boiling Nuclear Superheat |
| BWR | Boiling Water Reactor |
| CE | Combustion Engineering, Inc. |
| CMU | concrete masonry unit |
| COC | constituent of concern |
| CP | Characterization Plan |
| 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 |
| DU | depleted uranium |
| ELI | evaporator line investigation |
| EM | electromagnetic |
| EMC | elevated measurement comparison |
| EU | enriched uranium |
| FBR | Fast Breeder Reactor |
| FCE | Flexible Criticality Experiment |
| FIDLER | Field Instrument for the Detection of Low Energy Radiation |
| 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 |
| HWOOCR | Heavy Water Organic Cooled Reactor |
| ICM | Interim Corrective Measure |
| KAPL | Knolls Atomic Power Laboratory |
| LEU | low enriched uranium |
| LLRW | low-level radioactive waste |
| MACTEC | MACTEC Constructors, Inc. |
| MARSSIM | Multi-Agency Radiation Survey and Site Investigation Manual |
| MDC | Metropolitan District Commission |
| MSL | mean sea level |
| NaI | sodium iodide |
| NFM | nuclear fuel manufacturing (er) |
| NPDES | National Pollutant Discharge Elimination System |
| NRC | U.S. Nuclear Regulatory Commission |
| NUSU | Nuclear Superheat |
| NVLAP | National Voluntary Laboratory Accreditation Program |

Abbreviations and Acronyms

| | |
|--------|--|
| ORISE | Oak Ridge Institute for Science and Education |
| PCB | polychlorinated biphenyls |
| PDU | process development unit |
| PM | project manager |
| PPE | personal protective equipment |
| PVC | polyvinyl chloride |
| PWR | Pressurized Water Reactor |
| QAP | Quality Assurance Plan |
| QC | quality control |
| 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 |
| SHSRSO | Site Health, Safety, and Radiation Safety Officer |
| 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 |
| VCA | Voluntary Corrective Action |
| WRS | Wilcoxon 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", published in September 2000.

Decommissioning of the CE Site is complicated by the fact that certain buildings and areas on the Site are being addressed by the U.S. Army Corps of Engineers (USACE) under the Formally Utilized Sites Remedial Action Program (FUSRAP). These buildings and areas were used by the U.S. Atomic Energy Commission (AEC) prior to the early 1960s, when the first nuclear fuel manufacturing license was issued to CE. FUSRAP buildings and areas (some of which include co-mingled licensed and FUSRAP residues) are discussed in this DP. However, based on the Memorandum of Understanding between the NRC and U.S. Army Corps of Engineers (USACE), such areas are not subject to NRC jurisdiction during remediation activities; responsibility for the remediation process remains with the USACE. Since ABB does not control this process, a full presentation of information for these areas and buildings, following the guidance in NUREG-1727, is not provided in this DP. Rather, one must be referred to the USACE and documents that may be produced by the USACE for FUSRAP areas and buildings. Nonetheless, when the FUSRAP remediation effort is complete, ABB will demonstrate that NRC dose criteria are met for unrestricted license termination of the entire CE Site.

The areas in which commercial licensed activities were conducted include primarily Building Complexes 2, 5, and 17. With the exception of commercial and FUSRAP areas, other areas and buildings on the Site have been determined by characterization or historical site assessment to be unaffected, or will require verification only, or may require minor remediation only for unrestricted use. All areas of the site were investigated where contamination could be present above the proposed DCGLs based on the HSA. The areas are grouped into three classes:

- The four building complexes specified for Commercial D&D (Buildings 2, 5, 6A and 17 Complexes)
- The seven FUSRAP Areas (laydown area, drum burial pit, clam shell pile, debris pile, Site brook, Building 6 and waste lines, and Building 3)
- Other potentially contaminated areas (equipment storage yard, former gravel pit, sludge piles, area at the former WWTP, small pond, and southeast parcel)

For the Building Complexes, remediation is planned under the DP. Remediation will include decontamination of buildings, demolition of all structures within the complexes to ground surface, removal of floor slabs and footings 3 (or 4) feet below ground surface, removal of all underground utilities and any soils under the utilities impacted above the DCGLs, and final status survey.

The seven designated FUSRAP Areas may require remediation; however, the USACE is still studying these areas and making decisions about how to execute the cleanup.

Investigation of the other potentially contaminated areas has recently been completed and none have been found to contain radiological contamination above the DCGLs. These areas will be included in the final status survey.

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 Prospects, Inc.
501 Merritt 7
Norwalk, CT 06856-5308

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

ABB Prospects, Inc.
C/o John Conant
CEP 880-1406
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 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.

Building Complexes 2, 5, and 17 have been vacated and emptied of equipment, machinery, furniture, etc., and interior systems have been decontaminated and dismantled. Above grade structures in the Building 2, 5, and 17 Complexes are being deconstructed under a current License amendment, down to, but not including the removal of the building slabs and foundations. CE will decontaminate and dismantle the Building 6A Complex as well, although it currently houses Site maintenance operations. Dismantlement is expected within 24 months of the approval of the DP.

Other radiologically impacted areas have been identified at the CE Windsor Site, however, these areas are not part of the commercial operations. These areas have been accepted by the Formerly Utilized Site Remedial Action Program (FUSRAP) which is under the regulatory authority of the United States Army Corps of Engineers (USACE). The presence of FUSRAP materials is a result of the historic operations that supported the Naval nuclear programs under the Atomic Energy Commission. Figure 4-11 shows all of the impacted areas at the CE Windsor Site, and identifies each area as: Commercial D&D; FUSRAP; or area for final status survey.

1.4 DECOMMISSIONING OBJECTIVE

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

Areas designated under FUSRAP will be remediated under the authority of the USACE, and are not addressed by this DP except by reference, for completeness.

1.5 SITE SPECIFIC DCGLS

CE manufactured nuclear (uranium) fuels for both commercial reactors and for government reactors. As a result, there is shared responsibility between CE and the Federal government for the cleanup of environmental contaminants at the Site. The Federal government has assigned responsibility for the cleanup of residual radioactivity at the Site as a result of government sponsored or contracted activities to the U.S. Army Corps of Engineers (USACE) under the Formerly Utilized Site Remedial Action Program (FUSRAP).

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

Radiological contamination at the Site is being addressed following guidance in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NUREG-1575) through the preparation of a Historical Site Assessment and the development of the site-specific concentrations of licensed material in soil as derived concentration guideline levels (DCGLs). The DCGLs are presented in the Derivation of Site-Specific Soil DCGLs Report

(DCGLs). The DCGLs are presented in the Derivation of Site-Specific Soil DCGLs report (ABB, 2003), re-submitted for review and approval by letter dated October 15, 2003.

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 D of NUREG-1727 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." In this regard, 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, a simplified analysis (cost vs. soil activity levels, possible benefits and costs relating to decommissioning, and a determination of residual radioactivity levels that are ALARA) has been developed. The analysis was completed in the context of NUREG-1727, Appendix D. The analysis is based upon release criteria derived from site-specific dose modeling; i.e. the DCGLs referred to in Section 5 of this report. The ALARA analysis is presented in Section 7.0 of this report.

1.7 START AND END DATES

The project to remediate and dismantle Building Complexes 2, 5, and 17 is scheduled to begin upon NRC approval of the Decommissioning Plan and DCGLs and will continue for approximately 1 year. Building 6A is currently in use as a Site maintenance facility. It will be remediated as necessary and surveyed for release. Building 6A will be dismantled within two years of the NRC approval of the Decommissioning Plan and DCGLs. All other areas are subject or potentially subject to the FUSRAP project. The schedule for the FUSRAP areas is still being determined by the USACE.

1.8 POST-REMEDATION 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 submittal includes a request to amend NRC License No. 06-00217-06 to incorporate the Decommissioning Plan.

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.

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 5 and 17. Liquid radiological waste was processed 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.

CE'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 ft². 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 UO₂. The warehouse was used to store supplies and sealed drums of UO₂ 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, 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 being completed in 2002. 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.

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.

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; 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 residential and commercial developments to the east. The northwest corner of the property is bordered by the Rainbow Reservoir portion of the Farmington River. Within the property is a 10.6-acre enclave formerly owned by CE until 1960, known as S1C. This area is currently owned by the United States Government and is referred to as the former Knolls Atomic Power Laboratory (KAPL). This area has been decommissioned and demolished.

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 unlikely given: 1) the current land-use trend; 2) the current community growth, planning, and development strategies of the local municipality; and 3) the economic value of the land for sustained commercial use. Although unlikely, it is reasonable and credible to consider that the land might be used for locating residential dwellings in the future.

Subsistence farming is also 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.

Groundwater Flow Velocity

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

There are no known extant populations of Federal or State Endangered or Threatened Species that occur at the Site. This was confirmed by letter from the United States Department of the Interior, Fish and Wildlife Service to the United States Nuclear Regulatory Commission, Region I, dated August 21, 2002.

Based on information received from the State of Connecticut Department of Environmental Protection there is one freshwater mussel species of special concern, *Ligumia nastuta*, Eastern pond mussel, which occurs in close proximity to the Site.

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.

From 1955 to 1961 CE operated under contracts with the Atomic Energy Commission (AEC). Because of this work, several areas of the Site have been accepted as impacted areas by the FUSRAP under the USACE. Figure 4-11 shows areas that contain FUSRAP material.

The FUSRAP areas, as identified on Figure 4-11, include the Site Brook Area, the Industrial Waste Lines, the Waste Storage Pad (Woods Area), the Drum Burial Pit, clamshell pile, debris area, and Buildings 3 and 6 and the surrounding grounds.

The following descriptions include summaries of the types and extent of radioactive material contamination in all media at the Site: buildings, systems and equipment, surface and subsurface soil, and surface and groundwater. For completeness, both commercial and FUSRAP areas are presented.

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.

The Building 3 Complex, including the surrounding grounds, has been accepted for remediation under the FUSRAP.

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 includes 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 by the green shaded areas of 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 being completed in 2002 and 2003. License Amendment No. 53 authorized the decontamination and deconstruction of the above-grade structures of buildings in Building Complexes 2, 5, and Building 17.

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

The Building 2 complex was constructed in the mid 1950s and is located in the central portion of the Site, west of the former S1C facility (Figure 3-2). It is comprised of Buildings 1, 1A, 2, and 2A, and was used to support nuclear fuel research and development and commercial nuclear power plant outage and support services. An additional modular building, 2M, was used strictly for office space and was not used for licensed material or operations. It was removed in 2001.

Building 1

Physical Description. The Building 1 footprint is approximately 4,140 ft². Building 1 is a one-story concrete structure with Transite siding and a sheet metal roof. The building consists of two reinforced concrete vaults, a reactor control room, a reactor component assembly area and various rooms designed to support reactor research and development activities (Figure 4-1). The building includes a ventilation system to control airborne radioactivity and industrial and sanitary drains.

Historical Use. Building 1 was constructed in 1956. Building 1 was originally built to support the research and development of the Site's first nuclear reactor experiments from 1956 to 1959. Criticality experiments, including the Flexible Criticality Experiment (FCE) and the Submarine Combustion (S1C), were conducted in support of the various AEC contracts. The Site's first criticality experiment, the FCE, was located in the reactor vault of Building 1. The FCE utilized clad, HEU plates as fuel elements. The FCE was eventually

dismantled and moved to Building 2. By 1960, Building 1 was decontaminated and released for occupancy.

Since the 1960s, Building 1 was used for various activities including, x-ray analyses, photography and as a storage area for radioactive materials. In the early 1970s, Building 1 was then used as a radioactive materials storage area and as an area to refurbish reactor plant inspection and servicing equipment under the Byproduct Materials License 06-00217-06. A 1978 letter to the NRC for the renewal of license 06-00217-06 describes Building 1 as a storage facility for packaged radioactive materials.

Radiological History and Potential Contaminants. The radiological history and known or potential radiological contaminants for each room or area associated with the building are summarized in Table 4-1. In 2002, systems and components inside the building were being decontaminated and dismantled.

Building 1A

Physical Description. The footprint of the building is approximately 5,959 ft². The building is a one-story metal shed placed on a concrete slab. The building is divided into two storage areas (north/south)(Figure 4-2). There is a steel mezzanine located in the north storage area. There are no systems associated with this building (e.g., ventilation, sanitary, industrial).

Historical Use. Building 1A was constructed in 1976. The building has always been primarily for the storage of radioactive materials, and to a lesser extent a radioactive materials work area, where radioactive materials were handled within a contamination cell.

Radiological History and Potential Contaminants. The radiological history and known or potential radiological contaminants for each room or area associated with the building are summarized in Table 4-1. In 2002, systems and components inside the building were being decontaminated, dismantled, and disposed or staged for disposal.

Building 2

Physical Description. The Building 2 footprint is approximately 10,000 ft². Building 2 is a two-story concrete structure with Transite siding (Figure 4-3). Building 2 contains a 30-foot deep basement with 2 reactor cells constructed of reinforced concrete and a two story high bay (west high bay). A high-density concrete vault is located in the center of Building 2 on the first floor. The second story of the main building contains offices and two mechanical rooms to control climate and airborne radioactivity levels. There is a 60-foot, steel/metal addition (tower) located above the east high bay area. Buildings 2 and 2A are interconnected via a reinforced concrete sliding door. The building is also equipped with industrial and sanitary drains.

Historical Use. Building 2 was constructed in 1956. Building 2 was originally built to support the research and development of the Site's first nuclear reactor experiments from 1956 to 1961. Criticality experiments including the Advanced Critical Experiment (ACE), S1C assembly, Nuclear Superheat (NUSU), and Boiling Nuclear Superheat (BONUS) were conducted in support of the various AEC contracts. Approximately 1,000 separate runs, averaging a one-hour duration of criticality were achieved. The projects included research in temperature, pressure, and reactor component design. Throughout the 1960s, Building 2 continued as a research facility on the Heavy Water Organic Cooled Reactor (HWOCR),

Pressurized Water (PWR), Boiling Water (BWR), sodium cooled and Fast Breeder Reactor (FBR) systems. The research was conducted primarily in Cell 2.

In 1965 the 60-foot tower was added to the east high bay area to support reactor research on the HWOCR. The tower enclosed the cooling loops for the HWOCR design with a crane attached to the tower roof. The tower was initially equipped with steel floors at multiple levels that were open on the north end to allow for crane access.

In the late 1960s, Cell 1 was dedicated to seismic research and testing of fuel assembly bundles and various components to support the development of commercial nuclear power.

Since the early 1970s, the rest of Building 2 has been used exclusively to provide outage support services to nuclear power plants.

Radiological History and Potential Contaminants. The radiological history and known or potential radiological contaminants for each room within, or areas surrounding the building, are summarized in Table 4-1. In 2002, systems and components inside the building were being decontaminated and dismantled.

Building 2A

Physical Description. Building 2A is approximately 5,115 ft². The building is a one-story high bay constructed with metal siding situated on a concrete slab. Building 2A contained multiple control zones on the north wall and one control zone on the south wall (Figure 4-4). The control zones were wooden structures wrapped in Herculite, and were removed by Westinghouse in August 2001. Building 2A was built to the existing height to accommodate the reactor servicing equipment. An equipment pit is located within the control zones on the north wall. The pit was used for a large reactor fuel-servicing device called a "Super Stand". Building 2A is interconnected to Building 2 via a reinforced concrete sliding door.

Building 2A is equipped with a double HEPA ventilation system for the control zones. The building is also equipped with a sanitary drain line.

Radiological History and Potential Contaminants. The radiological history and known or potential radiological contaminants for each room, or areas surrounding Building 2A are summarized in Table 4-1. In 2002, systems and components inside the building were being decontaminated and dismantled.

Building 2 Complex Outdoor Storage Area

Building 2 Complex Outdoor Storage Area is a fenced area that surrounds the buildings (except Building 2M) forming a controlled access area. Equipment was packaged for shipment and stored in this area. In 1996, the fenced area was expanded to comply with the new 10 CFR part 20 dose criteria. After the lay down and storage area was expanded and the fence relocated, soil samples were collected from the area. Samples were analyzed via gamma spectroscopy for U235, Cs137, Co60, Th228 and Th232. No sample contained activity that was significantly greater than background.

4.1.2 Building 5 Complex

The Building 5 Complex is located within the southern portion of the Site and includes Buildings 5, 15, 16, and 18 (Figures 3-2 and 4-5).

Building 5

Building 5 was used to house the Research and Development Laboratories and Nuclear Fuel Manufacturing. Building 5 supported both government and commercial nuclear projects.

Physical Description. Building 5 consists of three wings, covering a total area of approximately 60,164 ft². The building is a one-story structure constructed of concrete with Transite siding, and a steel roof deck. The entire building floor was originally tiled with asbestos. Parts of the building were carpeted during the 1990s. The asbestos tiles remain in many areas.

Building 5 was constructed as a research and development facility and therefore it is equipped with multiple process control systems. These systems include multiple ventilation units for controlling the building climate and contamination levels, and a water supply and drainage system for controlling process waters, industrial waste, sanitary waste and radiological waste.

Historical Use. Building 5 was constructed in 1957 as a research and development laboratory to support AEC contract work. The building operated under contracts (non-licensed) with the AEC from 1957 to 1961 using natural, depleted, and enriched uranium.

Building 5, as described in a 1961 license application to the AEC, consisted of a Materials Development Laboratory, a Nuclear Manufacturing Facility, an Engineering Development Laboratory, and associated support operations.

The Materials Development Laboratories were located in the North Wing of Building 5. The Metallurgy Section was equipped for metallography, mechanical testing, non-destructive testing, x-ray diffraction, vacuum and inert atmosphere melting, heat-treating, brazing, welding, radiography, powder processing, vibratory compaction, swaging, and ceramics processing. Research and Development projects included: UO₂ powder densification, solid solution ceramic fuels, irradiation damage, ternary phase determinations, ceramic fuel thermal characteristics, welding techniques for cluster fabrication, and UO₂ fuel element cladding. The Chemistry Groups in the North and South Wings performed emission spectrography, corrosion testing, fuel burn-up evaluation, radiochemistry, uranium waste solution analyses, QC and process chemistry analyses, x-ray fluorescence of ceramic fuels, water chemistry, and isotopic analysis of uranium and boron. The Materials Development Laboratory also investigated the design and fabrication of long-life stainless steel and UO₂ matrix fuel.

During the 1960s, the Nuclear Manufacturing Facility was located in the central wing of Building 5 and included a Fuel Element Loading Shop, a machine and rolling area, an autoclave and braze area, a welding area, an abrasive cleaning area, a pickling area, an x-ray, and ultrasonics test area, an inspection area and an office area. The pickling, abrasive cleaning, autoclave and braze, and machine shop and rolling area were located in the high-bay space (37 feet high) and were serviced by a 10-ton crane. The Fuel Element Loading Shop was a completely enclosed area constructed of concrete blocks with a steel, interlocking type of roof located within the high-bay area.

The Nuclear Manufacturing Facility was equipped to provide a variety of services necessary in the development and manufacture of precision reactor components such as control rods, control rod drive mechanisms, and reactor core internals. Equipment was available to roll

and fabricate certain alloys of metals used for control rods and other special nuclear components. The facility was completely equipped to fabricate reactor fuel assemblies utilizing low-enriched uranium. The fuel assemblies were of the type employed in pressurized and boiling water reactor power plants. The assemblies could utilize either pellet or powder (vibratory compacted) type.

The Engineering Development Laboratory was located at the west-end of the center wing in Building 5. It occupies approximately 4,100 ft². The mechanical development group of the Engineering Development Laboratory conducted development and test operations in the fields of thermodynamics, fluid mechanics, mechanisms development, stress analysis, dynamic corrosion, power plant systems evaluation, and component evaluation. Equipment in the laboratory included a 4-inch 400 gpm high pressure/temperature test loop, a 4-inch and a 6-inch hydraulic test loop, and a 14-inch by 110-inch stainless autoclave. The area was serviced with a 10-ton crane in the high-bay and all laboratory services, including hot waste collection. The electrical development group performed work in the experimental evaluation of reactor control systems and nuclear/non-nuclear instrumentation.

During the early 1960s, Building 5 completed its largest fuel manufacturing project. CE fabricated a reactor for the Puerto Rican Water Resource Authority called the Boiling Nuclear Superheat (Bonus) Reactor. Documentation from August, 1961 states that the four assemblies of the boiler zone contain natural uranium (0.71 percent U235), and the other surrounding geometrically similar elements had the uranium enriched to 2.4 percent by weight. The total uranium weight in this region of the reactor in the form of U-elements was 2.8 tons. The sintered UO₂ in the superheater fuel element was enriched to 3.5 percent U235. The uranium operations consisted of filling the rods with UO₂ pellets that were purchased from an outside supplier.

In December 1961, CE was issued a license (SNM-551) for the possession of low-enriched uranium. This was revised in 1965 for the possession of uranium enriched to 15 percent. In 1968, license SNM-1067 replaced the original SNM-551 license with the addition of the new Building 17 Nuclear Fuel Fabrication Facility. Since the issuance of SNM-1067 and the Byproduct Materials License 06-00217-06, Building 5 has always supported Nuclear Fuel Fabrication and the Nuclear Outage Services operations.

Radiological History and Potential Contaminants. The radiological history and known potential contaminants are summarized in Table 4-2. In 2002, systems and components inside the building were being decontaminated and dismantled.

Previous Radiological Investigations. In 1996, CE performed radiological surveys and collected volumetric samples from the following locations throughout Building 5.

- roof ventilation exhaust units
- ceiling tiles
- exhaust systems
- chiller/heater units
- wall samples, paint, dust, insulation
- sink traps

- wall air handling units
- autoclave exhaust duct (highbay)
- sumps, pits & pipe chase
- industrial and hot waste piping systems

The investigation was focused on locating HEU from operations performed in the late 1950s to early 1960s. The sample results did not indicate the significant presence of HEU; however, the results indicated the presence of the following:

- 24 samples contained LEU from 3 to 221 pCi/g;
- 8 samples contained DU or natural uranium from 450 to 6,700 pCi/g; and
- 38 samples contained Byproduct Contamination (Co60 and Cs137) from 9 to 11 pCi/g.

The majority of the contamination was located in discrete locations such as the ventilation system, pipe penetrations and service space between the suspended ceiling and the underside of the roof.

In preparation of using the Metropolitan District Commission (MDC) for wastewater discharge, CE investigated the contribution of water going from the Building 5 Complex to Building 6 to evaluate the amount of liquid radiological wastewater that needed to be re-routed to the evaporator in Building 6. A summary of the locations in Building 5 which had previously contributed radiological waste to Building 6 is provided in an internal memo.

During the summer of 1995, the maintenance department performed several plumbing projects to eliminate non-radiological sources of liquid from entering the liquid radiological-waste processing system. The project required the removal or capping of contaminated waste lines and the installation of new drain lines connected to the sanitary waste system. During the project, several areas were identified as having both loose and fixed LEU contamination.

Former Building 13 Storage Shed

The original license application for SNM-1067 shows the former Building 13 located immediately west of Building 15. It describes Building 13 as a Nuclear Storage Building. This building no longer exists. The building appears in several figures associated with SNM-1067 amendment applications from 1967 to the early 1970s, but was not listed as a location for licensed operations.

As part of the HSA, it was determined that Building 13 was used for the storage of radiologically clean tools and chemicals. The facility and grounds crew used the tools and chemicals stored here for routine maintenance and pest control.

Building 15 Carpentry Shop

Building 15 was the Carpentry Shop. The building is a one-story metal shop on top of a concrete slab. The building is approximately 2,000 ft².

Building 15 is shown on the original SNM-1067 license application figures and described as a Nuclear Storage Building. This building never contained nuclear materials.

Building 16 Boronometer Test Area

Physical Description. Building 16 is a one-story metal structure situated on top of a concrete slab. The building is located at the west end of the Building 5 north wing. Building 16 is approximately 6,500 ft². The building contains a tool crib, stock cage, office area, Boronometer test area, and a shop area.

The building includes industrial and sanitary drains.

Historical Use. Building 16 was constructed in 1975. The building was originally used to test boronometers. Personnel from Buildings 5 and 18 used the tool crib and stock cage to maintain the test loops in the high bay.

Radiological History and Potential Contaminants. The use of radioactive materials in Building 16 was limited to sealed neutron (e.g., RaBe, PuBe, AmBe) sources and DU (see Table 4-2).

Sealed neutron sources were used routinely in Building 16 during the Boronometer operations. There is no evidence that any sealed source ever leaked in Building 16.

DU was also handled in Building 16. The test loop in Building 18 used DU to simulate the actual mass of a fuel bundle assembly. The DU powder was loaded into the fuel rods in the shop of Building 16. Also, prior to manufacturing fuel rods in Building 17 with LEU, the operations were tested using prototype rods containing DU. The DU rods were later unloaded and cut up in Buildings 5 and 16. As a result the industrial drains in Building 16 were clogged with DU. In 2002, systems and components inside the building were being decontaminated and dismantled. This task did not include the drain system.

Previous Radiological Investigations. Routine surveys of Building 16 do not reveal any contamination, however; in 1995, an excavation was planned to connect the existing industrial waste line serving Building 16 to the sanitary sewer system and the MDC. The purpose of this project was to eliminate sources of non-contaminated water from entering the liquid radiological-waste system. The project is summarized below.

A trench was excavated between the buildings starting near the entrance to Building 16 and traveling south. The industrial waste line was to be connected to the sanitary sewer line at 224F of Building 5. A 1-foot section of the industrial waste line was removed from below the alley-way between Buildings 5 and 16 (approximately 5 feet bgs), and volumetric soil samples were collected. Gamma spectroscopy results of the pipe sediments and soil indicated the presence of Co60 and Cs137 in duplicate samples. Insoluble radioactive materials were not allowed to enter the MDC (sanitary sewer system) and therefore the project was halted. Both ends of the section of pipe, where the piece had been removed, were sealed with concrete. The contaminated piping remains in place.

Building 18 Thermal Hydraulic Test Loop

Physical Description. The Building 18 footprint is approximately 3,600 ft². The building is a four story, steel frame high bay with a basement level and Transite siding. It contained a scale model reactor test loop. The test loop included standard CE fuel assembly bundles, charging pumps, reactor coolant pumps, a 40,000 gallon cooling pool, a control room and reactor components necessary to simulate the fluid mechanic operations of a commercial nuclear reactor.

The building includes industrial and sanitary drains.

Historical Use. Building 18 was constructed in 1968. The building operations included simulating the mechanical testing of actual (commercial) reactor components using fuel bundles composed of DU. The simulator operated at temperatures and pressures slightly greater than an actual facility (600°F, 2,200 psi) to test component reliability. The simulator held four actual size fuel assembly bundles.

Radiological History and Potential Contaminants. Routine surveys of Building 18 indicate that this building is not contaminated with uranium. In 2002, systems and components inside the building were being decontaminated and dismantled.

Previous Radiological Investigations. In August of 1995, surveys were performed on the Building 18 sump area. Results of the surveys indicated that the sump is contaminated with small quantities (less than 6 pCi/g) of byproduct contamination (primarily Co60 and Cs137).

4.1.3 Building 17 Complex

Building 17 was primarily a nuclear fuel manufacturing facility producing fuel for the commercial nuclear power industry. The building is located near the center of the Site (Figure 3-2) immediately north of the Building 5 Complex.

Physical Description The dimensions of Building 17 are 120 feet x 300 feet (36,000 ft²). Building 17 is a one-story concrete block structure with a concrete floor, corrugated asbestos siding and a poured gypsum roof deck approximately 30 feet high (Figures 4-6 and 4-7).

The front section of Building 17 is approximately 4,800 ft² (120 feet x 40 feet) has a concrete floor, exterior concrete block and a poured gypsum roof deck approximately 11 feet high. The front section of Building 17 was an office area.

The interior of Building 17 consisted of a shop area (approximately 12,000 ft²) including receiving and storage in addition to a fuel loading shop (Bay C). The remaining 20,000 ft² was separated by a wall and was designated as the machine shop area. The machine shop area included an autoclave, a welding area, an abrasive cleaning area, a pickling area, an inspection area, and a machining area.

The main roof of Building 17 was originally comprised of a form board (forming the ceiling of the Pellet Shop), above which approximately 2 inches of gypsum forms the majority of the roof. The gypsum layer is covered with a weather proofing skin comprised of multiple layers of paper and tar. A roof membrane was later placed over the original roof.

Building 17 is equipped with ventilation systems for controlling the building climate and airborne radioactivity levels. The building was also equipped with drain lines for radiological waste, industrial waste and sanitary waste.

4.1.3.1 Building History

Building 17 was constructed in 1967. In 1968, Building 17 was licensed under SNM-1067, as a NFM, to permit the manufacture of fuel rods/elements and assemblies/bundles utilizing partially enriched uranium.

The Nuclear Manufacturing capabilities are described in the SNM-1067 application dated January 2, 1968. The NFM was equipped to provide a variety of services necessary in the

development and manufacture of precision reactor components such as fuel rods and assemblies, control rods, control rod drive mechanisms, and reactor core internals.

Building 17 was shut down in 1993 when CE's commercial nuclear fuel manufacturing was consolidated to another licensed facility in Missouri.

In 1994, British Nuclear Fuel, Ltd. (BNFL) began the decontamination and decommissioning of Building 17 and in 1996, the operations in Building 17 were halted. It was determined that Building 17 would resume nuclear operations as a support facility for the commercial nuclear power industry. BNFL had decontaminated Bays A and B, and partially decontaminated Bay C before the project was terminated. The building was refitted for new activities under the NRC byproduct License No. 06-00217-06.

The building renovations included new exterior and interior walls for the radiologically clean areas (Bays A and B) and reconstructed exterior walls (concrete block) constructed around Bay C to provide shielding for the commercial support operations. Bay C was fitted with multiple control zones to provide containment for the new operations.

In 1998, Building 17 was reopened as office space for engineering support and as work space for nuclear fuel outage and field operation support. Operations in Building 17 were halted in 2001, and the equipment was removed.

4.1.3.2 Radiological History and Potential Contaminants

Building 17 began fuel manufacturing (EU approximately 2.5 to 4.2 percent) in 1968. The following paragraphs describe the radiological history by area within and around Building 17. In 2002, systems and components inside the building were being decontaminated and dismantled.

Pellet Shop (Bay C)

Initially, UO₂ pellets were received at the CE Windsor Site from an off-Site supplier, for fabrication into fuel assemblies. Later, to support production requirements, UO₂ powder was received on Site. The UO₂ powder entered the building into Bay C at the west end from Building 21. The powder was mixed, prepped, pressed, and sintered to form pellets. The pellets were loaded into rods and the rods were sealed by welding.

The pellet production continued until 1990. In 1990, pellet production was moved off-Site and the pellet shop was decontaminated. An area, approximately 80 feet by 40 feet, of the old pellet shop was scabbled until it was free releasable. The machine anchors were cut out and the surfaces scabbled. However, the seam between the floor and the curb around the perimeter of the hot shop was never removed and it contained contamination.

In 1991, Bay C resumed fuel rod fabrication utilizing UO₂ pellets. The fuel rods left Bay C to Bay B, only after being surveyed for contamination. Therefore, Bays A and B remained relatively clean from radiological contamination.

In 1996, BNFL had partially decontaminated Bay C before the project was terminated. Contamination still remained on the original walls behind the shielding walls, and under the paint on the floors, walls and ceiling. New exterior walls (concrete block) were constructed around Bay C to provide shielding for the commercial support operations.

The roof trusses inside of Bay C remain contaminated. The trusses are constructed of welded channel steel, and the welds provide a space between the adjacent pieces of channel steel. There is contamination trapped in the trusses of Bay C and the contamination has been painted over.

Radiological Waste Line

The Building 17 radiological liquid waste line was connected to Building 6 via manhole number 4, south of Building 17, and east of Building 6A. The radiological waste line was removed from the manhole up to Building 17 and from under Bays A and B, in 1994, during the BNFL decontamination effort. No soil samples, collected during this effort, contained radiological materials above background levels. Both the radiological waste line and the industrial waste line remain under Bay C.

All of the industrial [chemical] waste lines associated with Building 17 still remain in place and may contain radiological materials.

Building 17 Ventilation System and Roof

Engineering controls (e.g., hoods, vacuums) were used throughout the facility to control airborne contamination. Air handlers controlled the effluent flow from the fuel manufacturing operations through HEPA ventilation units before the air was exhausted to the outside. The ventilation units are shown in Figures 4-6 and 4-7. The air handlers (FA1 – FA4) and the roof of the building were contaminated from the building effluent.

FA3 is the only remaining air handler and it is still contaminated. This unit is located on the mezzanine at the east end of Bay C.

The roof was contaminated with UO₂ when the HEPA filters were changed out. The contamination was then encapsulated in the roof when the roof was re-asphalted. The roof was reportedly re-asphalted multiple times. A new roof membrane was installed during the work conducted from 1994 to 1996. This membrane was placed over the existing roof after the gravel was vacuumed. Former employees stated that the gravel was shipped to Barnwell.

The roof drains run through the inside of Building 17 before flowing to the storm drain north of the building. The storm drain flows to a spillway located in a swale down-gradient from the northwest corner of Building 17. The spillway was plugged during the building renovation project and a new spillway with new piping to the storm drain was installed. The original equipment was left in the north parking lot. A drainage swale is also located downgradient of the southwest corner of Building 17.

The under support structure for the roof consists of steel joists which are made up of angle iron sections which have been welded together. The angle iron in some parts of the joists are welded back to back forming a crevice in which loose radioactive contamination has been trapped. The contamination has been painted over.

Sanitary Waste Line

The sanitary sewer system is potentially contaminated because of historical commercial fuel operations. The original sanitary sewer lines in Building 17 are known to be contaminated. New sanitary lines were installed during the building renovations. Radiological surveys of Building 17 repeatedly show low levels of uranium contamination in the restrooms. Due to

known contamination throughout the building during fuel production, the sewer lines were considered contaminated. Uranium found in the manholes near Building 17 also indicate that operations in the building contaminated the sewer lines.

4.1.3.3 Previous Radiological Investigations

Radiological surveys were provided in the Buildings 17 and 21 Decommissioning Plan. Surveys were performed in 1995 prior to the refitting of Building 17 for the commercial nuclear support services program. A summary of the survey results is presented in Table 4-3.

4.1.4 Building 6A Complex

The Building 6A Complex is located near the center of the Site, immediately south of Building 17 (Figure 3-2). Prior to 1961, Building 6A was a liquid radiological waste processing facility. Liquid radiological wastes handled in Building 5 were processed in Building 6A prior to being released to Site brook. After 1961, Building 6A was used as a maintenance facility.

Building 6A is a one-story concrete structure with a steel roof deck and a basement level. The total footprint of Building 6A is approximately 9,560 ft² (Figure 4-10). The original Building 6A (poured concrete) was approximately 4,720 ft². A concrete block addition was added to the West End in 1967. The addition is approximately 4,840 ft². The majority of the building was built into the hillside north of the Building 5 Complex.

The building was originally constructed for the treatment of radioactive liquid waste from Building 5. Similar to Building 6, Building 6A contained ten 2,000-gallon steel storage tanks. The first level contained four 5,000-gallon steel dilution tanks. The tank system was 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, were referred to as the A and B waste storage systems. There is a shallow sump located in the southeast corner of Building 6A. In 1961, the tanks were cleaned and relocated to the tank farm, south of Building 3, to store fuel oil. The oil tanks were taken out of service and disposed in 2001.

Currently, Building 6A consists of a maintenance service building equipped with offices, tool crib, a shop, and garage. The basement is used for supply storage.

Building 6A is equipped with a ventilation system and sanitary waste lines. The ventilation unit is mounted on the roof. The radiological waste lines have been cut and capped.

Building History

Building 6A was constructed during 1956. The building was constructed as a liquid radiological waste collection and processing facility for Building 5. In 1960, the liquid radiological waste from the Building 5 Complex was re-routed to Building 6. The liquid radiological waste was sampled, diluted in dilution tanks, if necessary, and discharged to the Site brook.

Radiological History and Potential Contaminants

Building 6A functioned as a radioactive waste collection, monitoring, and dilution facility for fuel fabrication and laboratory operations in Building 5 from approximately 1957 until 1960. Therefore, the building is potentially contaminated with uranium used in the Building 5 Complex.

Liquid waste was generated from the handling of uranium in conjunction with the fuel handling programs. 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.

The waste was then processed in Building 6A through ten 2,000-gallon tanks. The tanks were connected in series in the hot waste dilution vaults. Wastewater from the radiologically contaminated processed generally went directly to 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 6A prior to being discharged into Site brook.

Liquid discharges from Building 6A to Site brook terminated in 1960 and the liquid radiological waste was re-routed to Building 6. In 1961, an addition was constructed on the west end of the building to function as facilities services and the tanks and control equipment for the treatment of liquid radiological waste were removed. Four of the tanks were relocated to the present Tank Farm for the storage of No. 4 fuel oil. Sludge samples collected in July 2001, from the bottom of two of the relocated tanks, did not contain radioactive materials above background. The tanks were cleaned of oil residues and disposed of. The remaining tanks were most likely disposed of off-Site, however no information is available.

Building 6A was built within the hillside just north of the Building 5 Complex. As such, there is periodic seepage through the concrete walls and floor as evidenced by wet walls and standing water. The water level in the basement sump will rise following a heavy rain. Therefore, the water and soil under and around the building are also potentially contaminated. The contaminated sump is located in the southeast corner of the building but is not connected to the sanitary system, however, prior to 1995, groundwater seepage that collected in this sump was discharged to the sanitary sewer system, and may have impacted the associated lines and manholes. Building 6A is currently in use as a Site maintenance facility. It will be remediated as necessary and surveyed for release. Building 6A will be dismantled within two years of the NRC approval of the Decommissioning Plan and DCGLs.

Sanitary System

The waste retention system in Building 6A originally discharged to manhole S-16 en route to the Site brook. Manhole S-16 was part of the industrial waste system. At some time, manhole S-16 was incorporated into the Sanitary Sewer System. Some of the piping from the exterior of the building to the manhole (S-16) is still in place. The sump in the basement of Building 6A was also discharged to the Sanitary Sewer System at manhole S-17. This caused manhole S-17 and the main sanitary discharge line from Building 6A (southeast corner) to be contaminated with uranium. The contamination started from the point under the locker room and continued downstream to the point where it exits the east wall of the building.

All sources of water to the building were terminated and the sanitary system was rerouted to prevent further release of contamination. The sanitary waste line was cut in several locations inside the building to prevent any discharges from leaving the building. A new discharge

line from the building was installed that connects with the sanitary sewer system downstream from manhole S-16. There are currently no discharges to manhole S-16.

Previous Radiological Investigations

In 1994, CE surveyed the Building 6A basement due to the discovery of a radiologically contaminated industrial waste line leading from Building 17 to a manhole adjacent to Building 6A. The survey identified contamination in the sump, which required posting the area.

In 1995, the radiological status of the Building 6A Complex and its sanitary system were assessed. Contamination in excess of 500 pCi/g total uranium was measured in the basement and sump areas. Surveys of the basement indicated maximum fixed readings of 100 dpm/100cm² alpha and less than 2000 dpm/100cm² beta/gamma. No significant removable contamination was found.

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, will be 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 FUSRAP. 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).

The ELI was conducted in July 1998 and included:

- a review of construction plans;
- interviews, conducted with Site personnel;
- a visual inspection of the evaporator line manholes;
- six water and accumulated sediment samples, collected from the manholes for chemical and radiological analyses;

- leak tests, performed on manholes and sections of piping;
- seven soil borings, (SB-1201 through SB-1207) drilled at areas where significant leaking was observed;
- four monitoring wells, (MW-1201 through MW-1204) installed at four of the soil boring locations (SB-1201 through SB-1204); and
- four groundwater samples, collected from the monitoring wells screened at approximately 12 to 30 feet bgs to assess groundwater quality beneath the line.

Results from the manhole samples indicated that there were both chemical and radiological materials within the sediments that accumulated in the manholes. Furthermore, the leak testing indicated that the evaporator line was compromised, and that the majority of the leaks were occurring at the manholes. All sediment and water samples collected from within the manholes were submitted for laboratory analysis by alpha and gamma spectroscopy. U235 activity levels in sediments ranged from 2.63 pCi/g at MH-9 to 684 pCi/g at MH-7. U238 activity levels in sediments ranged from 13.75 pCi/g at MH-9 to 3,562 pCi/g at MH-7.

Chemically, only the sediments located in MH-5 contained solvents at concentrations that are considered hazardous under RCRA guidance. Therefore during remediation, only the sediments in manhole MW-5 are likely to be a mixed waste.

Soil borings (via hollow stem auger) were installed to depths ranging from 17 to 32 feet bgs, and samples were collected at continuous intervals. Groundwater was encountered between 10 and 25 feet bgs. Saturated and unsaturated subsurface soil samples collected adjacent to the industrial waste lines contained chemical compounds at concentrations above the CTDEP Remediation Standard Regulation (RSR) Criteria; however, none of the soil samples contained residual radiological materials above background.

Monitoring wells were then installed in four of the seven soil borings to intercept the water table at approximately 12 to 30 feet bgs. All four wells contained chemical compounds at concentrations that exceeded CTDEP RSR Criteria, however no water samples contained residual radiological materials above background. More recent groundwater sampling has shown the presence of uranium at levels of 11.48 pCi/liter in groundwater in MW-1201.

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.

Based on information obtained during the ELI, it is assumed that all of the 'hot' lines south and west of Building 6 contain radioactive materials, including LEU, HEU, and byproduct materials. The portion of the industrial waste lines that run from Building 3 to the stream contain HEU above 20 percent and have been identified as Impacted Areas under FUSRAP.

USACE has recently completed collecting sediment samples from within the industrial waste lines associated with Building 3.

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 receives 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.

Sanitary Lines and WWTP

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.

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 on-Site manhole (M-0) 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 that 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.

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 receives sanitary waste from the John B. Faucette Wellness Center or Health Works. This facility was also the former PDU, and was a test coal gasification plant. No radiological materials have been used at this location; therefore, radiological contamination is not expected to be present.

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.

Several drains that service parking areas near Buildings 17 and 21 discharge to woods to the northwest of these buildings.

Several drains that service parking areas south of Building 2 discharge to a rip-rap swale adjacent to East Main Street.

Several drains within the Building 2 complex also flow northward and discharge to a wooded area to the north of Health Works.

The storm water lines at the Site currently only receive runoff during rain events. Historically, these lines received boiler blowdown, condensate, and miscellaneous water discharges from several floor drains from Site buildings (Buildings 1A, 2, 2A, 5, and 17). The floor drains leading from these buildings were plugged in 1995 to prevent non-permitted discharges from entering the storm lines. 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 5, 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.

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 Areas. 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.

Soils in areas of the Site designated as FUSRAP (see Figure 4-11) will be remediated by the USACE.

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, and are under the purview of the FUSRAP remediation, (Figure 4-11).

Great Pond

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 Building 5 complex, Building 6A, the Building 17 Complex, and a portion of the industrial waste lines that runs 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 are 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 be present, there is a potential for radiological material to have entered the pond via airborne emissions and/or surface runoff.

Small Pond

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 this time, as Small Pond is currently undergoing characterization for Ecological Risk Assessment.

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 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. 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). 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.

5.0 DOSE MODELING

This section describes the development of Derived Concentration Guidance Levels (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 U.S. Nuclear Regulatory Commission (NRC).

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.

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 as low as reasonably achievable (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 CE 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 CE and the USACE 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.

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 unlikely given: 1) the current land-use trend, 2) the current community growth, planning, and development strategies of the local

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

municipality, and 3) the economic value of the land for sustained commercial use. Although unlikely, it is reasonable 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 improbable. In addition to the factors that make a residential land use unlikely, the likelihood that subsistence farming might occur at this Site is thought to be even more 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.

**6.0 ALTERNATIVES CONSIDERED AND RATIONAL FOR CHOSEN
ALTERNATIVE**

It is the objective of CE to decommission the CE Windsor Site, including associated buried piping and adjacent grounds, such that the areas 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 As Low As Reasonably Achievable (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 approximately 600-acres of land located at the Combustion Engineering (CE) Site, Windsor, Connecticut.

It is the intention of the licensee (CE) 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).

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 D of NUREG-1727 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."

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, a simplified analysis (possible benefits and costs relating to decommissioning, a determination of residual radioactivity levels that are ALARA, and cost vs. soil activity levels,) is presented below.

NUREG-1727, Appendix D 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, the level of residual radioactivity are already ALARA without taking additional remediation action.

Table D1, NUREG-1727, Possible Benefits and Costs Related To Decommissioning,

| Possible Benefits | Possible Costs |
|--|---|
| <ul style="list-style-type: none"> • Collective Dose Averted • Regulatory Costs Avoided • Changes in Land Values • Esthetics • Reduction in Public Opposition | <ul style="list-style-type: none"> • Remediation Costs • Occupational Non-radiological Risk • Additional Occupational/Public Dose • Transportation Direct Costs and Implied Risks • Environmental Impacts • Loss of Economic Use of Site/Facility |

During this analysis, results from an appropriate dose modeling method were used to relate concentrations to dose. Information used for this analysis, regarding concentration to dose values, was obtained from the Derivation of the Site-Specific DCGLs (ABB, 2002).

7.2 DETERMINATION OF BENEFITS

7.2.1 Collective Dose Averted Benefit

Remediation of site soils to levels that meet the unrestricted use criteria in 10 CFR 20.1402, using appropriate dose modeling to relate concentrations to dose, are 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 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 DCGL) which allow for any and all land use scenarios, including the most restrictive - Suburban Residential Scenario (Child). 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

CE, in cooperation with the U.S. Army Corps of Engineers–New England District (CENAE) commissioned the formation of a steering group representing CE, ACE, the State of Connecticut, and the Town of Windsor. The CE Site Project Steering Group was charged with representing stakeholders interests in the process to derive a dose-based soil concentration guideline (DCGL) value specific to the CE Windsor Site. CE, the ACE, 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 includes all possible costs excluding Environmental Impacts and Loss of Economic Use of Site/Facility costs. These two areas of cost are excluded in the analysis 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 D3 of NUREG-1727 which states:

$$\text{Cost}_T = \text{Cost}_R + \text{Cost}_{WD} + \text{Cost}_{Acc} + \text{Cost}_{TF} + \text{Cost}_{WDose} + \text{Cost}_{PDose} + \text{Cost}_{Other}$$

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 Remedial Action – Removal of Contaminated Soil To A DCGL Value of 25 mrem/y.

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 Remedial Action is: \$12,146,235.

| Cost Function | Monetary Cost | Formula |
|----------------------|---------------|--|
| $Cost_R + Cost_{WD}$ | \$11,712,120 | based on the cost of $\$840/m^3 \times 13943 m^3$ |
| $Cost_{Acc}$ | \$2,846 | based on the movement of a total of $13943 m^3$ of materials, at a rate of 1.62 person-hours per m^3 , an accident rate of 4.2×10^{-8} fatalities per person-hour, and a cost of \$3,000,000 per fatality |
| $Cost_{TF}$ | \$431,269 | based on $13943 m^3$ being shipped / $13.6 m^3/shipment \times 3690 km \times 3.8 \times 10^{-8}$ 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 $\times 0.008$ 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 Function | Monetary Cost | Formula |
|-----------------------|---------------|--|
| Cost _{Other} | NA | Land restoration costs are not included in this analysis |
| Cost _T | \$12,146,235 | (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 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 concentrations that are ALARA are determined in accordance with Equation D8 of NUREG-1727, Appendix D, where:

$$\frac{\text{Conc}}{\text{DCGL}_w} = \frac{\text{Cost}_T}{\$2000 \times P_d \times 0.025 \times F \times A} \times \frac{r + \lambda}{1 - e^{-(r+\lambda)N}}$$

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 Remedial Action

Inputs and Assumptions:

| Cost Function | Monetary Cost | Formula |
|--|---------------|--|
| Cost _R + Cost _{WD} | \$11,712,120 | based on the cost of \$840/m ³ x 13943 m ³ |
| Cost _{Acc} | \$2,846 | based on the movement of a total of 13943 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} | \$431,269 | based on 13943 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 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 (0.0145 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 | \$12,146,235 | (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 | 0.03 |
| N | 1000 years |
| A | 2,023,435 m ² |
| λ | 0 |
| F | 0.5 (assuming 50% of the source term is removed during remediation activities) |

$$\frac{\text{Conc}}{\text{DCGL}_w} = \frac{\$12,146,235}{\$2000 \times (4 \times 10^{-4}) \times 0.025 \times 0.5 \times 2,023,435} \times \frac{0.03 + 0}{1 - e^{-(0.03+0)1000}}$$

$$\frac{\text{Conc}}{\text{DCGL}_w} = 18.0$$

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

7.5 CONCLUSION

This pre-remediation ALARA analysis demonstrates that the dose criteria in Subpart E of 10 CFR Part 20 will be met using the Remedial Action DCGL, concludes whether it is feasible to further reduce the levels of residual radioactivity to levels below those necessary to meet the dose criteria, and demonstrates that the Remedial Action DCGL is 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 Action.

As presented in Section 7.4.1.1, ALARA Analysis for Remedial Action, an ALARA analysis for the Remedial Action was performed. The ratio of conc to $DCGL_W$ was calculated to be 18.0. Since this value is greater than one, it is determined that the $DCGL_W$ for the Remedial Action for the unrestricted release of the area is ALARA, and no additional remediation for the area is justified.

8.0 PLANNED DECOMMISSIONING ACTIVITIES

The scope of decommissioning activities includes the decontamination and deconstruction of structures in Building Complexes 2, 5, 6A and 17, including concrete foundations, the removal of buried utilities (non-FUSRAP impacted), and the transportation of radioactive and mixed waste. Conduct of Final Status Surveys for non-FUSRAP impacted areas of the Site will be conducted as appropriate.

Decontamination and deconstruction of above-grade structures in Building Complexes 2, 5, and 17 was authorized by License Amendment No. 53. This includes removal of the structure down to, but not including the slabs and foundations. Where structures remain, such as Building 6A, the sequence of decontamination and deconstruction, of above grade structures will generally follow the outline shown below, (which is currently being applied to Buildings 2, 5, 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

8.1 CONTAMINATED STRUCTURES

8.1.1 Building Demolition

The above-grade structures in Building Complexes 2, 5, and 17 have been approved for decontamination and deconstruction down to, but not including, the slabs and foundations.

CE plans to decontaminate the concrete slabs and foundations of the structures in Building Complexes 2, 5, and 17. Building 6A will be remediated as necessary and surveyed for release. Building 6A will be dismantled within two years of the NRC approval of the

Decommissioning Plan and DCGLs. 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

The structural components of each building will be cut by the shear/grapppler 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 the skid-steer loaders.

8.1.3.2 Sequence for Building 17 Complex

Bays A and B will be structurally dismantled and packaged. The structural components of the buildings will be cut by the shear/grapppler 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 the skid-steer loaders.

8.1.3.3 Sequence for Building 6A

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

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 Authorized Under Existing License

Current NRC License No. 06-00217-06 authorizes the following activities and procedures:

- Decontamination of above grade and basement structures in Building Complexes 2, 5, and 17.
- Removal of interior systems, components, walls, floors, piping, wiring, conduit, etc.
- Use of invasive means such as scabblers, and CO₂ 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.
- Decontamination and deconstruction of above-grade structures in Building Complexes 2, 5, and 17.

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.

8.2 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.

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 Authorized Under Existing License

Current NRC License No. 06-00217-06 authorizes the activities and procedures described in this section.

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, all buried piping and equipment in the Building Complexes 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 utilizing a Cat 235 or equivalent. Trenches will be dug 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 Authorized Under the Existing License

Current NRC License No. 06-00217-06 does not authorize remediation of contaminated soils or subsurface utilities.

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.

8.4 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 and separate approvals of other submittals to the NRC (e.g., DCGLs). If it is determined that decommissioning cannot be completed as outlined in the schedule, an updated schedule will be submitted to the NRC.

9.0 PROJECT MANAGEMENT AND ORGANIZATION

CE has contracted with a decommissioning contractor to perform the decontamination and deconstruction (D&D) of Building Complexes 2, 5, and 17 at their CE Windsor Site. The decommissioning contractor has the overall responsibility for safe completion of all D&D activities conducted at Building Complexes 2, 5, and 17. The Building 6A Complex will be handled similarly. The Site areas designated for remediation under FUSRAP will be under the authority and control of the USACE.

9.1 DECOMMISSIONING MANAGEMENT ORGANIZATION

The functional organization for the completion of D&D activities includes the Project Manager, Site Manager, Site Health, Safety & Radiological Safety Officer, Quality Assurance Manager, Radiation Safety Officer and support personnel, as required. Figure 9-1 shows the functional organization and the relationship to Combustion Engineering and the corporate interface.

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 Commercial Area Decontamination and Decommissioning Project. This individual provides the senior project management oversight 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 Site Health, Safety & Radiological Safety Officer, and the Quality Assurance Manager.

9.2.1.1 Experience and Qualifications

The Project Manager will hold a degree in science or engineering, 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 Safe 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 and have at least five years supervisory or management experience, with at least five years decontamination and decommissioning experience, and ten years of construction experience.

9.2.3 Site Health, Safety & Radiological Safety Officer

The Site Health, Safety & Radiological Safety Officer (SHSRSO) is principally responsible for radiological health and safety, and regulatory compliance. This individual's duties include implementation of a site-specific Health and Safety Plan (HASP). The SHSRSO

shall perform daily safety inspections to ensure compliance with all regulatory requirements, review and validate analytical and air monitoring data, and perform monthly and quarterly self-assessments. In addition, the SHSRSO 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 SHSRSO 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 SHSRSO shall hold a degree in science or engineering and have at least five years experience in areas such as radiation safety, radiation monitoring, emergency preparedness, industrial safety, and personnel exposure evaluation. The SHSRSO 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 SHSRSO shall be capable of directing the surveillance activities of the Health Physics Technicians.

The duties and responsibilities of the SHSRSO 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, CE 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 responsible 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 work experience) 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.

9.3 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 managed 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 is normally 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. Subsequently,

the supervisor or Health and Safety will initiate the RWP and provide a description on the RWP of existing and/or anticipated radiological conditions.

After initiation, the supervisor or Health and Safety 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 his designee. Following development, the RWP is issued to the authorized users. 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 designee. 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 1000 dpm/100cm² 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 1000 dpm/100 cm² 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 1000 dpm/100cm².
 - 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 for CE Commercial Area. 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.

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 the RSO or designee, but shall not be a member of the contractor organization. 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 and Safety 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 as low as reasonably achievable (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 Prospects, Inc. establishes management commitment to the environmental ALARA program, establishing an ALARA policy and implementation of ALARA goals.

Policy

It is policy of ABB Prospects, Inc. 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 2) 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 these 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. Description 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 D&D of the Commercial Areas will include, but are not limited to Resource Conservation and Recovery Act (RCRA) or Toxic Substances Control Act (TSCA) wastes, low-level radioactive waste (LLW), 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. Waste materials and volumes derived from the Decommissioning Funding Plan submitted to the NRC on March 30, 2001, include 12,930 cubic feet of building rubble and 1,135 cubic feet of D&D waste. Additionally, an estimated 148,810 cubic feet of soil will be removed. An additional 201 cubic feet of radiologically impacted building rubble and 70 cubic feet of radiological impacted soil are designated for removal from Building 6A.

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 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 LLW

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 (LLW/RCRA or LLW/TSCA) will be managed in an area that meets the requirements of a LLW Staging Area/RCRA Accumulation Area (90-day or Satellite) or LLW 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 6A. No other solid mixed wastes are expected. No liquid mixed wastes are expected to be generated. Approximately 1,200 cubic feet of solid mixed waste is expected to be generated. 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 we intend to send mixed wastes for disposal at Envirocare 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 Waste

LLW 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). LLW will be segregated from uncontaminated wastes to minimize the amount of LLW generated. The staging area will have boundary marking and signs will be posted identifying the area.

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.

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

At present we only expect the wastewater to be contaminated at concentrations less than 100 pCi/L total Uranium, and less than 5pCi/L Co-60. Soil contaminant levels are expected to range from less than 100 pCi/g to 10,000 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.

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, Site Health, Safety & Radiological Safety Officer, 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 Commercial Area 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. The unrestricted release criteria for soils and sediments that remain on-site will be in accordance with the DCGLs described 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, and the pre-dismantlement decontamination of the buildings is complete for Building Complexes 2,5, and 17. The plan provided 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.

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 including FUSRAP, 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, including FUSRAP areas. 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.

Final surveys and release of remediated FUSRAP areas are under the cognizance and authority of the USACE. The remaining areas of the Site will be surveyed by CE. It is intended that the combined results will be submitted to the NRC to support license termination. The FUSRAP areas are indicated in brown on Figure 4-11. The "remaining areas" are indicated in yellow and green on Figure 4-11.

Survey Design

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. However, locations where HEU (greater than 20% enrichment) occurs are under the cognizance and authority of the USACE, and are not addressed in this Plan.

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 for structural surfaces will be determined at the time of the survey as part of instrument calibration.

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 Wilcoxon Rank Sum (WRS) statistical test will be used to evaluate the data from the final status surveys.

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. There have been no changes in the cost estimate for decommissioning since the submittal of the Decommissioning Funding Plan. By letter dated July 10, 2001, the NRC stated that the Decommissioning Funding Plan had been reviewed and that there were no further questions at that time. The Decommissioning Funding Plan (ABB, 2002b) was revised and submitted by letter dated December 6, 2002.

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. The level of funding for the Decommissioning Funding Plan remains below the existing bond, therefore the Surety Bond and Standby Trust Agreement remain unchanged.