DECOMMISSIONING PLAN

CE WINDSOR SITE
2000 DAY HILL ROAD
WINDSOR, CONNECTICUT 06095

Prepared for:
Combustion Engineering, Inc.
2000 Day Hill Road
Windsor, Connecticut 06095

Prepared by:
MACTEC Constructors
Golden, Colorado 80401

April 2003

MACTEC, Inc.

[redacted]

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US NRC LICENSE NUMBER 06-00217-06
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TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

List of Tables

List of Figures

List Of Effective Pages

i	04/04/03
ii	04/04/03
iii	04/04/03
iv	04/04/03
v	04/04/03
vi	04/04/03
vii	04/04/03
viii	04/04/03
ix	04/04/03
x	04/04/03
xi	04/04/03
xii	04/04/03
1-1	04/04/03
1-2	04/04/03
1-3	04/04/03
1-4	04/04/03
2-1	04/04/03
2-2	04/04/03
3-1	04/04/03
3-2	04/04/03
3-3	04/04/03
3-4	04/04/03
3-5	04/04/03
3-6	04/04/03
3-7	04/04/03
4-1	04/04/03
4-2	04/04/03
4-3	04/04/03
4-4	04/04/03
4-5	04/04/03
4-6	04/04/03
4-7	04/04/03
4-8	04/04/03
4-9	04/04/03
4-10	04/04/03
4-11	04/04/03
4-12	04/04/03
4-13	04/04/03
4-14	04/04/03
4-15	04/04/03
4-16	04/04/03
4-17	04/04/03
4-18	04/04/03
4-19	04/04/03
4-20	04/04/03

List Of Effective Pages

4-21	04/04/03
4-22	04/04/03
4-23	04/04/03
5-1	04/04/03
5-2	04/04/03
6-1	04/04/03
7-1	04/04/03
7-2	04/04/03
7-3	04/04/03
7-4	04/04/03
7-5	04/04/03
7-6	04/04/03
7-7	04/04/03
8-1	04/04/03
8-2	04/04/03
8-3	04/04/03
8-4	04/04/03
8-5	04/04/03
8-6	04/04/03
9-1	04/04/03
9-2	04/04/03
9-3	04/04/03
9-4	04/04/03
10-1	04/04/03
10-2	04/04/03
10-3	04/04/03
10-4	04/04/03
10-5	04/04/03
10-6	04/04/03
10-7	04/04/03
10-8	04/04/03
11-1	04/04/03
11-2	04/04/03
11-3	04/04/03
11-4	04/04/03
11-5	04/04/03
12-1	04/04/03
12-2	04/04/03
12-3	04/04/03
13-1	04/04/03
13-2	04/04/03

List Of Effective Pages

13-3	04/04/03
13-4	04/04/03
14-1	04/04/03
14-2	04/04/03
14-3	04/04/03
14-4	04/04/03
14-5	04/04/03
14-6	04/04/03
14-7	04/04/03
15-1	04/04/03

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
HWOCR	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	Wilcoxen Rank Sum
WWTP	Wastewater Treatment Plant

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1.0 EXECUTIVE SUMMARY

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

[ ] Ex. 4

- 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 Stated 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 (ABB, 2002a), submitted separately for review and approval by letter dated February 22, 2002. An addendum to the DCGL Report will be submitted to the NRC in April 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-REMEDIATION ACTIVITIES

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

1.9 AMENDMENT TO LICENSE TO INCORPORATE DP

This Decommissioning Plan 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 ft^2. The building was a one-story metal structure constructed on a concrete slab.

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

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

2.3.2 Building Complexes 2, 5, 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).[

[

] Ex. 4

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 ft^2. 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 ft^2. 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 ft^2. 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 ft^2. 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 ft^2. 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: UO_2 powder densification, solid solution ceramic fuels, irradiation damage, ternary phase determinations, ceramic fuel thermal characteristics, welding techniques for cluster fabrication, and UO_2 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 UO_2 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 ft^2. 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 UO_2 in the superheater fuel element was enriched to 3.5 percent U235. The uranium operations consisted of filling the rods with UO_2 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 ft^2.

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 ft^2. 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 ft^2. 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 ft^2). 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 ft^2 (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 ft^2) including receiving and storage in addition to a fuel loading shop (Bay C). The remaining 20,000 ft^2 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, UO_2 pellets were received at the CE Windsor Site from an off-Site supplier, for fabrication into fuel assemblies. Later, to support production requirements, UO_2 powder was received on Site. The UO_2 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 UO_2 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 UO_2 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 ft^2 (Figure 4-10). The original Building 6A (poured concrete) was approximately 4,720 ft^2. A concrete block addition was added to the West End in 1967. The addition is approximately 4,840 ft^2. 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/100cm^2 alpha and less than 2000 dpm/100cm^2 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, 2002), submitted separately to the NRC for review and approval on February 22, 2002. An addendum to the DCGL Report will be submitted to the NRC in April 2003. The development of the DCGLs was performed using a steering committee of stakeholders, including USACE, CTDEP, and NRC 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 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, 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.

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

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 was also considered as a future use scenario but eliminated as prohibitively unlikely. In addition to the factors that make a residential use unlikely, subsistence farming is 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.

DCGLs are calculated for five different potential exposure scenarios considered reasonably possible:

- An occupational worker employed at a facility located at the Site,
- A commercial truck farmer,
- A construction worker participating in a construction or excavation project at the Site,
- A recreational visitor using open park-like space (jogging, biking, etc.) at the Site, and
- A residential occupant in a suburban residential setting

The DCGLs for the CE Site were calculated using the RESRAD 6.0 modeling code. Each of the scenarios modeled is credible and foreseeable, and each results in a concentration corresponding to the 25 mrem/y dose limit. Considering the potential future land-use scenarios, the limiting scenario (the one that results in the smallest concentration yielding 25 mrem/y) is the Suburban Residential 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 were derived using appropriate techniques in accordance with governing guidance, standards, and regulations in concert with the input of a panel of experts and stakeholders assembled into a steering group and representing the interests of the stakeholders, CE, the USACE, and State and Federal regulators.

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

During 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:

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

Where:

$Cost_T$ = Total cost of the remediation action;

$Cost_R$ = Monetary cost of the remediation action;

$Cost_{WD}$ = Monetary cost for transport and disposal of the waste generated by the action;

$Cost_{Acc}$ = Monetary cost of worker accidents during the remediation action;

$Cost_{TF}$ = Monetary cost of traffic fatalities during transporting of the waste;

$Cost_{WDose}$ = Monetary cost of dose received by workers performing the remediation action and transporting waste to the disposal facility;

$Cost_{PDose}$ = Monetary cost of the dose to the public from excavation, transport, and disposal of the waste;

$Cost_{Other}$ = Other costs as appropriate for the particular situation.

7.3.1.1 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 $840/m^3$ 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
$Cost_{WDose}$	NA	based on dose modeling performed, dose to an average construction worker is estimated to be 8 mrem/y at a soil concentration of 1500 pCi/g Total U or 14.5 mrem/y at a soil concentration of 12.6 pCi/g Co-60. At $2000 per rem person x 0.008 rem/y (0.014.5 rem/y) = $16 ($29) per year per construction worker. This dollar value is insignificant and will not add significant cost to the total cost of remediation and need not be evaluated for the different alternatives

Cost Function	Monetary Cost	Formula
$Cost_{PDose}$	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	$(Cost_R + Cost_{WD}) + Cost_{Acc} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} + Cost_{Other}$

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:

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

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 Function	Monetary Cost	Formula
$Cost_{Acc}$	$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
$Cost_{WDose}$	NA	based on dose modeling performed during the generation of the DCGL Report, dose to an average construction worker is estimated to be 8 mrem/y at a soil concentration of 1500 pCi/g Total U or 14.5 mrem/y at a soil concentration of 12.6 pCi/g Co-60. At $2000 per rem person x 0.008 rem/y (0.014.5 rem/y) = $16 ($29) per year per construction worker. This dollar value is insignificant and will not add significant cost to the total cost of remediation and need not be evaluated for the different alternatives
$Cost_{PDose}$	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)

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

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

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

The steel components will be sheared to size, and whenever possible, loaded directly into containers. Smaller components will be shuttled to containers using 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/grappler starting at one side and working to the other side of a bay. Structural steel will then be pulled and set aside for sizing and removal. Other building components will be removed in a similar manner.

The steel components will be sheared to size, and whenever possible, loaded directly into containers. Smaller components will be shuttled to containers using 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 CO_2 or hydro blasting of surfaces to remove surface contaminants.
- Cutting of pipes and equipment using powered saws and torches to reduce the volume of radioactive waste materials and to allow for easier storage of waste materials in containers.
- Abatement of asbestos using stripping methods approved by the State of Connecticut Department of Public Health. A Connecticut licensed asbestos abatement contractor will be used to perform this work. Abatement contract personnel are provided radiological safety training to allow access to areas where the asbestos is radiologically contaminated.
- Abatement of lead and/or PCB based paints using stripping methods approved by the State of Connecticut Department of Health. A Connecticut licensed lead abatement contractor will be used to perform this work. Abatement contract personnel are provided radiological safety training to allow access to areas where the lead is radiologically contaminated.
- Use of pneumatic and electric driven chippers and jackhammers to remove volumetrically contaminated concrete surfaces.
- Storage, packaging and shipping of waste materials.
- 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 responsibility for auditing the implementation and execution of Quality Assurance and Control procedures and to evaluate conformance with procedures and other requirements. The Quality Assurance Manager will also be responsible for supervising the project's Document Control procedures. This individual is responsible for preparation, implementation, and oversight of the Self-Assessment and Audits procedures, including identification of deficiencies and improvements, corrective actions, and feedback.

9.2.4.1 Qualifications and Experience

The Quality Assurance Manager shall hold a degree in science of engineering (or equivalent 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/100cm^2 and the worker is actively working in the area for greater than one hour during that workday; or
2. Areas with total contamination greater than 1000 dpm/100 cm^2 and the work involves invasive activities such as drilling, scabbling, digging, or otherwise causing the release of contaminants or contaminated material into the air.

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

10.1.1.2 Action Level and Limit

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

10.1.2 Respiratory Protection Program

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

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

10.1.2.1 Medical Evaluation

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

10.1.2.2 Fit Test

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

10.1.2.3 Selection

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

10.1.2.4 Issue

Workers may be assigned respirators for their exclusive use or they shall otherwise be issued by Health Physics. Respirators shall only be assigned or issued to workers qualified, with

respect to the program, to use respiratory protection equipment. The type of respirator selected shall be documented on the Radiation Work Permit.

10.1.2.5 Inspection

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

10.1.2.6 Cleaning

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

10.1.2.7 Maintenance

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

10.1.2.8 Storage

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

10.1.2.9 Training

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

10.1.3 External Exposure Determination

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

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

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

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

10.1.4 Internal Exposure Determination

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

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

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

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

10.1.5 Summation of External and Internal Exposures

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

10.1.6 Contamination Control Program

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

10.1.6.1 Exposure Control

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

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

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

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

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

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

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

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

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

10.1.6.2 Radiation Surveys

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

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

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

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

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

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

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

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

i. Removable: The action level for removable alpha or beta-gamma radiation on a surface is 1000 dpm/100cm^2.

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

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

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

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

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

10.1.7 Instrumentation Program

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

10.1.7.1 Type and Use

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

10.1.7.2 Calibration

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

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

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

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

i. Unique identification (e.g. serial number),

ii. Initials or specific identifying mark of individual completing the calibration,

iii. Energy correction factors,

iv. Graph or table of calibration factors for each type of radiation for which the instrument may be used,

v. Instrument response to an identified check source,

vi. Unusual or special use conditions or limitations, and

vii. Date by which calibration is again required.

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

10.1.7.3 Verification

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

10.1.7.4 Sensitivity

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

10.2 NUCLEAR CRITICALITY SAFETY

Based on the License limit of 325 grams Uranium 235, there are no nuclear criticality safety concerns within the scope of the D&D 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 an significant unexplained increase in gross radioactivity, or (3) whenever a circumstance might cause a significant variation in radioactivity composition. The samples will be grab samples.

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

Air

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

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

11.3 EFFLUENT CONTROL PROGRAM

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

11.3.1 Procedures, Process Controls, and Engineering Controls

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

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

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

Dust Suppression

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

Erosion and Sediment Control

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

Ex. 4

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.

[ ] Ex. 4

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.

Ex. 4

FIGURE 4-11
SUMMARY OF IMPACTED AREAS
DECOMMISSIONING PLAN
CE WINDSOR SITE
WINDSOR, CONNECTICUT
MACTEC Constructors, Inc.

D-07

Figure 8-1
Conceptual Decommissioning Schedule

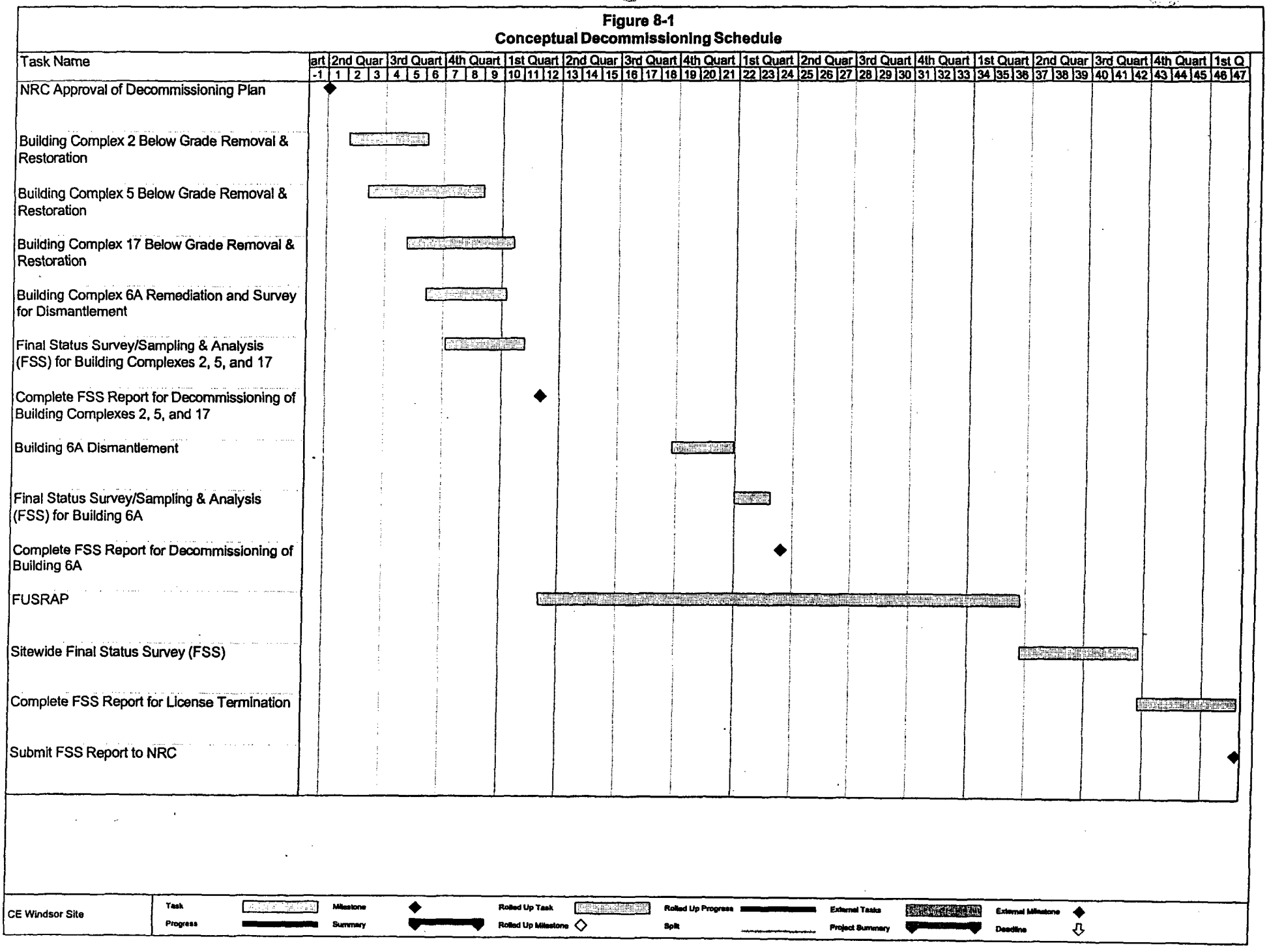
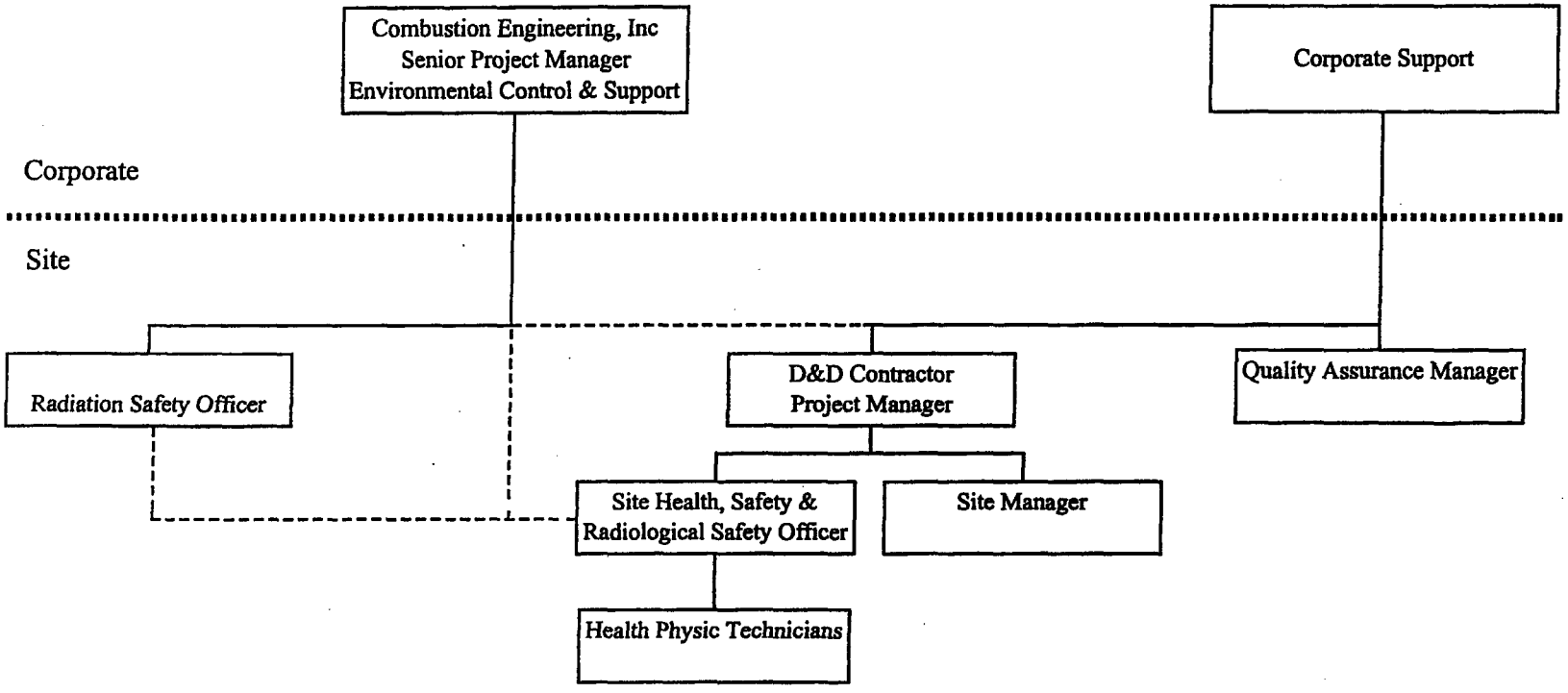


Figure 9-1
Decommissioning Organization Chart



Tables

List of Tables

Table 2-1
Authorized Radionuclides
NRC License No. 06-00217-06
CE Windsor Site, Windsor Connecticut

	Byproduct, source, and/or special nuclear material	Chemical and/or physical form	Maximum amount that licensee may possess at any one time under this license
A.	Any byproduct material with Atomic Numbers 1 through 83	Irradiated and/or contaminated debris, inspection and test equipment, test samples, calibration standards, or residues	50 curies
B.	Any byproduct material with Atomic Numbers 84 through 103	Irradiated and/or contaminated debris, inspection and test equipment, test samples, calibration standards, or residues	Not to exceed 3 millicuries per nuclide and 30 millicuries total
C.	Source material	Irradiated and/or contaminated debris, inspection and test equipment, test samples, calibration standards, or residues	75 kilograms
D.	Plutonium	Irradiated and/or contaminated debris, inspection and test equipment, test samples, calibration standards, or residues	1 milligram
E.	Uranium-235	Irradiated and/or contaminated debris, inspection and test equipment, test samples, calibration standards, or residues	325 grams including less than 5 kilograms UF_6

Table 2-2
Maximum Activities of Authorized Radionuclides
License Number 06-00217-06
CE Windsor Site, Windsor Connecticut

Byproduct, Source, and/or Special Nuclear Material	Chemical and/or Physical Form	Maximum amount that Licensee May Possess
A. Any byproduct material with Atomic Numbers between 1 and 83, inclusive	Irradiated and/or contaminated reactor components, inspection and test equipment, test samples, monitoring instruments, reactor coolant samples, or calibration sources	50 curies
B. Any byproduct material with Atomic Numbers 84 through 103	Irradiated and/or contaminated reactor components, inspection and test equipment, calibration sources or reactor coolant samples	Not to exceed 3 millicuries per nuclide and 30 millicuries total
C. Any source material	Irradiated and/or contaminated reactor components, inspection and test equipment, calibration sources or reactor coolant samples	75 kilograms
D. Any byproduct material	Irradiated and/or contaminated reactor components, inspection and test equipment, test samples, monitoring instruments, or reactor coolant samples	Not to exceed 51 curies total
E. Cesium 137	Sealed sources	215 curies
F. Cobalt 60	Sealed sources	250 millicuries
G. Americium 241	Any	1 millicurie
H. Americium 241	Sealed neutron sources	Not to exceed 1.0 curie per source and 10 curies total
I. Americium 241	Sealed neutron sources	Not to exceed 10 curies per source, 100 curies total
J. Neptunium 237	Sealed sources	10 sources not to exceed 0.5 millicuries per source
K. Plutonium 238	Sealed plutonium-beryllium neutron sources (MRC-PU8BE-XXX)	80 curies (3.5 grams) not to exceed 20 curies per source
L. Natural and/or depleted Uranium	Any	10,000 kilograms including less than 5 kilograms UF_6
M. Plutonium 238	Sealed sources	Not to exceed 1 gram per source and 4 grams total

Byproduct, Source, and/or Special Nuclear Material	Chemical and/or Physical Form	Maximum amount that Licensee May Possess
N. Neptunium 237	Oxide wires	Not to exceed 10 curies per source and 100 curies total
O. Uranium 233	Any	1 gram
P. Uranium 235	Any	7 grams
Q. Uranium 235	Fission Chambers	Not to exceed 1.7 grams per chamber and 13.6 grams total
R. Uranium 235	Any	325 grams, including less than 5 kilograms UF_6
S. Plutonium	Any	2 grams
T. Uranium 235	Contaminated debris and residues	325 grams, including less than 5 kilograms UF_6
U. Plutonium	Contaminated debris and residues	1 milligram

Notes:

A. through H. For use in research and development as defined in Section 30.4(q) 10 CFR Part 30, and for possession incident to maintenance repair, decontamination, and study of reactor components.

H. For use in testing and calibration of boron measuring devices and for distribution to persons holding operating reactor licenses and/or to persons authorized to receive the licensed materials pursuant to the terms and conditions of specific licenses issued by the Nuclear Regulatory Commission or an Agreement State.

I. through K. For possession, storage, and transfer to persons holding operating reactor licenses and/or to persons authorized to receive the licensed material pursuant to the terms and conditions of specific licenses issued by the Nuclear Regulatory Commission or an Agreement State.

L. Research and development as defined in 10 CFR 70.4 in the licensee's facilities located at Buildings 1, 2, 5, 6, 16, 17, and 18.

M. For storage only.

N. through Q. For possession, storage, and transfer to persons holding operating reactor licenses and/or to persons authorized to receive the licensed material pursuant to the terms and conditions of specific licenses issued by the Nuclear Regulatory Commission or an Agreement State.

R. Research and development as defined in 10 CFR 70.4 in the licensee's facilities located at buildings 1, 2, 5, 6, 16, 17, and 18.

S. For possession as surface contamination on tools or equipment incident to maintenance, repair, modification or storage.

T. through U. Possession and use for those activities directly or indirectly related to decontamination and decommissioning pursuant to a decommissioning plan approved by the NRC in writing.

Table 2-3
Maximum Activities of Authorized Radionuclides
License No. SNM-551
CE Windsor Site, Windsor Connecticut

Byproduct, Source, and/or Special Nuclear Material	Chemical and/or Physical Form	Maximum amount that Licensee May Possess
A. Source Materials		10,000 pounds
B. Uranium enriched in the U-235 isotope up to 3.5 weight percent		150,000 kilograms
C. Plutonium		16 grams plutonium encapsulated as a Pu-Be neutron source and 160 micrograms plutonium in the form of analytical samples
D. Uranium of any enrichment of the U-235 isotope		2 kilograms of the U-235 isotope at any enrichment
E. Uranium enriched to 5% in the isotope U-235		1000 kg U-235
F. Uranium enriched to 15% in the isotope U-235		27 kg U-235
G. Plutonium		10 microcuries

Notes: Authorized use and location were in accordance with the licensee's application at the licensee's Nuclear Manufacturing Facility located approximately five miles northeast of Windsor, Connecticut.

Table 2-4
Maximum Activities of Authorized Radionuclides
License No. SNM-1067
CE Windsor Site, Windsor Connecticut

Byproduct, Source, and/or Special Nuclear Material	Chemical and/or Physical Form	Maximum amount that Licensee May Possess
A. Uranium Enriched to Not More Than 5.0 w/o U-235	Oxide powder, pellets, rods, and finished fuel assemblies	500,000 kilograms total uranium
B. Uranium Enriched to Not More Than 3.5 w/o U-235		500,000 kilograms total
C. Uranium Enriched to Not More Than 4.1 w/o U-235	Oxide powder, pellets, rods and finished fuel assemblies	20,500 kilograms of contained U-235
D. Uranium enriched to or less than 5.0 w/o U-235	Residual Uranium oxides	700 grams U-235
E. Uranium enriched to greater than 5.0 w/o U-235	Residues	1000 grams U-235
F. Uranium Enriched to Not More Than any w/o U-235		4800 grams of the U-235 isotope at any enrichment
G. Uranium enriched to less than 20 w/o U-235	Any	4,800 grams contained U-235
H. Natural and/or depleted Uranium	Any	10,000 kilograms total uranium
I. Pu-238	Encapsulated Neutron Sources	5 sources, each containing less than 2.0 grams Pu-238
J. Pu	Any	160 micrograms as analytical samples
K. Uranium Enriched in the U-235 Isotope	Contained in Encapsulated U_3O_8	20 sources, each containing less than 1.7 grams U-235
L. Uranium Enriched to or Greater than 20 w/o U-235	Residues	1,000 g U-235
M. Pu-239		8 sources containing 16 grams of Pu-239
N. Plutonium	Analytical Samples	160 micrograms
O. Source Material		10,500 kg

Byproduct, Source, and/or Special Nuclear Material	Chemical and/or Physical Form	Maximum amount that Licensee May Possess
P. Enriched Uranium in the U-235 isotope	Any	<350 grams contained U-235*
Q. Pu	Sealed neutron sources	160 micrograms as analytical samples*
R. Uranium enriched in the U-235 isotope	Contained in encapsulated U_3O_8 sources in fission chambers	20 sources, each containing less than 1.7 grams U-235*

Notes: *The total special nuclear material will not exceed the limitation of:

$$\frac{\text{grams U-235}}{350 \text{ grams}} + \frac{\text{grams Pu}}{200 \text{ grams}} < 1$$

Authorized use and location were in accordance with the licensee's application at the licensee's existing facilities located approximately five miles northeast of Windsor, Connecticut.

Table 2-5
Maximum Activities of Authorized Radionuclides
License No. 06-30561-01
CE Windsor Site, Windsor Connecticut

Byproduct, Source, and/or Special Nuclear Material	Chemical and/or Physical Form	Maximum amount that Licensee May Possess
A. Any byproduct material with Atomic Numbers between 1 and 83, inclusive	Irradiated and/or contaminated reactor components, inspection and test equipment, test samples, monitoring instruments, reactor coolant samples, or calibration sources	50 curies
B. Any byproduct material with Atomic Numbers 84 through 103	Irradiated and/or contaminated reactor components, inspection and test equipment, calibration sources or reactor coolant samples	Not to exceed 3 millicuries per nuclide and 30 millicuries total
C. Source material	Irradiated and/or contaminated reactor components, inspection and test equipment, calibration sources or reactor coolant samples	75 kilograms
D. Cesium 137	Sealed sources	215 curies
E. Americium 241	Sealed neutron sources	Not to exceed 1.0 curie per source and 10 curies total
F. Americium 241	Sealed neutron sources	Not to exceed 10 curies per source, 100 curies total
G. Neptunium 237	Sealed sources	Not to exceed 0.5 millicuries per wire and 5 millicuries total
H. Uranium 233	Any	1 gram
I. Uranium 235	Any	7 grams
J. Uranium 235	Fission Chambers	Not to exceed 1.7 grams per chamber and 13.6 grams total
K. Plutonium	Any	1 milligram
L. Uranium 235	Any	325 grams, including less than 5 kilograms UF_6
M. Natural and/or depleted uranium	Any	10,000 kilograms including less than 5 kilograms UF_6
N. Plutonium 238	Sealed sources	Not to exceed 1 gram per source and 4 grams total

Notes: Authorized use:

A. through D. Research and development as defined in 10CFR30.4; possession incident to maintenance, repair, decontamination, study of reactor components, and processing of test samples.

E. through F. Testing and calibration of boron measuring devices and for distribution to persons holding operating reactor licenses and/or to persons authorized to receive the licensed materials pursuant to the terms and conditions of specific licenses issued by the Nuclear Regulatory Commission or an Agreement State.

G. through J. For possession, storage, and transfer to persons holding operating reactor licenses and/or to persons authorized to receive the licensed material pursuant to the terms and conditions of specific licenses issued by the Nuclear Regulatory Commission or an Agreement State.

K. For possession as surface contamination on tools or equipment incident to maintenance, repair, modification or storage.

L. Research and development as defined in 10 CFR 70.4 in the licensee's facilities located at Buildings 1, 2, 5, 16, 17, and 18.

M. Research and development in the licensee's facilities located at Buildings 1, 2, 5, 16, 17 and 18.

N. For storage only.

Table 2-6

List of CE'S AEC Licenses

DATE	LICENSE NO.	STATUS	AUTHORIZED FORM OF MATERIAL
1956	6-217-1	Terminated (date uncertain)	Polonium 210 sealed source
1957	6-217-2	Terminated 1962	Polonium 210 sealed source
1957	6-217-3	Terminated (date uncertain)	Iridium 192 sealed source
1957	6-217-4	Terminated 1965	Polonium 210 sealed source
1957	6-217-5	Terminated 1959	Polonium 210 sealed source
1958	6-217-6	Renewed (1968) as 06-00217-06	Various activation products
1958	6-217-7	Terminated 1960	Cobalt 60 sealed source
1962	6-217-8	Terminated 1970	Cobalt 60 sealed source
Date Uncertain	6-217-9	Uncertain	Uncertain
1958	C-4451	Terminated 1959	Th/Mg Scrap alloy
1961	STB-50	Uncertain	Th/Mg Scrap alloy

Table 3-1
Population Distribution Within A 50-Mile Radius
Of The CE Windsor Site

Miles	People	Cumulative People
0 to 5	56,429	56,429
5 to 10	286,341	342,770
10 to 20	868,651	1,211,421
20 to 30	717,683	1,929,104
30 to 40	532,391	2,461,495
40 to 50	963,795	3,425,290

Source: U.S. Department of Energy, Office of Naval Reactors. Environmental Impact Statement, S1C Prototype Reactor Plant Disposal, Volume 1 of 2, November 1996.

Table 3-2
Socioeconomic Factors For The
Town Of Windsor And The Hartford Region

	Town of Windsor	Hartford County
Geographic Area (square miles) [1]	29.6	735.5
Population [2]	28,237	857,183
White Population [2]	18,387	659,192
African American Population [2]	7648	99,936
Asian Population	887	20,775
Hispanic Population [2]	1405	98,968
Other Race Population [2]	643	57,481
Population Density (persons/square mile, 1998) [1]	930.1	1121.6
Civilian Labor Force (1997) [1]	14,534	421,856
Average Individual Per Capita Income (1998) [1]	$24,694	$24,032
Percentage Unemployment (1997) [1]	5.0	5.6

* Population figures may add up to more than the total population because individuals may report more than one race.

Sources:

1. Town of Windsor Data: Connecticut State Department of Economic and Community Development, Connecticut Town Profiles 1998-99; 2000.
2. U.S. Census Bureau, Census 2000 Redistricting (Public Law 94-171) Summary File Matrices PL1, PL2, PL3, PL4.

Table 3-3
Monitoring Well Construction Details

WELL	AREA	AOC #	DIA (in.)	FEET BGS DEPTH	SCREEN LENGTH	GROUND ELEV	PVC ELEV	SCREENED SOIL	INSTALL DATE	INSTALL BY
PDU AREA WELL NETWORK										
MW-2	PDU	3	2	40	5	185.1	186.94	F-M SAND	3/2/1981	F&O
MW-3	PDU	3	2	40	10	184.9	186.62	F SAND/SILT	3/3/1981	F&O
MW-4	PDU	3	2	45	5	185.2	186.56	F SAND	3/6/1981	F&O
MW-10D	PDU	3	1.5	68	10	185.0	185.81	F SAND	7/8/1982	F&O
MW-11D	PDU	3	1.5	74	10	185.0	185.98	F SAND	7/9/1982	F&O
MW-12S	PDU	3	1.5	40	10	186.0	185.63	SAND	7/7/1982	F&O
MW-13S	PDU	3	1.5	49	10	180.0	182.79	F SAND/SILT	7/7/1982	F&O
MW-14S	PDU	3	2	35	10	180.0	183.30	F-C SAND	1987	F&O
MW-14D	PDU	3	2	70	10	180.0	179.77	F SAND/SILT	1987	F&O
MW-15S	PDU	3	2	45	10	180.0	178.70	F-C SAND	1987	F&O
MW-15D	PDU	3	2	100	10	180.0	178.41	F-C SAND	1987	F&O
PCE WELL NETWORK										
WW-1	Burial Area	21	2	44	20	193.8	194.92	F SAND	10/19/1990	GZA
WW-2	Burial Area	21	2	45	20	191.1	192.72	F SAND	10/19/1990	GZA
E-1	Woods	1	2	47.5	10	198.6	202.02	SILT/SAND	8/10/1985	F&O
MW-101	Building 17	1	2	24.5	5	191.7	196.00	F SAND/SILT	3/25/1991	ABB-ES
MW-102	HWSA	1	2	48	10	191.9	196.60	F SAND	3/27/1991	ABB-ES
MW-103	Leachfield	1	2	39	10	179.2	183.94	F SAND/SILT	3/28/1991	ABB-ES
KAPL WELL NETWORK										
MW-1	KAPL	NA	2	29	15	168.7	170.62	F-C SAND	8/23/1995	KAPL
MW-2	KAPL	NA	2	31	15	172.4	174.43	F-C SAND/GRAVEL	8/24/1995	KAPL
MW-3	KAPL	NA	2	33	15	174.5	176.58	F-C SAND	8/22/1995	KAPL
MW-4	KAPL	NA	2	36	15	176.8	178.75	F-C SAND/SILT	8/31/1995	KAPL
MW-5	KAPL	NA	2	40	15	182.0	183.90	F SAND	8/21/1995	KAPL
MW-6	KAPL	NA	2	38	15	179.3	181.09	F-C SAND/GRAVEL	8/30/1995	KAPL
MW-7A	KAPL	NA	2	29	10	177.3	179.29	F-C SAND/GRAVEL	8/28/1995	KAPL
MW-7B	KAPL	NA	2	39	5	177.3	179.27	F-C SAND	8/28/1995	KAPL
MW-8	KAPL	NA	2	36	15	176.7	178.94	F SAND	9/1/1995	KAPL
TOWN OF WINDSOR LANDFILL WELLS										
MW-20S	Windsor Landfill	NA	2	55	10	180.5	184.91	F SAND	1/8/1998	F&O
MW-23	Windsor Landfill	NA	2	40	10	175.8	177.33	F-C SAND	8/7/1998	F&O
MW-24	Windsor Landfill	NA	2	35	25	170.4	172.90	SILT/GRAVEL	8/6/1998	F&O
MW-25	Windsor Landfill	NA	2	37	10	179.1	181.69	SILT/GRAVEL	8/7/1998	F&O
MW-26	Windsor Landfill	NA	2	47	10	180.1	182.60	F SAND	8/5/1998	F&O
MW-27	Windsor Landfill	NA	2	40	10	173.8	176.16	SAND/GRAVEL	8/7/1998	F&O

WELL	AREA	AOC #	DIA (in.)	FEET BGS DEPTH	SCREEN LENGTH	GROUND ELEV	PVC ELEV	SCREENED SOIL	INSTALL DATE	INSTALL BY
LFI WELLS										
MW-601	Building 6A	6	2	22	10	181.5	181.04	F-C SAND	7/16/1998	HLA
MW-602	Building 6A	6	2	20	10	181.4	180.82	F SAND	7/15/1998	HLA
MW-603	Building 6A	6	2	19	10	179.5	179.21	F SAND/SILT	7/16/1998	HLA
MW-1201	Evaporator Line	12	2	12	5	181.8	181.54	F SAND/SILT	9/9/1998	HLA
MW-1202	Evaporator Line	12	2	32	10	183.6	182.99	SILT/GRAVEL	9/15/1998	HLA
MW-1203	Evaporator Line	12	2	30	10	182.0	181.71	F SAND	9/9/1998	HLA
MW-1204	Evaporator Line	12	2	30	10	181.5	181.71	F SAND	9/9/1998	HLA
RFI WELLS										
MW-105	Woods	1	2	55	10	199.0	201.89	SAND	7/15/1998	HLA
MW-106	Woods	1	2	37	10	200.9	202.99	SAND/TILL	7/16/1998	HLA
MW-109	Woods	1	2	43	15	195.1	194.95	SAND	7/16/1998	HLA
MW-110	Woods	1	2	30	10	196.4	196.31	SAND	4/27/1999	HLA
MW-118	Woods	1	2	50	10	193.8	196.58	SAND/TILL	4/27/1999	HLA
MW-301	PDU	3	2	38	10	193.9	193.68	SAND	4/27/1999	HLA
MW-302	PDU	3	2	28	15	172.5	175.36	STRATIFIED SAND	4/28/1999	HLA
MW-607	Building 6A	6	2	18	10	177.1	176.86	SAND	4/29/1999	HLA
MW-608	Building 6A	6	2	24	10	181.2	180.81	STRATIFIED SAND/GRAVEL	4/29/1999	HLA
MW-609	Building 6A	6	2	13	10	181.8	181.36	SAND	4/30/1999	HLA
MW-904	Building 3	9	2	28	10	179.8	179.28	STRATIFIED SAND/GRAVEL	5/3/1999	HLA
MW-905	Building 3	9	2	26	10	179.2	181.50	SAND	5/3/1999	HLA
MW-906D	Building 3	9	2	50	10	179.9	202.94	SAND	5/4/1999	HLA
MW-906S	Building 3	9	2	27	10	180.0	179.77	SAND	5/4/1999	HLA
MW-907	Building 3	9	2	28	10	180.8	180.44	SAND	5/5/1999	HLA
MW-1004D	Building 20	10	2	29	10	171.6	171.34	SAND/PEAT	5/5/1999	HLA
MW-1004S	Building 20	10	2	18	10	171.4	171.00	SAND	5/5/1999	HLA
MW-1005	Building 20	10	2	16	10	169.8	172.26	SAND	5/5/1999	HLA
MW-1006	Building 20	10	2	19	10	170.9	173.98	STRATIFIED SAND/PEAT	5/6/1999	HLA
MW-1007	Building 20	10	2	17	10	167.9	170.76	SAND	5/6/1999	HLA
MW-1106	WWTP	11	2	23	10	167.4	170.43	SAND	5/6/1999	HLA
MW-1208	Ind Waste Lines	12	2	19	10	180.6	180.28	TILL	5/6/1999	HLA
MW-1209	Ind Waste Lines	12	2	27.5	10	181.6	181.42	STRATIFIED SAND/GRAVEL/TILL	5/7/1999	HLA
MW-1210	Ind Waste Lines	12	2	22	10	180.2	182.68	STRATIFIED SAND/GRAVEL	5/10/1999	HLA
MW-1211	Ind Waste Lines	12	2	28	10	182.1	181.38	STRATIFIED SAND/GRAVEL	5/11/1999	HLA
MW-1212	Ind Waste Lines	12	2	29	10	181.5	181.08	STRATIFIED SAND/GRAVEL	5/11/1999	HLA

WELL	AREA	AOC #	DIA (in.)	FEET BGS DEPTH	SCREEN LENGTH	GROUND ELEV	PVC ELEV	SCREENED SOIL	INSTALL DATE	INSTALL BY
MW-1213	Ind Waste Lines	12	2	23	10	174.8	177.29	SAND	5/12/1999	HLA
MW-1214	Ind Waste Lines	12	2	20	10	173.1	172.51	SAND	5/12/1999	HLA
MW-1215	Ind Waste Lines	12	2	22	10	172.7	172.04	SAND	5/12/1999	HLA
MW-1216	Ind Waste Lines	12	2	22	10	172.1	171.76	STRATIIED SAND AND GRAVEL	5/13/1999	HLA
MW-1217	Ind Waste Lines	12	2	24	10	174.1	176.57	SAND	5/13/1999	HLA
MW-1218	Ind Waste Lines	12	2	26	10	175.4	178.29	SAND	5/13/1999	HLA
MW-1219	Ind Waste Lines	12	2	29	10	176.3	179.04	SAND	5/13/1999	HLA
MW-1220	Ind Waste Lines	12	2	34	10	184.0	185.96	SAND	5/13/1999	HLA
MW-1221	Ind Waste Lines	12	2	20	10	169.6	169.73	STRATIFIED SAND	5/14/1999	HLA
MW-1222	Ind Waste Lines	12	2	12.5	10	146.9	150.01	SAND	5/14/1999	HLA
WP-1401D	Site Brook	14	1	2	1	NM	121.28	STREAM BED	5/17/1999	HLA
WP-1401S	Site Brook	14	1	5	1	NM	121.18	STREAM BED	5/17/1999	HLA
WP-1402D	Site Brook	14	1	2	1	NM	141.20	STREAM BED	5/17/1999	HLA
WP-1402S	Site Brook	14	1	5	1	NM	141.30	STREAM BED	5/18/1999	HLA
WP-1403D	Site Brook	14	1	2	1	NM	144.00	STREAM BED	5/18/1999	HLA
WP-1403S	Site Brook	14	1	5	1	NM	144.07	STREAM BED	5/19/1999	HLA
MW-1603	Coal Storage	16	2	20	10	171.9	171.41	SAND	5/20/1999	HLA
MW-1804	Tank Farm	18	2	25	10	179.9	179.47	SAND	5/20/1999	HLA
MW-1805	Tank Farm	18	2	27	10	179.7	179.60	SAND	5/20/1999	HLA
MW-1806	Tank Farm	18	2	27	10	180.0	179.70	STRATIFIED SAND	5/21/1999	HLA
MW-1807	Tank Farm	18	2	27	10	180.0	179.74	SAND	5/21/1999	HLA
MW-1808	Tank Farm	18	2	23	10	179.8	179.14	SAND	5/21/1999	HLA
MW-1809	Tank Farm	18	2	23	10	179.5	179.24	STRATIFIED SAND	5/27/1999	HLA
MW-2102	Burial Area	21	2	51.5	10	201.1	203.98	SAND	5/27/1999	HLA
MW-2202	Gravel Pit	22	2	58	10	200.8	203.51	SAND	5/27/1999	HLA
MW-2401	Drainage Ditch	24	2	20	10	178.7	178.58	SAND	5/27/1999	HLA
MW-2402	Drainage Ditch	24	2	15	10	170.6	173.03	STRATIFIED SAND	5/27/1999	HLA
MW-E01	Site Wide	NA	2	21	10	172.8	175.37	SAND	5/27/1999	HLA
MW-E02D	Building 20	10	2	49.5	10	180.4	182.83	SAND	5/28/1999	HLA
MW-E02S	Building 20	10	2	29.5	10	180.1	182.84	STRATIFIED SSAND	5/28/1999	HLA
MW-E03	Building 20	10	2	27	10	171.9	174.33	SAND	6/2/1999	HLA

WELL	AREA	AOC #	DIA (in.)	FEET BGS DEPTH	SCREEN LENGTH	GROUND ELEV	PVC ELEV	SCREENED SOIL	INSTALL DATE	INSTALL BY
MW-N01	Site Wide	NA	2	38	10	138.5	140.91	STRATIFIED SAND/GRAVEL	6/3/1999	HLA
MW-N02	Site Wide	NA	2	28	10	130.0	132.68	SAND	6/3/1999	HLA
MW-N03D	Site Wide	NA	2	53	10	174.8	177.30	STRATIFIED SAND/GRAVEL	6/3/1999	HLA
MW-N03S	Site Wide	NA	2	35	10	175.0	177.42	SAND	6/3/1999	HLA
MW-N04D	Site Wide	NA	2	64.5	10	172.8	175.31	SAND	6/3/1999	HLA
MW-N04S	Site Wide	NA	2	32	10	172.3	175.15	SAND	6/4/1999	HLA
MW-N05	Site Wide	NA	2	65	120	201.9	203.26	SAND	6/4/1999	HLA
MW-N06	Site Wide	NA	2	47	10	185.2	184.90	SAND	6/8/1999	HLA
MW-S01	Site Wide	NA	2	24	10	177.0	177.57	STRATIFIED SAND/GRAVEL	6/8/1999	HLA
MW-S02	Site Wide	NA	2	20	10	175.4	177.80	SAND	6/8/1999	HLA
MW-W01	Site Wide	NA	2	29	10	189.0	191.45	SAND	6/8/1999	HLA
MW-W02D	Site Wide	NA	2	44	10	183.8	186.55	SAND	6/9/1999	HLA
MW-W02S	Site Wide	NA	2	24	10	183.7	186.27	SAND	6/9/1999	HLA
SUPPLEMENTAL RFI WELLS										
MW-1004DD*	Building 20	10	2	110	10	172.1	171.75	C SAND	10/21/1999	HLA
MW-1014	Building 20	10	2	20	10	169.6	172.75	SAND	10/18/1999	HLA
MW-2202D	Gravel Pit	22	2	85	15	201.4	203.64	SAND	10/26/1999	HLA
MW-N06D	Site Wide	50	2	67	10	179.1	181.63	F SAND	11/4/1999	HLA
MW-N07	Site Wide	NA	2	68	10	184.2	186.01	F SAND	10/26/1999	HLA
MW-N08D	Site Wide	NA	2	77	10	192.6	195.36	F SAND	10/21/1999	HLA
MW-N09D	Site Wide	NA	2	75	10	200.3	202.94	F SAND	11/9/1999	HLA
RFI 2000 WELLS										
MW-0144	Woods	01	2	48	10	NM	202.34	F SAND	8/24/2000	H-ESE
MW-0145	Woods	01	2	57	10	NM	200.13	F SAND	8/30/2000	H-ESE
MW-1004DI	Building 20	10	2	48	10	NM	172.21	F SAND	8/24/2000	H-ESE
MW-1004DD	Building 20	10	2	104	10	NM	172.16	SAND/GRAVEL	9/18/2000	H-ESE
MW-1005D	Building 20	10	2	30	10	NM	172.81	F SAND	9/11/2000	H-ESE
MW-1005DI	Building 20	10	2	59	10	NM	172.64	STRATIFIED SAND	9/8/2000	H-ESE
MW-1006DI	Building 20	10	2	38	10	NM	174.47	M SAND	8/30/2000	H-ESE
MW-1007DI	Building 20	10	2	47.5	10	NM	171.3	STRATIFIED SAND/PEAT	8/29/2000	H-ESE
MW-1016	Building 20	10	2	23	10	NM	174.16	M SAND	9/5/2000	H-ESE
MW-1016DI	Building 20	10	2	47	10	NM	173.22	F SAND	8/31/2000	H-ESE
MW-1214DI	Ind Waste Lines	12	2	30	10	NM	172.36	STRATIFIED SAND\GRAVEL	8/8/2000	H-ESE
MW-1217DD	Ind Waste Lines	12	2	104	10	NM	176.83	STRATIFIED SAND/GRAVEL	8/22/2000	H-ESE
MW-1225	Ind Waste Lines	12	2	10	10	NM	149.96	STRATIFIED SAND	8/1/2000	H-ESE
MW-1810	Tank Farm	18	2	29	10	NM	180.06	F SAND	8/15/2000	H-ESE
MW-1811	Tank Farm	18	2	33	10	NM	185.64	F SAND	8/22/2000	H-ESE
MW-1812	Tank Farm	18	2	31	10	NM	180.69	F SAND	8/11/2000	H-ESE
MW-1813	Tank Farm	18	2	28	10	NM	177.02	F SAND	8/17/2000	H-ESE

WELL	AREA	AOC #	DIA (in.)	FEET BGS DEPTH	SCREEN LENGTH	GROUND ELEV	PVC ELEV	SCREENED SOIL	INSTALL DATE	INSTALL BY
MW-2601	Fmr Target Ranges	26	2	47	10	NM	186.64	F SAND	9/13/2000	H-ESE
MW-W03	Site Wide	NA	2	25	10	NM	178.94	SAND	9/11/2000	H-ESE
MW-W04	Site Wide	NA	2	22	10	NM	179.63	SAND	9/7/2000	H-ESE
AOC 10 SUPPLEMENTAL WELLS										
MW-1018	Building 20	10	2	17.5	10	NM	172.8	SILTY SAND	4/20/2001	H-ESE
MW-0E05S	Building 20	10	2	31	10	NM	176.47	F SAND	3/22/2001	H-ESE
MW-0E05DI	Building 20	10	2	87	10	NM	176.15	SAND	4/17/2001	H-ESE
MW-0E06S	Building 20	10	2	27	10	NM	177.26	SAND	3/27/2001	H-ESE
MW-0E07D	Building 20	10	2	58	10	NM	176.72	SAND	4/18/2001	H-ESE
MW-0E07DI	Building 20	10	2	94	10	NM	175.87	SAND	4/18/2001	H-ESE
MW-0E08D	Building 20	10	2	53	10	NM	167.97	SAND	4/26/2001	H-ESE
MW-0E08DI	Building 20	10	2	92	10	NM	167.29	F SAND	4/25/2001	H-ESE
MW-0E09DI	Building 20	10	2	81	10	NM	163.21	F SAND	4/25/2001	H-ESE
MW-0E10	Building 20	10	2	52.2	10	NM	175.61	F SAND	6/27/2001	H-ESE
RFI 2001 WELLS										
MW-1507	Buildings 17 and 21	15	2	20	10	NM	183.32	F SAND	4/23/2001	H-ESE

Notes:

ABB-ES	"ABB Environmental Services, Inc."	HLA	Harding Lawson Associates	M	MEDIUM
AOC	Area of concern	HWSA	Hazardous waste storage area	C	COARSE
GZA	"Goldberg, Zoino & Associates"	F&O	"Fuss & O'Neill, Inc."	NM	NOT MEASURED
IWL	Industrial Waste Line	ELEV	Feet above Mean Sea Level	NA	Not Applicable
H-ESE	Harding ESE	F	FINE		

* MW-1004DD was redrilled on 9/18/00 --- Screened Auger Sampled/No Boring Info

Table 3-4
Groundwater Elevations

Well	GW Elev. (feet MSL) 8/28/2000	GW Elev. (feet MSL) October 2-3, 2000	"GW Elev. (feet MSL) April 26, 2001"	"GW Elev. (feet MSL) October 3-4, 2001"	"GW Elev. (feet MSL) January 24, 2002"	"GW Elev. (feet MSL) April 24, 2002"
MW-2	159.65	159.17	158.45	158.31	156.2	155.34
MW-3	158.6	158.12	157.95	157.29	155.48	154.78
MW-4	153.85	153.57	154.82	154.14	151.94	151.89
MW-6	-	-	150.95	151.02	148.36	147.5
MW-10D	153.83	153.54	154.81	-	152.01	151.93
MW-12S	158.65	-	-	156.08	154.24	153.5
MW-13S	-	-	-	151.99	150.98	150.89
MW-20S	142.71	142.27	143.45	142	140.68	140.46
MW-21	-	-			189.99	189.99
MW-22	-	-			183.44	146.08
MW-23	146.9	146.81	148.11	146.62	145.36	145.22
MW-24	143.45	143.25	144.12	143.04	143.34	142.18
MW-25	152.94	152.6	154.42	151.99	149.87	149.85
MW-26	143.73	143.48	144.67	143.14	142.08	141.83
MW-27	145.18	144.96	146.19	144.66	143.6	143.39
MW-101	172.96	172.06	176.24	171.07	168.98	169.76
MW-102	155.54	155.15	154.63	154.45	152.77	151.84
MW-103	152.02	151.59	152.1	150.92	149.55	148.86
MW-105	155.47	155.06	154.24	154.23	152.32	151.23
MW-106	171.17	171.36	172.88	171.29	170.51	170.18
MW-109	164.46	163.92	163.9	157.17	161.07	159.93
MW-110	160.06	159.54	159.4	158.77	156.8	155.69
MW-118	153.21	152.21	152.13	151.12	148.98	147.63
MW-144	161.94	162.64	165.28	161.78	159.34	158.58
MW-145	181.66	157.12	156.89	156.24	154.2	152.81
MW-301	162.51	161.75	161.71	161.47	157.74	156.86
MW-302	155.04	154.6	156.1	154.06	152.9	152.68
MW-601	166.23	165.97	166.69	165.44	165.35	165.63
MW-602	168.75	168.23	169.72	167.82	166	167.82
MW-603	167.86	167.51	169.36	165.92	165.21	164.37
MW-607	167.77	167.38	169.06	166.45	165.66	166.2
MW-608	164.69	164.43	165.51	164.26	163.06	163.08
MW-609	176.05	175.45	176.81	-	173.62	175.42
MW-904	158.95	158.89	160.15	158.34	157.19	157.17
MW-905	163.08	159.74	160.61	158.68	161.01	159.49
MW-906D	159	158.94	160.13	158.44	157.29	157.27
MW-906S	158.77	158.66	159.88	158.22	157.04	157.04
MW-907	158.79	158.93	159.97	159.26	157.04	157.06
MW-1004S	163.59	163.08	164.3	161.08	160.7	162.67
MW-1004D	163.63	163.12	164.34	161.42	160.79	162.74
MW-1004DI	159.34	158.39	158.41	157.81	158.19	157.56
MW-1004DD	158.73	158.13	159.26	157.7	156.61	156.51
MW-1005	161.43	161.5	161.49	160.61	159.84	161.42
MW-1005D	-	158.18	159.31	157.64	156.67	156.35
MW-1005DI	-	158.21	159.39	157.7	156.68	156.64

Well	GW Elev. (feet MSL) 8/28/2000	GW Elev. (feet MSL) October 2-3, 2000	"GW Elev. (feet MSL) April 26, 2001"	"GW Elev. (feet MSL) October 3-4, 2001"	"GW Elev. (feet MSL) January 24, 2002"	"GW Elev. (feet MSL) April 24, 2002"
MW-1006	159.26	159.12	160.38	158.55	158.23	159.46
MW-1006DI	-	157.84	158.97	157.34	156.32	156.25
MW-1007	158.18	158.29	159.46	157.61	157.59	158.31
MW-1007DI	-	157.59	158.75	157.08	156.17	156.05
MW-1014	161.98	158.9	159.6	158.32	159.93	158.52
MW-1016	-	159.63	160.51	158.64	158.41	159.76
MW-1016DI	-	157.92	159.12	157.49	156.43	156.35
MW-1018	-	-	-	159.37	160.18	158.5
MW-1106	152	151.12	154.13	149.85	148.77	150.21
MW-1201	173.8	173.81	175.04	173.83	173.69	174.54
MW-1202	158.91	158.84	159.99	155.14	157.04	157.01
MW-1203	-	158.56	159.76	158	156.92	156.86
MW-1204	159.16	158.98	160.26	158.41	157.23	157.24
MW-1208	172.22	171.78	172.98	171.18	170.63	172.09
MW-1209	161.19	161.15	161.32	161.06	160.78	161.13
MW-1210	163.91	163.58	164.88	162.99	161.86	161.94
MW-1211	161.93	161.66	162.98	161.07	159.67	159.87
MW-1212	159.55	159.34	160.68	158.75	157.47	157.5
MW-1213	162.98	159.3	160.19	157.9	159.48	157.91
MW-1214	164.48	158.51	159.56	157.75	160.31	157.87
MW-1214DI	-	158.05	159.12	157.36	159.2	157.2
MW-1215	160.91	157.77	158.73	157.14	158.59	157.18
MW-1216	156.95	156.63	158.02	156.13	155.2	155.14
MW-1217	156.36	156.23	157.41	154.77	154.81	154.67
MW-1217DD	-	155.88	157.07	155.44	154.51	154.38
MW-1218	154.85	154.62	156.16	154.24	153.38	153.36
MW-1219	154.15	153.88	155.35	154.97	152.43	152.21
MW-1220	158.07	157.6	157.59	156.63	154.95	154.13
MW-1221	151.84	151.55	152.21	151.13	dry	dry
MW-1222	144.65	144.55	145.04	144.41	144.03	143.86
MW-1225	144.75	144.77	145.08	144.57	144.29	144.09
MW-1507	-	-	-	168.15	164.49	164.36
MW-1603	159.02	158.44	159.56	158.18	159.95	158.88
MW-1804	159.64	159.59	160.63	159.02	157.68	157.79
MW-1805	158.76	-	160.27	158.15	NA	-
MW-1806	159.33	159.32	160.37	158.74	-	157.56
MW-1807	159.32	159.29	160.53	158.7	157.37	157.62
MW-1808	162.42	162.05	164.12	160.96	-	159.79
MW-1809	159.88	159.74	161.74	159.06	-	157.78
MW-1810	159.46	159.21	160.8	158.77	157.49	157.56
MW-1811	159.99	159.87	161.7	159.21	157.58	157.44
MW-1812	160.87	160.46	162.68	159.78	158.29	158.25
MW-1813	157.96	157.67	159.52	157.41	156.02	155.98
MW-2102	157.6	157.19	156.89	156.42	164.65	153.68
MW-2202	151.8	151.56	151.1	150.89	149.41	148.59
MW-2202D	152.03	151.76	151.33	151.07	149.62	148.76
MW-2401	167.47	166.92	168.35	166.18	164.54	164.66

Well	GW Elev. (feet MSL) 8/28/2000	GW Elev. (feet MSL) October 2-3, 2000	"GW Elev. (feet MSL) April 26, 2001"	"GW Elev. (feet MSL) October 3-4, 2001"	"GW Elev. (feet MSL) January 24, 2002"	"GW Elev. (feet MSL) April 24, 2002"
MW-2402	167.99	167.5	169.81	166.65	165.08	165.11
MW-2601	144.99	144.82	145.23	144.5	143.68	143.35
MW-N01	109.12	109.03	109.54	108.66	107.94	107.57
MW-N02	110.67	110.58	110.97	110.35	109.81	109.53
MW-N03S	147.63	147.25	148.38	147	145.77	145.41
MW-N03D	147.57	147.28	148.32	146.97	145.75	145.38
MW-N04D	149.54	149.35	150.68	149.25	147.8	147.59
MW-N04S	149.6	149.39	150.75	149.32	147.87	147.77
MW-N05	145.93	145.69	145.91	145.29	144.33	143.91
MW-N06	140.7	140.56	140.74	140.03	138.89	138.26
MW-N06D	140.36	140.2	140.41	139.69	138.57	137.93
MW-N07	151.05	150.71	150.61	147.06	148.64	147.85
MW-N08D	155.53	155.23	154.92	154.6	152.75	151.86
MW-N09D	143.29	143.06	143.28	142.71	141.8	141.37
E1	160.95	160.41	160.22	159.55	157.5	156.09
MW-E01	158.4	158.24	159.43	157.56	156.01	155.93
MW-E02D	157.26	157.13	158.6	156.7	155.57	155.52
MW-E02S	157.72	157.44	159.45	156.98	155.8	155.7
MW-E03	154.7	154.48	155.78	154.11	152.92	152.76
MW-E05S	-	-	-	156.98	155.82	155.75
MW-E05DI	-	-	-	156.18	155.1	155.08
MW-E06S	-	-	-	154.65	154	154.03
MW-E07DI	-	-	-	156	153.62	154.96
MW-E07D	-	-	-	155.25	155.82	154.51
MW-E08DI	-	-	-	153.98	153.04	152.94
MW-E08D	-	-	-	153.56	153.7	152.62
MW-E09DI	-	-	-	152.95	152.06	151.96
MW-E10DI	-	-	-	152.61	151.66	151.56
MW-S01	161.2	161	162.24	160.39	159.05	159.11
MW-S02	161.49	160.8	162.52	159.98	158.48	158.53
MW-W01	167.45	166.95	168.51	166.19	164.49	164.45
MW-W02D	167.41	166.94	168.33	166.12	164.35	164.14
MW-W02S	167.42	166.95	168.31	166.11	164.4	164.16
MW-W03	-	-	159.23	158.24	156.43	155.71
MW-W04	-	-	-	162.49	160.61	160.06
WW-1	158.59	158.24	157.81	-	152.54	154.72
WW-2	152.62	152.28	151.98	-	152.99	148.91
MW-101 (Parcel B)	-	-			174.57	174.57
MW-102 (Parcel B)	-	-			174.94	174.94
MW-103 (Parcel B)	-	-			175.99	175.99
MW-104 (Parcel B)	-	-			175.72	175.72
WP-1401S	-	119.55	119.2	119.08	119	118.93
WP-1401D	-	120.38	119.01	118.17	118.06	118.1
WP-1402S	-	139.37	139.49	139.38	139.23	139.3

Well	GW Elev. (feet MSL) 8/28/2000	GW Elev. (feet MSL) October 2-3, 2000	"GW Elev. (feet MSL) April 26, 2001"	"GW Elev. (feet MSL) October 3-4, 2001"	"GW Elev. (feet MSL) January 24, 2002"	"GW Elev. (feet MSL) April 24, 2002"
WP-1402D	-	140.3	140.63	140.29	139.3	140.03
WP-1403S	-	141.96	142.48	142.29	142.06	141.91
WP-1403D	-	141	143.56	142.88	141.7	142.24

Notes

MSL Mean Sea Level
"-" not measured

Table 3-5

Groundwater Vertical Gradients at Monitoring Well Clusters

Well ID	PVC Riser Elevations	Top of Screen Elevations	Wellscreen Separation[1]	October-99		November-99		August-00		October-00		April-01		May-01		July-01	
				Elevation Head	Vertical Gradient	Elevation Head	Vertical Gradient	Elevation Head	Vertical Gradient	Elevation Head	Vertical Gradient	Elevation Head	Vertical Gradient	Elevation Head	Vertical Gradient	Elevation Head	Vertical Gradient
MW-906S	179.77	162.77		158.63		158.15		158.77		158.66		159.88		--		159.22	
MW-906D	179.89	139.89	22.88	158.18	0.019667832	158.39	-0.01048951	159	-0.010052448	158.94	-0.012237762	160.13	-0.010926573	--		159.46	-0.01048951
MW-1004S	171	163		160.68		163.01		163.59		163.08		164.3		164.35		163.55	
MW-1004D	171.34	152.34	10.66	162.67	-0.186679174	163.02	-0.000938086	163.63	-0.003752345	163.12	-0.003752345	164.34	-0.003752345	164.34	0.000938086	163.63	-0.00750469
MW-1004DI	172.21	134.21	28.79	--		--		159.34	0.147620702	158.39	0.162903786	158.41	0.204584925	159.31	0.175060785	158.81	0.1646405
MW-1004DD	171.75	77.75	85.25	--		154.67	0.097829912	158.73	0.057008798	158.73	0.051026393	159.26	0.059120235	158.75	0.06568915	158.55	0.058651026
MW-1005	172.94	166.94		163.11		162.66		161.43		161.5		164.49		161.64		161.03	
MW-1005D	172.8	152.81	14.13	--		--		--		158.18	0.234961076	159.3	0.367303609	159.2	0.172682236	158.6	0.171974522
MW-1005DI	172.65	123.64	43.3	--		--		--		158.21	0.075981524	159.4	0.117551963	159.25	0.055196305	158.68	0.054272517
MW-1006	173.98	164.98		158.49		159.24		159.26		159.12		160.38		159.98		159.32	
MW-1006DI	174.47	146.47	18.51	--		--		--		157.84	0.06915181	158.97	0.076175041	158.77	0.06537007	158.25	0.057806591
MW-1007	170.76	163.76		159.53		158.43		158.18		158.29		159.46		158.86		157.86	
MW-1007DI	171.3	133.8	29.96	--		--		--		157.59	0.023364486	158.75	0.023698264	158.5	0.012016021	158.01	-0.005006676
MW-1016	174.16	161.16		--		--		--		159.63		160.51		160.61		--	
MW-1016DI	173.22	136.22	24.94	--		--		--		157.92	0.068564555	159.12	0.055733761	158.92	0.06776263	--	
MW-1214	172.51	162.51		164.63		160.8		164.48		158.51		159.56		159.66		158.94	
MW-1214DI	172.36	152.36	10.15	--		--		--		158.05	0.045320197	159.12	0.043349754	159.21	0.044334975	158.51	0.042364532
MW-1217	176.57	162.57		155.89		155.96		156.36		156.23		157.41		157.17		156.65	
MW-1217DD	176.83	82.83	79.74	--		--		--		155.88	0.004389265	157.07	0.004263858	157.72	-0.006897417	156.33	0.004013042
WP-1401S	121.18	120.28		119.37		119.4		--		119.55		119.2		--		--	
WP-1401D	121.28	117.18	3.1	118.56	0.261290323	118.53	0.280645161	--		120.38	-0.267741935	119.01	0.061290323	--		--	
WP-1402S	141.3	140.2		139.52		139.43		--		139.37		139.49		--		--	
WP-1402D	141.2	137.3	2.9	140.4	-0.303448276	140.27	-0.289655172	--		140.3	-0.320689655	140.63	-0.393103448	--		--	
WP-1403S	144.07	143		142.85		142.66		--		141.96		142.48		--		--	
WP-1403D	144	140.07	2.93	143.19	-0.116040956	143.09	-0.146757679	--		141	0.327645051	143.56	-0.368600683	--		--	
MW-2202	203.51	155.51		149.9		149.94		151.8		151.56		151.1		--		--	
MW-2202D	203.64	115.5	40.01	--		--		--		--		151.33	-0.005748563	--		--	
MW-E02S	182.84	163.34		156.61		156.84		157.72		157.44		159.45		159.04		158.04	
MW-E02D	182.83	143.33	20.01	156.37	0.011994003	156.59	0.012493753	157.26	0.022988506	157.13	0.015492254	158.6	0.042478761	158.38	0.032983508	157.61	0.021489255
MW-E05S	176.47	152.81		--		--		--		--		--		158.97		158.01	
MW-E05DI	176.15	97.19	55.62	--		--		--		--		--		157.55	0.025530385	157.03	0.017619561
MW-E07D	176.72	126.24		--		--		--		--		--		155.27		155.82	
MW-E07DI	175.87	90.02	36.22	--		--		--		--		--		157.37	-0.057979017	156.84	-0.028161237
MW-E08D	167.97	122.32		--		--		--		--		--		154.72		155.45	
MW-E08DI	167.29	83.08	39.24	--		--		--		--		--		155.29	-0.014525994	153.61	0.046890928
MW-N03S	177.42	152.42		146.99		146.92		147.63		147.25		148.38		--		--	
MW-N03D	177.3	134.3	18.12	146.94	0.002759382	146.9	0.001103753	147.57	0.003311258	147.28	-0.001655629	148.32	18.12	--		--	
MW-N04S	175.15	143.15		149.49		149.26		149.6		149.39		150.75		--		--	
MW-N04D	175.31	120.81	22.34	149.4	0.004028648	149.18	0.003581021	149.54	0.002685765	149.35	0.00179051	150.68	26.06333333	--		--	
MW-N06	182.06	145.06		139.65		139.44		140.7		140.56		140.74		--		--	
MW-N06D	181.63	144.63	0.43	--		139.11	0.76744186	140.36	0.790697674	140.2	0.837209302	140.41	0.417352941	--		--	
MW-W02S	186.27	172.27		165.72		165.74		167.42		166.95		168.31		--		--	
MW-W02D	186.55	152.55	19.72	165.7	0.001014199	165.74	0	167.41	0.000507099	166.94	0.000507099	168.33	-0.001014199	--		--	
MW-E1	202.02	155.02		157.97		157.91		160.95		160.41		160.22		--		--	
MW-145	200.13	147	8.02	--		---		---		157.12	0.410224439	156.89	0.41521197	--		--	

Notes: 1. Wellscreen separation is the difference (feet) between the top of a wellscreen and the top of the corresponding shallowest wellscreen.
"---," indicates not measured"
All elevations are in feet above Mean Sea Level

Table 3-6
Estimated Groundwater Velocities

Location	Well	K ft/d	Gradient ft/ft	Velocity ft/d	Velocity ft/y
Waste Pad Area	MW-0106	3.3	0.008	0.088	32
Building 6A	MW-0601	1.51	0.026	0.13	48
	MW-0602	0.36	0.026	0.031	11
	MW-0607	2.5	0.0074	0.062	23
Building 3	MW-0907	30.5	0.000457	0.046	17
Industrial Waste Lines	MW-1213	2.5	0.0042	0.035	13
	MW-1214	40	0.002	0.27	98
	MW-1218	4.5	0.00375	0.056	21
	MW-1219	5.3	0.0044	0.078	29
Tank Farm (near Building 3)	MW-1804	16.6	0.000736	0.041	15
Waste Pad Area (near Drum Burial)	MW-0110	11.6	0.00318	0.12	45
Former Gravel Pit	MW-2204	6.5	0.0089	0.19	70

Notes: An effective porosity of 0.3 is assumed for all locations.
K - estimated hydraulic conductivity
ft - feet
d - day
y - year

Table 4-1
Building 2 Complex
Radiological History And Potential Contaminants

Building/Location	History	COCs	Notes
Building 1	"Used for R&D for the Sites first criticality experiments. Decontaminated and released for occupancy in 1960. After 1960, used for x-ray analyses, photography and radioactive materials storage."	"Byproduct materials, Uranium, LEU"	Building 1 was decontaminated and restored in May 1960. Repeated scrubbing and painting were necessary to release the building. radioactivity may still be present beneath the paint. Decontamination and decommissioning of internal systems and components to the release criteria of RG 1.86 was completed in 2002. Authorization has been received to decontaminate and deconstruct the above-grade structure has been received.
Vault 1	"Interim extended radiological waste storage area, storage area for sealed neutron sources, and records storage area."	"Byproduct materials, LEU"	
Vault 2	"Housed the Sites first criticality experiments, and later used for highly radioactive materials and x-ray analysis of reactor components."	"Byproduct materials, LEU"	
Room 3	"Health physics office, lay down work areas for criticality experiments, and a radioactive materials storage area."	"Byproduct materials, LEU"	
Room 4	Restroom. Sanitary line may be potentially contaminated.	none	

Building/Location	History	COCs	Notes
Room 5	"Used for miscellaneous activities involving radioactive materials, including a reactor component assembly area for the ""hot cell"", a dark room, and later a receiving and storage area for radioactive materials."	"Byproduct materials, LEU"	
Room 6	Contains the ventilation system for Control Zone 5 (CZ5). Previously used as a Photostat laboratory where reactor components were photographed.	"Byproduct materials, LEU"	
Control Zone 5	"Control room for experimental criticality projects in the late 1950s. Refurbished in the mid-1980s, and original surfaces are potentially contaminated."	"Byproduct materials, LEU"	
Boiler Room and Office Area	Boiler Room and Office Area	none	
Outside Storage Area	Historically used to store radioactive materials. This area also included the CZ5 recirculation/ventilation exhaust system for Building 1.	"Byproduct materials, LEU"	
Industrial Waste System	"Effluents containing radiological and chemical constituents were discharged to the industrial waste lines, and eventually to the Site brook."	"Byproduct materials, HEU, LEU, DU"	
Building 1A	Used for the storage of radioactive materials and occasionally used for a radioactive work area.	Byproduct materials	

Building/Location	History	COCs	Notes
North Side	"Originally used to house a prototype LLRW incinerator (never used for RAM). Later used to support non-radiological electronics work, and finally as a temporary radioactive materials work and storage area."	Byproduct materials	
South Side	Inventory area and storage area for packaged radioactive materials. Southeast corner contained a contamination cell under negative pressure with a portable HEPA unit.	Byproduct materials	
Building 2	Building 2 was used for R&D.	"Uranium, Byproduct materials"	
Reactor Cell 1	Used for reactor research including the sodium-cooled reactor projects and seismic tests on fuel assemblies containing DU.	DU	
Reactor Cell 2	"Used for multiple criticality experiments, and then used to support work conducted under a Byproduct License."	"Natural Uranium, LEU, HEU, and byproduct materials"	
Self Contained HEPA system	Processes exhaust air from Cell 2.	"Natural Uranium, LEU, HEU, and Byproduct materials"	
Control Zone 1 (CZ1)	Area used for decontamination and repair of reactor servicing equipment.	"Natural Uranium, LEU, HEU, and Byproduct materials"	

Building/Location	History	COCs	Notes
Cell 1 Sump	Joined the industrial waste lines from Cell 1 and Building 1.	"Natural Uranium, LEU, HEU, and Byproduct materials"	
Cell 2 Sump	Joined the industrial waste lines from Cell 2 and Building 1.	"Natural Uranium, LEU, HEU, and Byproduct materials"	
Cell 2 Mezzanine	Contained former glove boxes used for uranium carbide research that was enriched to 14 percent.	Uranium Carbide and Byproduct materials	
Building 2 Tower	"Contained at least one glove box, used to support uranium carbide research."	Uranium carbide and Byproduct materials	
Control Room 1	"Housed controls and necessary equipment to support reactor experiments. After Cell 1 was converted to a seismic testing, the control room was used as a health physics laboratory, and was used to conduct strength tests on non-contaminated coupons collected from commercial power reactors."	"Natural Uranium, LEU, HEU, and Byproduct materials"	
Control Room 2	"Used after the mid-1980s as a Health Physics office. Maintained radiologically clean, although some residual radioactivity was periodically detected."	"Natural Uranium, LEU, HEU, and Byproduct materials"	

Building/Location	History	COCs	Notes
Storage Vault	"Storage of radioactive materials including fuel from the FCE and S1C, uranium from Building 3, UO2 pellets, fuel rods, pellets containing uranium enriched to less than 20 percent, contaminated steam generator tubes and nuclear sources."	"Natural Uranium, LEU, HEU, and Byproduct materials"	
Machine Shop	Used to fabricate radiologically clean components. Later was converted to a training room.	none	
Whole Body Count Room	"From mid-1950s to 1991, used as a general-purpose work space. After 1991, it was used as the whole body count room."	none	
Data Management and Storage Area	Used for clean storage and data management.	none	
Second Floor/ Office Area	"Primarily office spaces, rest rooms, and conference rooms."	none	
Building 2A	Buildings used to support the commercial nuclear power outage services.	Byproduct materials	
Control Zones	"Control zones, serviced by negative pressure ventilation systems, were designed to use two HEPA banks to recirculate the air. Floor tiles were periodically replaced and the residual radioactivity was not completely removed."	Byproduct materials	
Sanitary Drain System	"Potentially contaminated from historic DOE operations and subsequent outage support operations. Prior to 1975, Building 2 waste discharged to the leach field north of the Woods Area. "	"Natural Uranium, LEU, HEU, and Byproduct materials"	

Building/Location	History	COCs	Notes
Outdoor Storage Area	"A fence surrounds the building forming a ""controlled access area"". Equipment was packaged for shipment and properly stored. In 1996, the area was expanded and soil samples were collected to characterize the area."	"U235, Co60, Cs137, Th228, Th232"	

Notes:

AEC - Atomic Energy Commission
cm^2 - centimeters square
COCs - Compounds of Concern
DOE - Department of Energy
dpm - disintegrations per minute
DU - Depleted uranium
HEPA - high efficiency particulate air
HEU - high enriched uranium
HSA - Historical Site Assessment
HVAC - heating, ventilation, air conditioning "
LEU - low enriched uranium
MDC - Metropolitan District Commission
SEM - scanning electron microscope
SNM - Special Nuclear Materials
R&D - Research and Development
uCi - microcuries
UST - underground storage tank
VCA - Voluntary Corrective Action

Table 4-2
Building 5 Complex
Radiological History And Potential Contaminants

Building	Location	History	COCs	Notes
Building 5		"R&D laboratories originally used to support AEC contracts, and later to support commercial nuclear R&D."	Uranium and Byproduct materials	Decontamination and decommissioning of internal systems and components to the release criteria of RG 1.86 was completed in 2002. Authorization has been received to decontaminate and deconstruct the above-grade structure has been received.
Building 5 North Wing (Nuclear Laboratories)				
	Room 212	"Metal Testing, Emission Spectrograph Laboratory. This laboratory was equipped with an autoclave for high temperature testing of metal properties. Zirconium properties were routinely tested in this laboratory, and later the room contained two spectrometers."	Byproduct materials	
	Rooms 213 through 216	Office spaces.	none	
	Rooms 217 and 218	Laboratories used to support the Metallurgy and Chemistry groups.	LEU and byproduct materials	
	Rooms 219 and 220	Office spaces.	none	

Building	Location	History	COCs	Notes
	Rooms 221 through 224	Fuel Fabrication and Ceramics Laboratory used to support Nuclear Fuels Manufacturing from the mid-1950s to 1992. These rooms were set up to develop pellet fabrication processes.	"LEU, Cs, Co"	Final Survey for uranium contamination in Rooms 224A and B was not approved by the NRC because the floor drains and joints were not completed characterized. CE decontaminated the byproduct contamination the area was free released.
	Room 225	Chemistry laboratories to support the uranium and byproduct material operations.	LEU	
	Room 226	Laboratory set up to support the Metallography operations.	Uranium and Byproduct materials	
	Room 227	Chemistry laboratories to support the uranium and byproduct material operations. Contained an autoclave testing area to test the physical properties of metals used in welds and steam generator tubes.	LEU and byproduct materials	

Building	Location	History	COCs	Notes
	Rooms 228 and 229	Chemistry laboratories to support the uranium and byproduct material operations. Room 229 (x-ray diffraction laboratory) was used primarily for manufacturing and commercial operations rather than R&D.	Uranium and Byproduct materials	
	Rooms 230 and 231	Chemistry laboratories to support the uranium and byproduct material operations.	Uranium and Byproduct materials	
	Rooms 232 and 233	Metallurgy laboratories for analyzing fuel samples. These rooms were equipped with electron microscopes and processed radioactive samples from the late 1950s through the early 1960s.	Uranium and Byproduct materials	
	Rooms 234 and 235	Metallurgy laboratories for analyzing fuel samples.	Uranium and Byproduct materials	
Building 5 Central Wing (Machine Shop and High Bay Testing Area)				

Building	Location	History	COCs	Notes
	Room 105A	Fuel Element Loading Shop (previously located in the northeast corner of the high-bay test area). The Fuel Loading Shop was removed in the early 1970s.	Uranium and Byproduct materials	
	Room 106	"Room used to weld fuel rod tubes, using an automatic heliarc process."	Uranium and Byproduct materials	
	Rooms 107 and 107A	Final Inspection Area (originally one room and later divided). Inspected elements were then returned to the vault for storage and off-Site shipment.	Uranium and Byproduct materials	
Building 5 East Corridor (Administrative Area)				
	Rooms 100 through 103	Offices for management and human resource personnel	none	

Building	Location	History	COCs	Notes
	Room 108	Offices for management and human resource personnel. Later converted to a health physics office and was contaminated when archived samples were being counted on the gamma spectroscopy system.	Uranium and Byproduct materials	
	"Rooms 210A, B, and C"	Men's bathroom and showers. Showers used to decontaminate personnel.	LEU	
	Rooms 301 through 304	Offices for management and human resource personnel	none	
Building 5 South Wing (Health Physics and Radiochemistry)				
	Rooms 305 and 306	"Radiochemistry count room and laboratory. These rooms supported all Site operations with a liquid scintillation counter, gamma spectroscopy system and various alpha/beta counters."	"HEU, LEU and byproduct materials"	

Building	Location	History	COCs	Notes
	Room 308	Storage of DU and small quantities of low enriched uranium (less than 5 percent).	"DU, LEU"	
	Room 309	Storage of DU and small quantities of low enriched uranium (less than 5 percent).	"Uranium (DU, LEU)"	
	Room 310	Standards library.	none	
	Room 311	Radiochemistry laboratory used for low-level sample analyses. Mainly used for urine samples. Also used for storing liquid nuclear standards. This room was later used for contaminated equipment within the fume hood.	"Pu, LEU and Byproduct materials"	
	Room 312	Electronics laboratory that contained at least one hood for radiological work.	LEU and byproduct materials	
	Room 313	Tool storage / Component supply crib.	none	

Building	Location	History	COCs	Notes
	Room 314	Shipping and storage area set up to receive or ship all source or special nuclear materials. Once contained welding area for fuel assemble parts (grid plates) prior to fuel loading.	LEU and byproduct materials	
	Room 315	Originally a machine shop and was converted to a computer room.	none	
	Rooms 317 and 318	"Electronics and maintenance laboratories, and routinely contained contaminated components. Later these rooms were used as offices."	LEU and byproduct materials	
	Room 319	Health Physics and Ultrasonics Laboratory.	LEU and byproduct materials	
	Room 320A	X-ray vault and was also used for the storage of nuclear sources and depleted and enriched uranium.	"DU, LEU, HEU, and byproduct materials"	
	Room 321	"Radiochemistr y ""hot"" laboratory."	LEU and byproduct materials	

Building	Location	History	COCs	Notes
Building 5 Waste Pad Area		"Former waste pad located adjacent to the northwest corner of Building 5, under the current footprint of Building 16. Stored spent ion exchange resin and regenerant, SNM scrap material in pellet or powder form, used absolute air filters, laboratory samples, used acid solution, combustibles scraps, and miscellaneous equipment."	Uranium	"All items placed on the pad were supposed to be in AEC approved, sealed containers."
Building 5 Ventilation		"Ventilation provided by an air handler. There were eight stacks, all equipped with sampling locations and six of the eight stacks used absolute HEPA filters. The two exceptions were the hydrogen burn-off stack and the radiochemistry laboratory."	LEU and byproduct materials	
Building 15		Carpentry Shop	none	

Building	Location	History	COCs	Notes
Building 16		"Boronometer Test Area. This building also contained a tool crib, stock cage, shop and office area. DU was handled in Building 16 to simulate the mass of a full bundle assembly."	"Sealed neutron (RaBe, RuBe, AmBe sources), DU. LEU, and byproduct materials"	
Building 18		"Building 18 is a high bay, connected to the west end of the central wing of Building 5. The high bay contained multiple test loops that used DU to simulate the weight of a standard fuel assembly bundle. "	Uranium (DU) and byproduct materials	
Sanitary System		The last sanitary line cleanout before the discharge to manhole S-19 is in Room 305.	LEU and byproduct materials	Manhole S-19 was remediated by CE in 1996. See subsection 5.2 of this HSA Report.

Building	Location	History	COCs	Notes
Radiological Waste Water		"Radiological waste water first discharged to dilution tanks in Building 6A. After 1961, the wastewater routed to dilution tanks in Building 6. In 1992, all industrial waste water discharged to MDC, and CE used the former industrial waste lines to carry radiological waste water to an evaporator in Building 6. (Subsection 5.1)"	"HEU, LEU, DU, and byproduct materials. "	See subsection 5.1 of this HSA Report.

Notes:

AEC - Atomic Energy Commission
cm^2 - centimeters square
COCs - Compounds of Concern
DOE - Department of Energy
dpm - disintegrations per minute
DU - Depleted uranium
HEPA - high efficiency particulate air
HEU - high enriched uranium
HSA - Historical Site Assessment
HVAC - heating, ventilation, air conditioning "
LEU - low enriched uranium
MDC - Metropolitan District Commission
SEM - scanning electron microscope
SNM - Special Nuclear Materials
R&D - Research and Development
uCi - microcuries
UST - underground storage tank
VCA - Voluntary Corrective Action

Table 4-3
Summary Of Radiological Surveys Of Building 17
Performed Prior To Renovations

Location	Comments
Office Area	No contamination was found that exceeded the release criteria levels from either license. The highest fixed contamination levels were 180 alpha and 2,000 beta dpm/100cm^2. No loose contamination was detected.
Compressor Building	No contamination was found that exceeded the release criteria levels from either license. The highest fixed contamination levels were 40 alpha and 280 beta dpm/100cm^2. No loose contamination was detected.
Bay A	No contamination was found that exceeded the release criteria levels from either license. The highest fixed contamination levels were 60 alpha and 1,300 beta dpm/100cm^2. No loose contamination was detected.
Bay B	No contamination was found that exceeded the release criteria levels from either license. The highest fixed contamination levels were 60 alpha and 1,600 beta dpm/100cm^2. No loose contamination was detected.
Bay B / C Partition	The partition between B and C is a concrete block wall that was shot blasted to remove paint from the C side wall surface. During the characterization BNFL removed some inner sections of the blocks from the wall for sampling. BNFL did not remove whole blocks or put holes through the wall in order to obtain samples.
Bay C	Fixed contamination levels up to 3,600 alpha and 6,800 beta dpm/100cm^2 with an average of 880 alpha and 6,300 beta dpm/100cm^2 were found.
Bay C Roof	Volumetric samples of the roof tar material indicated contamination levels up to 16,200 alpha and 7,000 beta dpm/100cm^2.

Notes:

cm^2: centimeter square
dpm: disintegrations per minute
BNFL: British Nuclear Fuel, Ltd.

Table 4-4
June 2002 Radiological Groundwater Sampling Program
Background Location Results

Location ID	Field Sample ID	Sample Date	Sample Type	Analysis Method Units	Cesium-137 Gamma pCi/L	Cobalt-60 Gamma pCi/L	$Uranium_{tot}$ Calc pCi/L	Uranium-234 Alpha pCi/L	Uranium-235 Alpha pCi/L	Uranium-238 Alpha pCi/L
MW-E01	MW0E01	06/24/02	FS	Result	3.26	2.98	0.321	0.218	0.0479	0.0551
				Qual	U	U	U	U	U	U
				Error	1.93	1.7		0.199	0.11	0.109
MW-E03	MW0E03	06/25/02	FS	Result	1.19	0.409	0.6522	0.466	0.0852	0.101
				Qual	U	U	U	U	U	U
				Error	1.77	1.6		0.263	0.137	0.135
MW-2401	MW2401	06/24/02	FS	Result	0.369	1.04	0.612	0.242	0.233	0.137
				Qual	U	U		U	J	U
				Error	1.78	1.71		0.196	0.183	0.136
MW-S02	MW0S02	06/24/02	FS	Result	3.04	3.74	0.3597	0.236	0.0512	0.0725
				Qual	U	U	U	U	U	U
				Error	1.87	2.05		0.182	0.101	0.116
MW-W01	MW0W01	06/24/02	FS	Result	1.38	2.87	0.6546	0.227	0.33	0.0976
				Qual	U	U	U	U	U	U
				Error	1.79	1.76		0.185	0.083	0.142

Summary Statistics for Results							
Analysis	Method	Units	Freq.	Min	Max	Mean	Median
Cesium-137	Gamma	pCi/L	0 / 5	0	0	1.68	1.79
Cobalt-60	Gamma	pCi/L	0 / 5	0	0	2.22667	2.05
Uranium, Total	Calc.	pCi/L	1 / 5	0	0	0.6546	0.6546
Uranium-234	Alpha	pCi/L	0 / 5	0	0	0.198	0.185
Uranium-235	Alpha	pCi/L	1 / 5	0	0	0.17133	0.101
Uranium-238	Alpha	pCi/L	0 / 5	0	0	0.11853	0.116

NOTES:
"Uranium, Total / Calc. = sum of U-isotopes"
U = not detected; value represents the sample MDA.
J = estimated value; QC out of criteria
Sample Type = FS (field sample); FD (field duplicate
Gamma = EPA Method 901.1

Freq. = frequency of detections
Min = minimum result observed
Max = maximum result observed
Mean = arithmetic average of result
Median = arithmetic middle of the result series

Alpha = DOE EML HASL Method 300

Background = to assess background water quality conditions

Table 4-5
June 2002 Radiological Groundwater Sampling Program

Location ID	Field Sample ID	Sample Date	Sample Type	Analysis Method Units	Cesium-137 Gamma pCi/L	Cobalt-60 Gamma pCi/L	Uranium, Total Calc pCi/L	Uranium-234 Alpha pCi/L	Uranium-235 Alpha pCi/L	Uranium-238 Alpha pCi/L
E-1	E00001	06/27/02	FS	Result	2.67	4.48	1.291	0.486	0.42	0.385
				Qual	U	J	U	U	U	U
				Error	1.54	1.85		0.177	0.136	0.121
MW-0101	MW0101	06/27/02	FS	Result	0.628	2.9	0.9878	0.678	0.0358	0.274
				Qual	U	U		J	U	J
				Error	1.88	1.6		0.318	0.101	0.195
MW-0102	MW0102	06/28/02	FS	Result	0.083	0.647	0.5603	0.0723	0.236	0.252
				Qual	U	U	U	U	U	U
				Error	1.67	1.89		0.129	0.0679	0.0693
MW-0103	MW0103	06/26/02	FS	Result	3.67	0.614	0.3281	0.175	0.0321	0.121
				Qual	U	U	U	U	U	U
				Error	2.15	1.86		0.194	0.108	0.149
MW-0105	MW0105	06/26/02	FS	Result	1.25	3.75	0.707	0.182	0.401	0.124
				Qual	U	U	U	U	U	U
				Error	1.85	2.46		0.188	0.0612	0.153
MW-0106	MW0106	06/26/02	FS	Result	0.363	1.21	0.4028	0.169	0.0518	0.182
				Qual	U	U	U	U	U	U
				Error	2.05	2.25		0.144	0.083	0.0196
MW-0109	MW0109	06/27/02	FS	Result	3.36	3.58	0.0738	0.0222	0.0294	0.0222
				Qual	U	U	U	U	U	U
				Error	2.02	2.08		0.0591	0.0577	0.0591
MW-0110	MW0110	06/27/02	FS	Result	3	3.21	0.2071	0.148	0.0179	0.0412
				Qual	U	U	U	U	U	U
				Error	1.76	2.07		0.135	0.0474	0.0661
MW-0118	MW0118	06/27/02	FS	Result	1.44	0.177	0.6323	0.319	0.0633	0.25
				Qual	U	U		J	U	U
				Error	0.806	0.946		0.219	0.11	0.033
MW-0144	MW0144	06/27/02	FS	Result	1.58	3.05	0.525	0.315	0.086	0.124
				Qual	U	U		J	U	U
				Error	1.94	1.81		0.206	0.125	0.139

Location ID	Field Sample ID	Sample Date	Sample Type	Analysis Method Units	Cesium-137 Gamma pCi/L	Cobalt-60 Gamma pCi/L	Uranium, Total Calc pCi/L	Uranium-234 Alpha pCi/L	Uranium-235 Alpha pCi/L	Uranium-238 Alpha pCi/L
MW-0301	MW0301	06/27/02	FS	Result	2.92	0.302	0.524	0.259	0.008	0.257
				Qual	U	U	U	U	U	U
				Error	1.68	1.92		0.193	0.0607	0.0361
MW-0302	MW0302	06/28/02	FS	Result	0.701	4.71	0.2275	0.0484	0.0761	0.103
				Qual	U	U	U	U	U	U
				Error	5.65	2.56		0.0776	0.0952	0.11
MW-0601	MW0601	06/27/02	FS	Result	1.62	3.52	0.478	0.233	0.145	0.1
				Qual	U	U	U	U	U	U
				Error	2.56	2		0.181	0.162	0.151
MW-0602	MW0602	06/27/02	FS	Result	3.32	1.61	1.3	0.626	0.287	0.387
				Qual	U	U		J	U	J
				Error	2.01	1.83		0.31	0.0722	0.222
MW-0602	MW0602DUP	06/27/02	FD	Result	2.8	2.03	0.8761	0.57	0.0811	0.225
				Qual	U	U		J	U	J
				Error	1.65	1.89		0.28	0.112	0.17
MW-0603	MW0603	06/27/02	FS	Result	0.202	2.66	0.4967	0.191	0.0617	0.244
				Qual	U	U	U	U	U	U
				Error	2.93	1.84		0.165	0.0949	0.0627
MW-0607	MW0607	06/24/02	FS	Result	0.137	2.87	0.6283	0.285	0.0543	0.289
				Qual	U	U	U	U	U	U
				Error	2.17	2.09		0.202	0.102	0.038
MW-0608	MW0608	06/26/02	FS	Result	3.3	3.12	0.4196	0.156	0.0366	0.227
				Qual	U	U	U	U	U	U
				Error	1.91	1.63		0.164	0.0719	0.0244
MW-0609	MW0609	06/28/02	FS	Result	3.57	0.455	0.809	0.41	0.08	0.319
				Qual	U	U		J	U	U
				Error	2.15	1.92		0.253	0.143	0.237
MW-0609	MW0609DUP	06/28/02	FD	Result	3.37	0.823	0.978	0.535	0.079	0.364
				Qual	U	U		J	U	J
				Error	1.91	1.82		0.256	0.0987	0.209
MW-0904	MW0904	06/25/02	FS	Result	1.03	3.87	0.744	0.371	0.122	0.251
				Qual	U	U		J	U	U
				Error	2	2.11		0.227	0.129	0.196

Location ID	Field Sample ID	Sample Date	Sample Type	Analysis Method Units	Cesium-137 Gamma pCi/L	Cobalt-60 Gamma pCi/L	Uranium, Total Calc pCi/L	Uranium-234 Alpha pCi/L	Uranium-235 Alpha pCi/L	Uranium-238 Alpha pCi/L
MW-0905	MW0905	06/25/02	FS	Result	0.515	5.16	0.7185	0.291	0.0295	0.398
				Qual	U	U		J	U	J
				Error	7.3	3.01		0.221	0.0784	0.261
MW-0905	MW0905DUP	06/25/02	FD	Result	0.654	2.69	0.3649	0.0829	0.162	0.12
				Qual	U	U	U	U	U	U
				Error	1.85	1.48		0.147	0.161	0.181
MW-0906S	MW0906S	06/25/02	FS	Result	2.55	1.21	0.6718	0.519	0.0765	0.0763
				Qual	U	U	U	U	U	U
				Error	1.57	1.59		0.274	0.118	0.117
MW-0907	MW0907	06/25/02	FS	Result	0.458	3.61	0.459	0.178	0.115	0.166
				Qual	U	U	U	U	U	U
				Error	1.96	2.01		0.188	0.144	0.165
MW-1004S	MW1004S	06/25/02	FS	Result	3.35	3.55	0.8867	0.688	0.109	0.0897
				Qual	U	U		J	U	U
				Error	1.9	2.65		0.339	0.145	0.124
MW-1005	MW1005S	06/25/02	FS	Result	3.3	3.28	0.59	0.387	0.018	0.185
				Qual	U	U		J	U	J
				Error	1.81	1.67		0.246	0.0966	0.157
MW-1006	MW1006S	06/25/02	FS	Result	0.186	1.63	0.4203	0.109	0.246	0.0653
				Qual	U	U	U	U	U	U
				Error	1.81	2.04		0.13	0.0653	0.1
MW-1007	MW1007	06/28/02	FS	Result		5.18	0.09364	0.0693	0.00854	0.0158
				Qual	R	U	U	U	U	U
				Error		2.91		0.107	0.0647	0.063
MW-1014	MW1014	06/25/02	FS	Result	6.43	5.15	0.35033	0.325	0.0169	0.00843
				Qual	U	U	U	U	U	U
				Error	8.1	2.94		0.212	0.0907	0.0639
MW-1016	MW1016S	06/26/02	FS	Result	0.184	1.34	0.4105	0.137	0.0555	0.218
				Qual	U	U		U	U	J
				Error	2.09	1.78		0.145	0.104	0.178
MW-1018	MW1018	06/26/02	FS	Result	3.27	2.71	0.30721	0.225	0.00901	0.0732
				Qual	U	U		J	U	U
				Error	1.9	1.65		0.128	0.0518	0.082

Location ID	Field Sample ID	Sample Date	Sample Type	Analysis Method Units	Cesium-137 Gamma pCi/L	Cobalt-60 Gamma pCi/L	Uranium, Total Calc pCi/L	Uranium-234 Alpha pCi/L	Uranium-235 Alpha pCi/L	Uranium-238 Alpha pCi/L
MW-1106	MW1106	06/24/02	FS	Result	3.25	0.175	0.433	0.12	0.102	0.211
				Qual	U	U	U	U	U	U
				Error	2.02	1.73		0.136	0.117	0.0227
MW-1201	MW1201	06/28/02	FS	Result	2.94	1.09	11.48	9.37	0.34	1.77
				Qual	U	U			J	
				Error	1.65	1.82		1.78	0.223	0.545
MW-1202	MW1202	06/28/02	FS	Result	1.3	3.24	0.699	0.187	0.242	0.27
				Qual	U	U	U	U	U	U
				Error	2.07	1.73		0.159	0.0666	0.191
MW-1203	MW1203	06/26/02	FS	Result	3.21	3.69	0.1945	0.044	0.0215	0.129
				Qual	U	U	U	U	U	U
				Error	2	2.1		0.101	0.0571	0.141
MW-1204	MW1204	06/27/02	FS	Result	1.53	0.0448	0.2471	0.161	0.0585	0.0276
				Qual	U	U		J	J	U
				Error	2.08	2.04		0.0859	0.0475	0.0489
MW-1208	MW1208	06/28/02	FS	Result	0.442	0.974	0.9575	0.475	0.0245	0.458
				Qual	U	U		J	U	J
				Error	2.03	1.89		0.264	0.065	0.254
MW-1209	MW1209	06/28/02	FS	Result	2.42	0.389	1.267	0.611	0.112	0.544
				Qual	U	U		J	U	J
				Error	1.44	1.9		0.286	0.134	0.264
MW-1209	MW1209DUP	06/28/02	FD	Result	3.39	3.57	1.2521	0.687	0.0801	0.485
				Qual	U	U		J	U	J
				Error	2.15	4.04		0.296	0.1	0.245
MW-1210	MW1210	06/26/02	FS	Result	2.51	3.26	0.5747	0.318	0.164	0.0927
				Qual	U	U	U	U	U	U
				Error	2.11	1.99		0.208	0.0146	0.106
MW-1211	MW1211	06/27/02	FS	Result	3.53	3.68	0.09624	0.079	0.00124	0.016
				Qual	U	U	U	U	U	U
				Error	2.03	2.32		0.127	0.0673	0.0639

Location ID	Field Sample ID	Sample Date	Sample Type	Analysis Method Units	Cesium-137 Gamma pCi/L	Cobalt-60 Gamma pCi/L	Uranium, Total Calc pCi/L	Uranium-234 Alpha pCi/L	Uranium-235 Alpha pCi/L	Uranium-238 Alpha pCi/L
MW-1212	MW1212	06/27/02	FS	Result	0.904	1.05	0.2993	0.132	0.132	0.0353
				Qual	U	U	U	U	U	U
				Error	1.4	1.94		0.137	0.123	0.0799
MW-1213	MW1213	06/25/02	FS	Result	0.803	1.73	0.27999	0.00859	0.0234	0.248
				Qual	U	U	U	U	U	U
				Error	6.43	2.64		0.0651	0.0621	0.0682
MW-1214	MW1214	06/25/02	FS	Result	3.24	3.46	0.5029	0.238	0.0859	0.179
				Qual	U	U	U	U	U	U
				Error	1.91	1.86		0.18	0.119	0.0686
MW-1214	MW1214DUP	06/25/02	FD	Result	0.851	3.23	0.4608	0.339	0.0158	0.106
				Qual	U	U	U	U	U	U
				Error	2.32	1.83		0.215	0.0846	0.112
MW-1215	MW1215	06/26/02	FS	Result	6.17	2.46	0.3302	0.0391	0.234	0.0571
				Qual	U	U	U	U	U	U
				Error	5.76	2.39		0.11	0.0252	0.107
MW-1216	MW1216	06/26/02	FS	Result	0.301	2.83	0.4966	0.168	0.0166	0.312
				Qual	U	U	U	U	U	U
				Error	2.36	1.74		0.157	0.0659	0.0987
MW-1217	MW1217	06/27/02	FS	Result	0.474	1.04	0.1674	0.128	0.0227	0.0167
				Qual	U	U	U	U	U	U
				Error	1.58	1.71		0.134	0.0604	0.0896
MW-1217	MW1217DUP	06/27/02	FD	Result	3.47	3.22	0.1963	0.135	0.0154	0.0459
				Qual	U	U	U	U	U	U
				Error	2.06	1.72		0.16	0.0612	0.106
MW-1218	MW1218	06/26/02	FS	Result	3.3	2.63	0.2005	0.101	0.0389	0.0606
				Qual	U	U	U	U	U	U
				Error	1.87	1.9		0.139	0.088	0.0844
MW-1219	MW1219	06/26/02	FS	Result	3.33	3.17	0.1932	0.0377	0.126	0.0295
				Qual	U	U	U	U	U	U
				Error	1.84	1.77		0.0853	0.133	0.0579
MW-1222	MW1222	06/24/02	FS	Result	1.19	2.84	0.2292	0.0935	0.0217	0.114
				Qual	U	U	U	U	U	U
				Error	1.65	1.68		0.115	0.0577	0.113

Location ID	Field Sample ID	Sample Date	Sample Type	Analysis Method Units	Cesium-137 Gamma pCi/L	Cobalt-60 Gamma pCi/L	Uranium, Total Calc pCi/L	Uranium-234 Alpha pCi/L	Uranium-235 Alpha pCi/L	Uranium-238 Alpha pCi/L
MW-1225	MW1225	06/24/02	FS	Result	2.86	0.752	0.1586	0.0987	0.0261	0.0338
				Qual	U	U	U	U	U	U
				Error	1.78	2.24		0.132	0.0967	0.0952
MW-1507	MW1507	06/27/02	FS	Result	1.39	3.42	0.3687	0.011	0.298	0.0597
				Qual	U	U	U	U	U	U
				Error	1.88	2.12		0.0834	0.0856	0.112
MW-1603	MW1603	06/26/02	FS	Result	0.0487	3.7	0.226	0.0161	0.18	0.0299
				Qual	U	U	U	U	U	U
				Error	5.69	2.07		0.0864	0.0598	0.0844
MW-1810	MW1810	06/26/02	FS	Result	0.204	3.1	1.532	0.938	0.153	0.441
				Qual	U	U		J	U	J
				Error	1.71	1.78		0.353	0.143	0.236
MW-2	MW2	06/26/02	FS	Result	0.233	0.332	0.4077	0.302	0.0197	0.086
				Qual	U	U		J	U	U
				Error	1.79	1.64		0.214	0.0784	0.132
MW-2	MW2DUP	06/26/02	FD	Result	3.17	2.86	0.5215	0.251	0.0655	0.205
				Qual	U	U		J	U	U
				Error	1.89	1.78		0.197	0.105	0.182
MW-2102	MW2102	06/28/02	FS	Result	3.42	2.31	0.31065	0.286	0.00865	0.016
				Qual	U	U	U	U	U	U
				Error	1.96	2.03		0.197	0.0656	0.0638
MW-2202	MW2202	06/28/02	FS	Result	2.93	0.00473	0.74	0.286	0.198	0.256
				Qual	U	U		J	U	U
				Error	1.91	1.83		0.201	0.0234	0.0659
MW-2202	MW2202DUP	06/28/02	FD	Result	1.12	0.618	0.484	0.185	0.239	0.06
				Qual	U	U	U	U	U	U
				Error	1.87	2		0.157	0.0657	0.104
MW-3	MW3	06/25/02	FS	Result	0.334	1.25	0.4626	0.208	0.0946	0.16
				Qual	U	U	U	U	U	U
				Error	1.69	1.84		0.201	0.131	0.169
MW-4	MW4	06/26/02	FS	Result	3.19	2.93	0.4985	0.305	0.101	0.0925
				Qual	U	U	U	U	U	U
				Error	1.79	1.65		0.199	0.129	0.114

Location ID	Field Sample ID	Sample Date	Sample Type	Analysis Method Units	Cesium-137 Gamma pCi/L	Cobalt-60 Gamma pCi/L	Uranium, Total Calc pCi/L	Uranium-234 Alpha pCi/L	Uranium-235 Alpha pCi/L	Uranium-238 Alpha pCi/L
MW-N04D	MW0N04D	06/26/02	FS	Result	1.1	0.292	1.1565	0.546	0.0735	0.537
				Qual	U	U		J	U	J
				Error	1.78	1.62		0.272	0.118	0.257
MW-N04S	MW0N04S	06/26/02	FS	Result	0.593	3.92	0.3246	0.203	0.0971	0.0245
				Qual	U	U	U	U	U	U
				Error	2.07	2.28		0.172	0.111	0.0652
MW-N07	MW0N07	06/25/02	FS	Result	3.1	0.367	0.2106	0.0551	0.0917	0.0638
				Qual	U	U	U	U	U	U
				Error	1.8	2.24		0.104	0.127	0.102
MW-N07	MW0N07DUP	06/25/02	FD	Result	1.29	1.03	0.2033	0.139	0.037	0.0273
				Qual	U	U	U	U	U	U
				Error	4.27	1.62		0.171	0.125	0.101
WP-1401D	WP1401D	06/25/02	FS	Result	3.07	0.578	0.622	0.376	0.127	0.119
				Qual	U	U		J	U	U
				Error	1.72	1.75		0.235	0.152	0.152
WP-1401S	WP1401S	06/25/02	FS	Result	1.3	3.14	1.04	0.647	0.219	0.174
				Qual	U	U		J	J	J
				Error	1.92	1.65		0.29	0.166	0.142
WP-1402D	WP1402D	06/25/02	FS	Result	0.613	2.8	0.3215	0.0906	0.0979	0.133
				Qual	U	U	U	U	U	U
				Error	1.57	2.01		0.145	0.123	0.141
WP-1402S	WP1402S	06/25/02	FS	Result	1.07	2.99	0.9176	0.614	0.0296	0.274
				Qual	U	U		J	U	U
				Error	2.03	1.6		0.287	0.0582	0.189
WP-1403D	WP1403D	06/25/02	FS	Result	2.83	3.13	1.101	0.976	0.104	0.021
				Qual	U	U		J	U	U
				Error	1.67	1.69		0.355	0.111	0.0558
WP-1403S	WP1403S	06/27/02	FS	Result	1.28	0.0698	18.566	17.5	0.839	0.227
				Qual	U	U			J	U
				Error	1.96	1.62		3.01	0.359	0.192
WW-1	MWWW1	06/28/02	FS	Result	0.534	3.47	0.1093	0.086	0	0.0233
				Qual	U	U	U	U	U	U
				Error	1.83	2.18		0.125	0.0604	0.062

Location ID	Field Sample ID	Sample Date	Sample Type	Analysis Method Units	Cesium-137 Gamma pCi/L	Cobalt-60 Gamma pCi/L	Uranium, Total Calc pCi/L	Uranium-234 Alpha pCi/L	Uranium-235 Alpha pCi/L	Uranium-238 Alpha pCi/L
WW-2	MWWW2	06/28/02	FS	Result	0.392	0.689	0.4478	0.159	0.0458	0.243
				Qual	U	U	U	U	U	U
				Error	2.98	1.96		0.148	0.0862	0.0321

NOTES:
"Uranium, Total / Calc. = sum of U-isotopes"
U = not detected; value represents the sample MDA.
J = estimated value; QC out of criteria
Sample Type = FS (field sample); FD (field duplicate)
Gamma = EPA Method 901.1
Alpha = DOE EML HASL Method 300

Table 4-6

Impacted Buildings and Locations

Buildings and Locations that are impacted	Status
Building 2 Complex Buildings 1, 1A, 2, & 2A and surrounding soils and utilities	Building decontamination complete. NRC Approval for above grade demolition.
Building 5 Complex Buildings 5, 16, & 18 and surrounding soils and utilities	Building decontamination complete. Submitted report to NRC for approval.
Building 17 and surrounding soils and utilities	Building decontamination complete. Preparing report for demolition approval by NRC.
Building 6A and surrounding soils and utilities	Ready for spot decontamination
Sanitary system	Ready for final status survey
Storm water system	Ready for final status survey
Great Pond	Ready for final status survey
Goodwin Pond	Ready for final status survey
Equipment Storage Yard	Ready for final status survey
Former WWTP	Ready for final status survey
Small Pond	Ready for final status survey
Digester Sludge Piles	Ready for final status survey
Former Gravel Pit	Ready for final status survey
Southeast Parcel	Ready for final status survey

Ex. 4

Ex. 4

Ex. 4

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Ex. 4

Table 10-1
Types of Radiation Detection Instrumentation
Decommissioning Plan
CE Windsor Site, Windsor Connecticut

Measurement	Meter	Detector
Direct alpha	Multipurpose scaler/ratemeter	ZnS(Ag) scintillation
Direct beta	Multipurpose scaler/ratemeter	Dual phosphor ZnS(Ag) scintillator
Direct alpha/beta/gamma	Multipurpose scaler/ratemeter	Gas filled (Geiger-Mueller) pancake
Removable	Computer software	Gas-flow proportional
Exposure rate	Multipurpose scaler/ratemeter or, integral with detector	NaI(Tl) scintillator Ion chamber or NaI(Tl) scintillator
Gamma spectrometry or spectroscopy	Computer software	HPGe

Table 14-1
Final Status Survey Investigation Levels
Decommissioning Plan
CE Windsor Site, Windsor Connecticut

Survey Unit Classification	Investigate When Sample Result:	Investigate When Scanning Measurement:
Class 1	>$DCGL_{EMC}$	>$DCGL_{EMC}$
Class 2	>DCGL	>DCGL
Class 3	>fraction of DCGL	>Minimum Detectable Concentration