CONNECTICUT YANKEE ATOMIC POWER COMPANY

**HADDAM NECK PLANT**
362 INJUN HOLLOW ROAD • EAST HAMPTON, CT 06424-3099

February 24, 2005
Docket No. 50-213
CY-05-054

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Haddam Neck Plant
Characterization Report for the West Section of the Excavation Associated with the Northeast Protected Area Grounds

Connecticut Yankee Atomic Power Company (CYAPCO) herein provides the enclosed characterization report for the west section of the excavation associated with the Northeast Protected Area Grounds. This technical support document describes the methodology and means used to perform radiological characterization of soil, subsurface structures and exposed bedrock in the area of interest. The discussion includes an evaluation of survey and sampling results to date relative to the expectations established during survey design. The results of this evaluation will be used to determine the nature and extent of contamination in this area. Finally, this document provides data for input to the radiological assessment. The radiological assessment establishes the final radiological condition of the area prior to backfill as described in the License Termination Plan.

The characterization effort in this area has identified residual activity on exposed surfaces of concrete and footings. Strontium-90 and tritium have been identified in and on concrete surfaces. No residual contamination is believed to be present in bedrock, however, strontium-90 and tritium are assumed to be present for the purposes of characterization and future radiological assessment. Cobalt-60 and cesium-137 are the principal radionuclides identified in soil. Further soil remediation is ongoing.

The report was discussed during a teleconference with the NRC Staff on February 22, 2005 in preparation for an NRC inspection and independent verification by ORISE in April 2005.

NMSS01

If you should have any questions, please contact me at (860) 267-3938.

Sincerely,

Gerard van Noordennen

Gerard P. van Noordennen
Regulatory Affairs Manager

Enclosure

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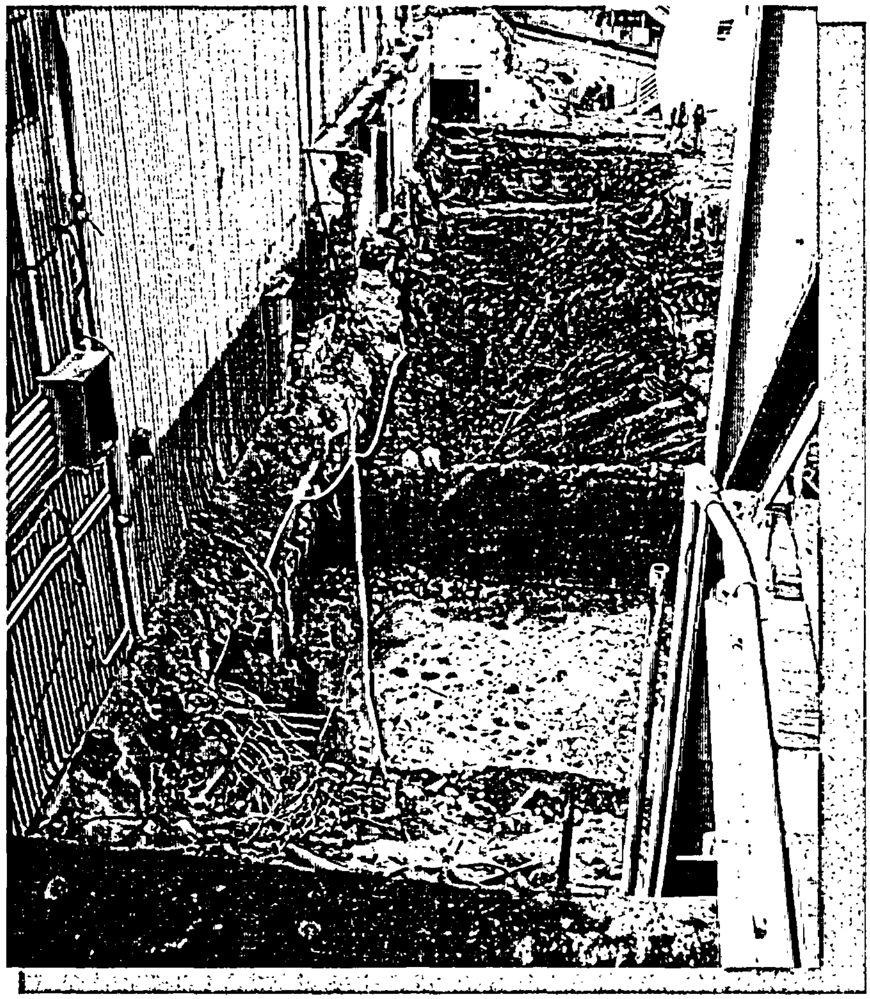
February 2005

# Connecticut Yankee Atomic Power Company



# Characterization Report

## West Section of the Excavation Associated with the Northeast Protected Area Grounds

## Haddam Neck Plant
## Haddam, Connecticut

REMOVE · RESTORE · REUSE
CY DECOMMISSIONING PROJECT

CONNECTICUT YANKEE ATOMIC POWER COMPANY

# Connecticut Yankee Decommissioning

# Health Physics Department Technical Support Document

**HP Number: CY-HP-0192 Revision #: 0**

**Subject: Characterization Report for the West Section of the Excavation Associated with the Northeast Protected Area Grounds**

**Date: 02/21/05**

Performed By: J. McCarthy Date: 02/21/05

Reviewed By: Date: 2/22/05

Approved By: Date: 2/22/05

**Characterization Report for the West Section of the Excavation Associated with the Northeast Protected Area Grounds**

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**Characterization Report for the West Section of the Excavation Associated with the Northeast Protected Area Grounds**

# 1. INTRODUCTION

## 1.1 Purpose of Technical Support Document (TSD)

To describe the methodology and means used to perform radiological characterization of remaining soil; subsurface structures and footings; and exposed bedrock in the area of interest. The discussion will include an evaluation of survey and sampling results relative to the expectations established during survey design. The output of this product will be a determination of the nature and extent of contamination. Finally, this TSD will provide data for input to the radiological assessment. The radiological assessment establishes the final radiological condition of the area prior to backfill as described by the License Termination Plan (LTP) (Reference 4.1).

# 2. DISCUSSION

## 2.1 Background

The general approach to decommissioning the site was changed in 2003 as described in Section 3 of the LTP. The previous plan was to decontaminate, perform Final Status Survey (FSS) and then demolish the System, Structures and Components (SSCs) as appropriate. The revised strategy is to decontaminate to permit demolition using appropriate controls, dispose of the majority of the SSCs at regulated disposal sites, and perform FSS on the remaining subsurface structures and footings and backfill. Generally, the scope is to remove SSCs to 4 feet below ground surface (bgs), perform a radiological assessment and backfill.

The scope of work is somewhat expanded for the demolition of Health Physics (HP) Facility, Primary Auxiliary Building (PAB), Waste Disposal Building (WDB); Residual Heat Removal (RHR) pit and Tank Farm SSCs. These will generally be demolished to bedrock, and the waste materials generated will be disposed of at a licensed disposal facility. A radiological assessment will be performed on remaining soil, subsurface structures and footings, and exposed bedrock in the excavation. An overall depiction of the excavation and affected SSCs is provided in Attachment 5.1.

The overall radiological assessment of the excavation is sub-divided into two sections. The two (2) sections, referred to as east and west, are defined by a physical boundary which is the west wall of the RHR pit (refer to Attachment 5.1). The east section contains the footprint of the Tank Farm SSCs and RHR pit. The west section contains the footprint of the WDB, PAB and HP Facility. Each section will receive an assessment appropriate for the existing conditions (i.e., extent of exposed bedrock, remaining subsurface structures and footings, remaining soil, etc), the expected radionuclides of concern (e.g., Sr-90, Cs-137), and the applicable target level for volumetric samples in picocuries per gram (pCi/g) or areal direct measurements in disintegrations per minute per 100 square centimeters (dpm/100 $cm^2$). As such, this characterization report is limited to one (1) section of the excavation, in this case, the west end. A little more than half (½) of this section was available for characterization. The portion of the section that was evaluated will herein be described as "the area of interest" and is depicted in Attachment 5.2. This characterization report captures data that has been collected and evaluated up to February 2005 in the area of interest.

## 2.2 Data Quality Objectives

The Data Quality Objectives (DQO) process described by MARSSIM is a series of planning steps found to be effective in establishing criteria for data quality and in developing survey plans associated with this characterization report. The DQO process allows for systematic planning and was particularly useful in addressing problems requiring a decision between two (2) alternatives. The DQO process includes the selection of appropriate instrumentation to meet the survey plan objective. The DQO process is flexible, in that the level of effort associated with planning a survey is based on the complexity of the survey and nature of the hazards. Finally, the DQO process is iterative, thereby providing the new knowledge as input to the design of the radiological assessment. The use of the DQO process supports the purpose of this TSD and is consistent with the approach used for characterization, remedial action and FSS plans.

## 2.3 Historical Assessment

A review of the historical files indicates a number of operational events to have impacted the area of interest. Operational events were considered to be spills and leakage from contaminated systems for the purpose of this characterization report. These events would have had the most impact on subsurface structures and footers; and underlying soil and bedrock.

The review of historical files was comprehensive and included several sources (refer to References 4.2 through 4.7). Most of the operational events were confined to the Radiologically Controlled Area (RCA). Table 2.1 provides a summary of operational events known, or suspected, to have been associated with the area of interest.

**Table 2.1 –Summary of operational events potentially impacting, or relating to, the area of interest.** Some leaks appear to originate from the same source.

| Date | Reference Document | Brief Description Of Event |
|---|---|---|
| 5/13/1976 | PIR 76-66 | Leak in Steam Generator and Test Tank Discharge line under Drumming Room floor. |
| 5/22/1976 | LER 76-13/990 | Tritium identified in leak through Safety Injection Pump wall. |
| 12/1/1976 | PIR 76-233 | Backed up floor drains contaminate floors. |
| 12/14/1976 | PIR 76-138 | Leak in Steam Generator and Test Tank Discharge line under Drumming Room floor. |
| 1/14/1977 | LER 77-1/3L | Increase in Tritium activity in Containment External Sump. |
| 7/29/1977 | PIR 77-82 | Increase in background detected above Chemistry Weir box. |
| 3/17/1978 | PIR 78-33 | Leak in Steam Generator and Test Tank Discharge line under Drumming Room floor. |
| 8/10/1979 | PIR 79-92 | Release of contaminated water from drain line. |
| 9/29/1979 | PIR 79-105 | Radioactive spill to RCA Yard. |
| 2/14/1980 | PIR 80-26 | Activity found in yard storm drain near the Drumming Room overhead door. |
| 2/28/1980 | PIR 80-34 | Soil under drumming room found shifted, wet and contaminated. |
| 3/25/1980 | LER 80-07/3L | A drain line was found to be cracked allowing water to leak into the area under the Drumming Room. |
| 4/29/1980 | PIR 80-58 | Sample of yard drain near Drumming Room found to contain activity. |
| 4/18/1983 | PIR 83-42 | Flooding of drain line to yard drain. |
| 5/25/1984 | PIR 84-72 | Over flow of drain to RCA yard. |
| 3/20/1985 | PIR 85-51 | Yard drains found to contain activity. |
| 3/25/1985 | PIR 85-52 | Drain culvert overflows to RCA yard and yard drain. |
| 5/17/1985 | PIR 85-74 | Yard drains show activity. |

PIR – Plant Information Report
LER – Licensee Event Report
ACR – Adverse Condition Report

The majority of the events refer to an area of the PAB identified as the Drumming Room. The Drumming Room was located in the southwest corner of the PAB as shown on Attachment 5.3. Discussions with CY personnel indicate the area was used to compact solid waste into fifty-five (55) gallon drums and to solidify liquid waste.

As previously discussed, review of historical documents shows the below floor area to have been affected by radioactive contaminated liquid spills and leaks. Soil was found to have been contaminated by some of these events and was removed. The following paragraphs describe in more detail the major events impacting this area.

In May 1976 a small leak of water through the wall into the safety injection cubicle indicated tritium at 8.23E-02 $\mu$Ci/ml. Several days later, another leak through the wall to the lower level of the Waste Disposal Building indicated tritium at 1.00E-01 $\mu$Ci/ml. The source of the water was determined to be a fracture in the piping junction where the Steam Generator Blowdown Tank line tied into the Service Water Effluent line.

In December 1976 the junction was again found to be leaking, but in a different location. Records show that sand was collected underneath the leak and an isotopic analysis was performed. Historical records report Cs-137 at 7.44E-05 $\mu$Ci/ml, Co-58 at 2.23E-05 $\mu$Ci/ml and Co-60 at 1.16E-04 $\mu$Ci/ml. These concentrations are at least and order of magnitude above our current site release criteria specified by the LTP. The record also reports that the soil in the immediate area was removed.

In March 1978 a visual observation of the area during a routine surveillance identified a leak at the junction. Historical records indicate the intent to relocate the junction downstream from the original point. An analysis was performed at the time to ensure structural integrity of the line after rerouting.

In 1980 visual inspection found sand in the vicinity of the junction to have been shifted. The soil was found to have been contaminated from a broken drain. The drain was not intended for contaminated liquids, but was being used since the normal floor drains in the Drumming Room were plugged. The record reported long lived fission and corrosion products where activity ranged from 3E-03 $\mu$Ci/ml to 6E-03 $\mu$Ci/ml. The sand was removed according to the historical records.

## 2.4 Initial Characterization

Characterization of this area was performed in 1998 as part of an augmented study to support the selection of a Decommissioning Operations Contractor (DOC). According to the report, five (5) samples were collected from a single survey measurement location accessed through a small opening in the floor of the drumming room. The accessible area was a tight crawl space containing process piping of different systems. This was the location identified as having the historical leaks. The dirt surface was about 2 or 3 feet from the bottom of the drumming room floor at that access opening according to the report. Due to elevated background in this area survey measurements were not performed. The soil samples were collected at various depths. Table 2.2 provides the results for this characterization. No analysis results for Hard-To-Detect (HTD) radionuclides were provided in this report for this area (Am-241 is not considered HTD, refer to LTP Table 2-12).

**Table 2.2 – Augmented characterization results from the soil below the Drumming Room.** Concentrations appear to decrease as expected with increased depth.

| | Radionuclide Concentration[2] | | | |
|---|---|---|---|---|
| Depth[1] | Cs-134 | Cs-137 | Co-60 | Am-241 |
| 0-6" | -----[3] | 126.6 | 8.73 | 0.29 |
| 6-12" | 0.16 | 49.1 | 3.57 | 0.18 |
| 12-24" | ----- | ----- | ----- | ----- |
| 24-36" | ----- | 11.15 | 2.84 | 0.18 |
| 36-48" | ----- | 4.6 | 0.46 | ----- |

[1] Depth reported as bgs
[2] All concentration results in pCi/g
[3] Indicates no information available

Survey and sampling was conducted January 2004 to obtain characterization data to determine how much remediation may be required to reduce contamination levels to allow open air building demolition. The Cable Vault was determined to be a SSC that needed to be assessed to support this objective. The survey and sampling plan included sample locations inside the Cable Vault along the outer wall (Reference 4.8). The locations for the samples were specified by the FSS Engineer using observation and process knowledge as the basis for professional judgment. Locations were in or near the saturated zone (seasonal dependent).

Mobile direct-rotary drilling was used to drill through the concrete. The direct-rotary drilling system consists of mechanical components and water, free of radiological contamination, to cool the drill bit and flush the cuttings. Direct-rotary drilling has advantages, for example, it is faster than some other methods and collected samples are as representative as those collected using other techniques. Direct-rotary drill is especially useful in obtaining sample media from consolidated materials, such as concrete. The survey and sampling plan specified two (2) core locations. Direct-rotary drilling was to continue until soil was obtained. The total core lengths ranged from 24" to 36" depending on location.

Cores were cut into 1" to 2" wafers on-site using a standard brick saw with a diamond tipped blade. Selected wafers were given special sample nomenclature and were designated for off-site analysis. The FSS Engineer selected wafers throughout the core for analysis.

A total of seventeen (17) samples were processed off-site using approved procedures and methodologies. The sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations ($2\sigma$), data qualifiers, and the required and observed Minimum Detectable Concentration (MDC).

The maximum reported concentration for Sr-90 was 2.06E-01 pCi/g. The maximum reported tritium concentration was 7.04E+00 pCi/g. The data show Sr-90 to be present on the Cable Vault Outer wall (side of the structure in the soil and in contact with groundwater) and tritium to be diffused volumetrically to some extent throughout the concrete.

## 2.5 Demolition and Remediation Activities

Demolition and remediation activities began with the erection of a large tent on the east end of the excavation for engineering controls over the tank farm area. The tent was completed and in service in August 2002. Removal of contaminated materials was performed to ensure contamination levels were below levels acceptable for controlled demolition. Confirmatory radiological surveys were performed using approved procedures. Tank Farm demolition began in November 2002. Removal of contaminated materials was performed in the HP Facility, PAB and WDB to ensure contamination levels were below levels acceptable for controlled demolition. Confirmatory radiological surveys were performed using approved procedures. No tent was required as contamination levels were low enough to support open air demolition. Demolition of the HP Facility and the WDB began August 2004. Demolition of the PAB began September 2004.

Soil remediation began in the area under the former Drumming Room during the demolition of the PAB. Survey and sampling was conducted during the period of November 10, 2004 through November 20, 2004. All characterization was performed under a survey and sampling plan using approved procedures (References 4.9 through 4.17). Instrumentation used in the field was in calibration and in good working order. Detailed field notes and observations were documented during survey and sampling.

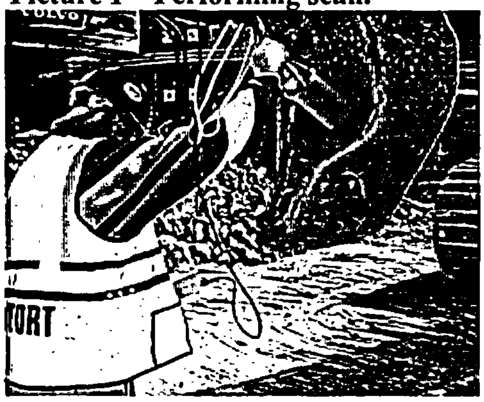
Soil samples were collected from the excavator bucket for safety purposes given the physical condition of the excavation and the potential for a hazardous confined space. The size of the excavation was small at that time, and did not permit personnel entry. The locations for the samples were defined by the sampling plan and at other locations as specified by the field FSS Engineer using observation and process knowledge as the basis for professional judgment.

The excavator bucket was surveyed prior to soil sampling using gamma sensitive equipment during each sample collection. Scanning was performed over 100% of the available soil using an E-600 digital survey meter with a SPA-3 2" x 2" sodium iodide detector. The instrument was operated in the rate meter mode with audio output enabled. This choice was reasonable given the expected radionuclides as shown by Table 2.2. The scanning was to be used as a screening mechanism to identify those samples having the potential for radioactivity above the Operational DCGL specified by the survey and sampling plan. A review of the field notes indicates several elevated readings were identified.

Soils were collected from the center of the excavator bucket. Following collection, the soil was homogenized and split into two (2) separate samples. The purpose of the separate samples was to have the ability to analyze samples at on-site and off-site laboratories. A concrete chunk was collected from the Service Building wall at a location determined by the field FSS Engineer. The basis for the decision was that the FSS Engineer wanted a location in the saturated zone near the area of leakage in the former Drumming Room. Samples were maintained under Chain-of-Custody (COC) until turned over to the appropriate laboratory for analysis.

**Characterization Report for the West Section of the Excavation Associated with the Northeast Protected Area Grounds**

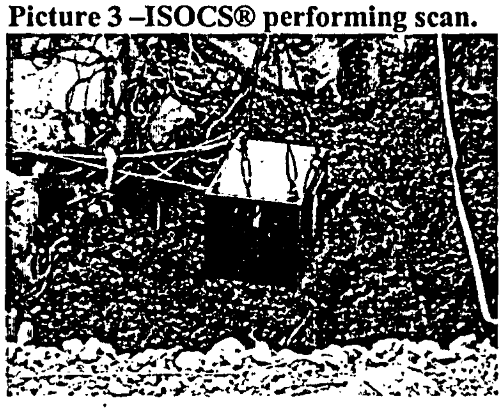
**Picture 1 – Performing scan.**



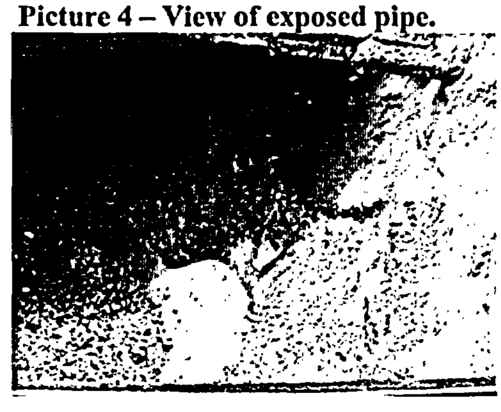
**Picture 2 – Sample collection.**

Gamma scans of the excavation area were conducted using the Canberra In-Situ Object Counting System (ISOCS®). The detector was housed in a custom made tube steel frame to support and protect the detector and electronics. Swivel hoist rings at each corner allowed the ISOCS® instrument cart to be rigged and positioned over the desired locations. Measurements taken inside the excavation identified one area of elevated activity. Photos obtained during ISOCS® evaluation identified a pipe extending from the soil (refer to Picture 3 and Picture 4). This pipe was removed and surveyed by HP. Additional description of the ISOCS® system and capabilities is provided in a later section.

**Picture 3 –ISOCS® performing scan.**



**Picture 4 – View of exposed pipe.**



Samples were processed at on-site and off-site laboratories using approved procedures and methodologies. All samples were analyzed using gamma spectroscopy to the specified sensitivities required by the survey and sampling plan. Calculated results were reported instead of <MDC. On-site sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations ($2\sigma$). The off-site laboratory report provided the same information as well as data qualifiers, and the required and observed MDC.

On-site sampling results identified Cs-137 and Co-60 in the soil above the Operational DCGL. A review of the off-site laboratory gamma spectroscopy results showed good agreement between on-site and off-site analyses results from the same sample location. The off-site laboratory reported tritium and Sr-90 at concentrations below the Operational DCGL for soil. However, the off-site laboratory did report tritium results above the Operational DCGL for concrete debris in the concrete chunk based on the current approach described by revision 2 of the LTP. A methodology to assess radiological conditions to meet site release criteria in cases like this is described in Section 2.7.

The Planning Team decided to halt further sampling and spot remediation based on in-situ, soil and concrete analyses results and the presence of potentially contaminated components in the area of concern. The recommendation was made to sample the area in the future following additional soil and structure removal. Initial characterization was therefore complete, and remediation and demolition continued in the area of interest. The scope of remediation and demolition was to remove the PAB structure down to bedrock and remove most of the soil from the area. Characterization would continue once the excavation was in a condition to properly evaluate conditions.

The initial data, although not conclusive, indicated that future characterization of remaining soils would be evaluated for gamma emitters, primarily Cs-137 and Co-60, and Sr-90. Remaining subsurface structures and footings surfaces would be evaluated for tritium and Sr-90. Exposed bedrock would be evaluated for tritium and Sr-90. At least 5% of the characterization samples would be analyzed for HTD radionuclides.

## 2.6 Excavation Condition at the Time of Characterization

The demolition contractor continued demolition and removal of soils and debris in the area of interest during the month of December 2004. In early January 2005 the excavation was in a physical condition suitable for performing characterization. Weather was acceptable to safely access the area and to perform survey and sampling. Temperatures at the time did not appear to adversely affect equipment. Detailed safety and work briefings were performed daily. Workers in the area were made aware of the physical conditions in the area, the work to be performed, the expectations with regards to changing conditions, the caution of working near moving equipment, the required personnel protective equipment (e.g., high visibility vests, hard hats, safety shoes, etc.), and the person in charge at the work site. Daily inspections were performed to evaluate hazardous conditions and atmospheres. Fall protection was not required for survey and sampling. Access to the excavation was from the south down a ramp sloped to 1 ½ to 1 (slope trench back 1 ½ foot for every 1 foot of depth).

The remaining subsurface structures and footing include the Service Building east wall, a remnant of the PAB northwest wall, Cable Vault Footing Wall, Cable Vault wall and miscellaneous footings and slabs on the bedrock. Bedrock was pretty much exposed in the area. The intrusion of groundwater was controlled by suppressing the water table using a series of pumps. Standing water was limited to the northwest corner of the excavation. Some residual soil remained near the eastern boundary (i.e., the boundary between the area of interest and the remaining western excavation) and in the northwest corner. This characterization report will include pictures of the excavation as attachments to subsequent sections.

## 2.7 Characterization Survey and Sampling Planning

The primary objective of the characterization survey and sampling plan design was to demonstrate that the level of residual radioactivity in the area of interest did not exceed the release criteria specified in the LTP and was below administrative levels established to ensure compliance with criteria agreed to with the State of Connecticut Department of Environmental Protection (CT DEP). The LTP criteria apply at the time of license termination or removal of the area from the license. The LTP criteria include a maximum dose to the average member of the selected critical group of 25 mrem in a year Total Effective Dose Equivalent (TEDE) from all pathways. The CTDEP criteria apply at the transfer of property and include a commitment to a remediation standard of 19 mrem in a year TEDE from all pathways as well as achieving the Maximum Contaminated Levels (MCL) as established by the United States Environmental Protection Agency (USEPA).

Two (2) documents have been generated by Connecticut Yankee (CY) to provide direction and basis for meeting these criteria. The first document is CY memo ISC 04-049, Initial Target Operational DCGLs for CY (Reference 4.18). This memo provides target values to be used for designing survey and sampling plans. These target values or Operational DCGLs are based on the exposure from three (3) potential media. The Operational DCGLs are typically below the established criteria described above. These media consist of residual activity in soil, existing groundwater (GW) radioactivity and future GW radioactivity from subsurface structures and concrete debris. The survey and planning design for the area of interest associated with this excavation considered all three (3) media when determining compliance with the criteria previously described. In addition to the above criteria, the survey and sampling design needed to address bedrock radiological assessment.

Bedrock radiological assessment is described by the second CY document, HP CY-HP-0190, Technical Basis for Performing Radiological Assessments of Bedrock, Soil and Subsurface Concrete Using an Inventory-Based Approach (Reference 4.19). The LTP requires that a radiological assessment of bedrock areas be performed prior to backfill.

The TSD (CY-HP-0190) describes a process using an approach which will use radioactivity inventories when calculating future groundwater dose. The TSD methodology explains the post-remediation field assessment and dose assessment described in the LTP (refer to LTP Section 5.7.3.2.5) and is similar to that proposed in a license amendment request to modify the current "Buried Concrete Debris Model" to a "Basement Fill Model". The Basement Fill Model, applied to the area of interest, uses the quantity of radioactivity released to the saturated zone (i.e., the volume below the water table) to calculate the resulting GW concentration. A more conservative assumption is made that radiological release is not controlled by diffusion over time; rather, radiological release occurs instantly. The resulting GW contamination is determined from the equilibrium between GW and the backfill soil. The equilibrium GW concentration associated with the saturated zone is based on the principles of linear sorption theory and is expressed by Equation 6.1 in the LTP. The process is relatively straightforward and may be explained as follows:

- An assessment of the media (e.g., remaining subsurface structures and footings, bedrock and residual soil) will determine the radionuclides and their respective average concentration (i.e., surface in dpm/100 $cm^2$ or volumetric pCi/g).
- The surface area or volume of media.
- The total inventory of radioactivity to be released to the saturated zone.
- Calculating the equilibrium concentration between GW and the backfill soil using the results of a Brookhaven National Laboratory distribution coefficient ($K_d$) study of backfill (Reference 4.20).
- Comparing the GW concentration to the MCL.
- Comparing the GW concentration to the DCGL.

The TSD (CY-HP-0190) also provides a method of determining Operational DCGLs for remediation guidelines. It essentially is the above process in reverse. Assume a target GW dose is established (e.g., the TSD assumes ½ of the MCL). Application of the process allows the survey and sampling plan designer to calculate target values for the surface and volumetric contamination that may remain. Once this is known, instrumentation selection and determination of analytic techniques may be established. Both methods will be used for the future radiological assessment. That is, data collected during this characterization are found acceptable and are then evaluated to determine the future GW dose or, if more data is needed, a target value will be determined to ensure data is acceptable (or more remediation is required).

Both documents are included in this characterization report as attachments (refer to Attachment 5.4 and 5.5).

## 2.8 Characterization Synopsis

Survey and sampling was conducted on October 27, 2004 and again during the period of January 10, 2005 through January 13, 2005. Additional bedrock samples were collected between January 31, 2005 and February 2 2005. Additional soil samples were collected on February 8, 2005. All characterization was performed under a survey and sampling plan using approved procedures. Three (3) survey and sampling plans were required given the scope and complexity of the characterization and the different sampling involved (References 4.21 through 4.23). Weather and field conditions were acceptable, as previously discussed. Survey and sampling was performed by qualified senior health physics technicians under the direction of a FSS Engineer. Instrumentation used in the field was in calibration and in good working order. Detailed field notes and observations were documented during survey and sampling.

Survey and sampling consisted of radiation scans of available subsurface structures and footings and exposed bedrock and soils, soil collection, cores of concrete in the Service Building, PAB remnant wall, Cable Vault Footing Wall, cores of bedrock, and in-situ gamma measurements using ISOCS®. These are discussed in the following sections.

## 2.9 Radiation Scans

Radiation scans were performed to determine areas of elevated activity warranting additional evaluation. Scanning was performed using gamma and beta radiation sensitive portable instrumentation. The use of two (2) types of instrumentation was determined during the DQO process based on initial characterization results.

Scanning was performed over 100% of the available bedrock, soil and exposed concrete footings and slabs using an E-600 digital survey meter with a SPA-3 2" x 2" sodium iodide detector. The instrument was operated in the rate meter mode with audio output enabled. This instrument was determined suitable for the type and energy of radiation expected (i.e., Cs-137 and Co-60). Prior to the scan survey ambient background was determined and the criterion for an elevated area established. An elevated reading was defined as twice ambient background for the scan survey. Ambient radiation background was low enough to achieve a Minimum Detectable Activity (MDA) of about 5.8 pCi/g for Cs-137 and 1.5 pCi/g for Co-60 at a scan rate of twenty (20) inches a second. One (1) area meeting the elevated criterion was identified on soil during the survey. Two (2) small pieces of metal were found and were removed from the area. Subsequent survey showed the metal to be radioactive (maximum dose rate was 0.5 mR/hr). A sample was collected from this area.

A portable gamma spectroscopy device, the Exploranium GR-130 MINISPEC, was used to perform a qualitative assessment of the soil sample from the elevated area. The Exploranium was stabilized prior to use using a small Cs-137 check source. A ten (10) minute count was started with the Exploranium in direct contact with the soil sample package (plastic bag). The Exploranium identified Cs-137 and Co-60 in the sample. The analytical results of the soil sample will be discussed in a subsequent section.

Another area meeting the elevated criterion was identified on bedrock. A ten (10) minute count was started with the Exploranium in direct contact with the bedrock. The exploranium identified progeny associated with the decay of naturally occurring radionuclides on the bedrock. No sample was collected.

Additional scanning was performed over 100% of the available bedrock, soil and exposed concrete using a NE Electra portable rate meter with a 100 $cm^2$ dual scintillation detector. The Electra was operated in the $\beta$ mode with the audio output enabled. This instrument was determined suitable for the type and energy of radiation expected (i.e., Sr-90). Tritium was considered to be uniformly distributed making areas of elevated activity unlikely. Prior to the scan survey ambient background was determined and the criterion for an elevated area established. An elevated reading was defined as twice ambient background for the scan survey. Ambient radiation background was low enough to achieve an MDA of about 1000 dpm for Cs-137 (which would mean a lower MDA for Sr-90 based on the higher expected efficiency of the instrument to the SrY-90 betas) at a scan rate of two (2) inches a second. One (1) area meeting the elevated criterion was identified on the bedrock.

Another area meeting the elevated criterion was identified on bedrock. Visual examination of the bedrock identified a thin vein appearing to be quartz in the elevated area. A ten (10) minute count was started with the Exploranium in direct contact with the bedrock. The Exploranium did not identify any radionuclides as being present. However, field evaluation of the reported gamma energy peaks by the FSS Engineer indicated progeny (i.e., Pb-214, Bi-214) associated with the decay of naturally occurring radionuclides. No sample was collected. The field evaluation was later confirmed by the FSS engineer using standard technical reference (Reference 4.24).

Additionally, scanning was performed over 100% of the available Service Building, PAB remnant, Cable Vault Footing and Cable Vault Outer wall surfaces up to a height of two (2) meters using the NE Electra portable rate meter with a 100 $cm^2$ dual scintillation detector. Scanning was performed from bedrock level. This provided scan coverage of roughly two-thirds ($^2/_3$) of the area of concrete in the saturated zone. The mode of operation, applicability of use, establishment of elevated criterion and method of use were the same as previously discussed. No areas meeting the elevated criterion were identified.

## 2.10 Characterization of Soil

The survey and sampling plans specified soil collection along the bottom and sides of the excavation. There was a relatively shallow layer of soil covering the bedrock in the excavation as previously discussed. Most of the soil was contained on the sides of the excavation. Radionuclide specific analyses of soil were performed to determine whether soil concentrations meet the Operational DCGLs. The Operational DCGLs for soil were radionuclide specific and were set to achieve a dose limit of eight (8) mrem in a year TEDE. These Operational DCGLs were well below the LTP and CTDEP criterion previously specified. The Operational DCGLs were used to determine radionuclide specific MDCs to ensure adequate sensitivity during analysis.

The locations for the samples were defined by the survey and sampling plan and at other locations as specified by the field FSS Engineer using observation and process knowledge as the basis for professional judgment. In one (1) case, the scanning was the screening mechanism to identify a sample having the potential for radioactivity above the Operational DCGL specified by the survey and sampling plan (refer to Section 2.9).

A total of thirty-one samples (31) were collected from the interior or base of the excavation and sides of the excavation. The sample results were treated separately for two reasons. First reason, the TSD (CY-HP-0190) does not include soil that makes up the side walls of the excavation. Second reason, the analytic results are different for the three (3) areas. It was decided that separating the three (3) areas would facilitate the interpretation of the data.

The soil samples were maintained under COC until turned over to the off-site laboratory for analysis. The location of the soil samples is denoted on Attachment 5.6.

2.10.1 Base of Excavation

Seventeen (17) of the thirty-one (31) samples were collected from the interior or base of the excavation. All samples were analyzed using gamma spectroscopy to the specified sensitivities required by the survey and sampling plans. All samples were analyzed for tritium and Sr-90 to the specified sensitivities required by the survey and sampling plan. Two (2) samples, including the one associated with the elevated scan measurement, were analyzed for all the radionuclides listed in Table 2-12 of the LTP. Those radionuclides not specified by the survey and sampling plan were analyzed to the specific sensitivities specified by the Statement of Work (SOW) (Reference 4.25). Calculated results were reported instead of <MDC. The sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations ($2\sigma$), data qualifiers, and the required and observed MDC.

The off-site laboratory certificate of analysis reported Co-60 in concentrations above the Operational DCGL in one (1) of the seventeen (17) analyzed samples. Cesium-137 was reported in concentrations above the Operational DCGL in four (4) of the seventeen (17) analyzed samples. Strontium-90 was reported in concentrations that met the accepted criterion for detection (i.e., a result greater than two (2) standard deviations uncertainty) in ten (10) of the seventeen (17) analyzed samples. Tritium was reported in concentrations that met the accepted criterion for detection in one (1) of the seventeen (17) analyzed samples. All reported results for Sr-90 and Tritium were below the Operational DCGLs. Most of the radionuclides listed in Table 2-12 of the LTP were identified with reported concentrations that met the accepted criterion for detection in one (1) of the seventeen (17) analyzed samples (this was the sample associated with the elevated scan). Some of these were above the Operational

DCGLs. Table 2.3 summarizes the currently available data for the principle radionuclides associated with soil in the excavation.

**Table 2.3 – Basic statistical quantities for Cs-137 and Co-60 from the population data set associated with the excavation interior.** The data indicate further remediation is necessary in the excavation.

| Quantity: | Cs-137[1] | Co-60[1] |
|---|---|---|
| Operational DCGL: | 2.53E+00 | 1.220E+00 |
| Minimum Value: | -1.68E-02 | -2.66E-02 |
| Maximum Value: | 5.63E+01 | 1.32E+01 |
| Mean: | 4.22E+00 | 8.85E-01 |
| Median: | 9.54E-02 | 4.09E-02 |
| Standard Deviation: | 1.35E+01 | 3.18E+00 |

[1] All concentration results in pCi/g

### 2.10.2 Northwest Side of the Excavation

Eight (8) of the thirty-one (31) samples were collected from the northwest side of the excavation. All samples were analyzed using gamma spectroscopy to the specified sensitivities required by the survey and sampling plan. All samples were analyzed for tritium and Sr-90 to the specified sensitivities required by the survey and sampling plan. One (1) sample was analyzed for all the radionuclides listed in Table 2-12 of the LTP. Those radionuclides not specified by the survey and sampling plan were analyzed to the specific sensitivities specified by the SOW. Calculated results were reported instead of <MDC. The sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations ($2\sigma$), data qualifiers, and the required and observed MDC.

The off-site laboratory certificate of analysis reported Cs-137 in concentrations above the Operational DCGL in all seven (7) of the eight (8) analyzed samples. Cobalt-60 was reported in concentrations that met the accepted criterion for detection (i.e., a result greater than two (2) standard deviations uncertainty) in all eight (8) analyzed samples. All reported results for Co-60 and Cs-137 were below the Operational DCGLs. Strontium-90 was reported in concentrations that met the accepted criterion for detection in all eight (8) analyzed samples. Tritium was not identified in any of the analyzed samples. A few of the radionuclides listed in Table 2-12 of the LTP were identified with reported concentrations that met the accepted criterion for detection. All were below the Operational DCGLs. Table 2.4 summarizes the currently available data for the principle radionuclides associated with soil in the excavation.

**Table 2.4 – Basic statistical quantities for Cs-137 and Co-60 from the population data set associated with the north side of the excavation.** The data indicate further remediation is necessary along the north side of the excavation.

| Quantity: | Cs-137[1] | Co-60[1] |
|---|---|---|
| Operational DCGL: | 2.53E+00 | 1.220E+00 |
| Minimum Value: | 1.89E+00 | 2.09E-01 |
| Maximum Value: | 6.83E+00 | 4.03E-01 |
| Mean: | 4.08E+00 | 3.03E-01 |
| Median: | 3.60E+00 | 3.09E-01 |
| Standard Deviation: | 1.59E+00 | 7.56E-02 |

[1] All concentration results in pCi/g

2.10.3 Southeast Side of the Excavation

Six (6) of the thirty-one samples (31) were collected from the southeast side of the excavation. All samples were processed off-site using approved procedures and methodologies. All samples were analyzed using gamma spectroscopy to the specified sensitivities required by the survey and sampling plan. All samples were analyzed for tritium and Sr-90 to the specified sensitivities required by the survey and sampling plan. Calculated results were reported instead of <MDC. The sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations ($2\sigma$), data qualifiers, and the required and observed MDC.

The off-site laboratory certificate of analysis did not report Cs-137 or Co-60 in concentrations that met the accepted criterion for detection (i.e., a result greater than two (2) standard deviations uncertainty) in any of the analyzed samples. Strontium-90 and tritium were not identified in any of the analyzed samples. A few of the radionuclides listed in Table 2-12 of the LTP were identified with reported concentrations that met the accepted criterion for detection. All were below the Operational DCGLs. Table 2.5 summarizes the currently available data for the principle radionuclides associated with soil in the excavation.

**Table 2.5 – Basic statistical quantities for Cs-137 and Co-60 from the population data set associated with the south side of the excavation.** The data indicate no further remediation is necessary along the south side of the excavation.

| Quantity: | Cs-137[1] | Co-60[1] |
|---|---|---|
| Operational DCGL: | 2.53E+00 | 1.220E+00 |
| Minimum Value: | -1.30E-03 | -2.73E-03 |
| Maximum Value: | 2.38E-02 | 1.29E-02 |
| Mean: | 4.29E-03 | 4.63E-03 |
| Median: | 7.10E-04 | 3.46E-03 |
| Standard Deviation: | 9.70E-03 | 7.41E-03 |

[1] All concentration results in pCi/g

The data indicate that the remaining soil in the excavation and along the sides would contain Cs-137 and Co-60, and perhaps some Sr-90. The finding of elevated radioactivity in the northwest corner has identified the need for further remediation and confirmatory sampling in this area. Demolition work and spoils removal is still in-progress in the northeast corner of the PAB footprint and the PAB. Additional sampling will be necessary in these areas to fully characterize the excavation prior to the radiological assessment planned for March 2005.

## 2.11 Characterization of Service Building Wall Cores

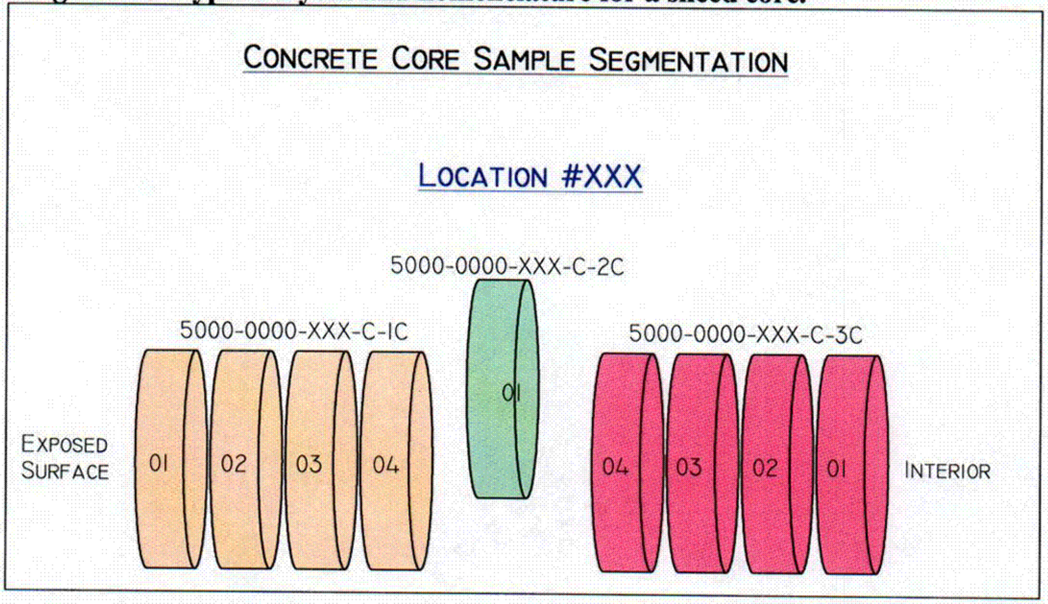
The survey and sampling plans specified concrete core collection through the Service Building wall. Radionuclide specific analyses of concrete cores were performed to determine whether concrete concentrations met the Operational DCGLs. The Operational DCGLs for concrete were radionuclide specific and were set to achieve a dose limit of two (2) mrem in a year TEDE for tritium and ten (10) mrem in a year TEDE for Sr-90. The difference in the dose limits for the Operational DCGLs was based on the potential impact to future groundwater. These Operational DCGLs were well below the LTP and CTDEP criterion previously specified. The Operational DCGLs were used to determine radionuclide specific MDCs to ensure adequate sensitivity during analysis.

The locations for the cores were specified by the FSS Engineer using observation and process knowledge as the basis for professional judgment. Locations were in the saturated zone, and all but one (1), were collected about three (3) feet above bedrock level. Cores were the appropriate choice for characterization of the walls given the potential for tritium and Sr-90 diffusion. The cores would then be cut in wafers, analyzed as necessary to determine the radionuclide concentration and depth of contamination.

Mobile direct-rotary drilling was used to drill through the concrete. The survey and sampling plans specified seven (7) core locations. Direct-rotary drilling was to continue until soil was obtained, when possible. This could not be done at one (1) location (location 106, refer to Attachment 5.6). The total core lengths ranged from 24" to 60" depending on location.

Cores were cut into 1" to 2" wafers on-site using a standard brick saw with a diamond tipped blade. Selected wafers were given special sample nomenclature and were designated for off-site analysis. Experience with core analysis from previous characterization campaigns has shown that only a few wafers are necessary to evaluate the quantity and extent of contamination. The FSS Engineer selected a few wafers from the end of the core that was collected first (i.e., the top 0" to 10" from the exposed surface of the wall), a wafer from the middle of the core, and a few wafers from the end of the last core collected. Diagram 1 depicts an exploded view of a typical core after cutting.

**Diagram 1 – Typical layout and nomenclature for a sliced core.**




The wafers were maintained under COC until turned over to the off-site laboratory for analysis. The location of the core samples is denoted on Attachment 5.6.

A total of forty-three (43) wafer samples from seven (7) core locations were processed off-site using approved procedures and methodologies. All samples were analyzed using gamma spectroscopy to the specified sensitivities required by the survey and sampling plan. All samples were analyzed for tritium and Sr-90 to the specified sensitivities required by the survey and sampling plan. The first wafer (i.e., the closest



C01

to the exposed surface) from five (5) of the seven(7) core locations was analyzed for all the radionuclides listed in the LTP. Those radionuclides not specified by the survey and sampling plan were analyzed to the specific sensitivities specified by the SOW. Calculated results were reported instead of <MDC. The sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations (2σ), data qualifiers, and the required and observed MDC.

The off-site laboratory certificate of analysis reported Sr-90 in concentrations above the Operational DCGL in two (2) of forty-three (43) analyzed samples. Tritium was reported in concentrations above the Operational DCGL in twenty-two (22) of the forty-three (43) analyzed samples. Cobalt-60 was reported in concentrations that met the accepted criterion for detection (i.e., a result greater than two (2) standard deviations uncertainty) in three (3) of the forty-three (43) analyzed samples. Cesium-137 was reported in concentrations that met the accepted criterion for detection in one (1) of the forty-three (43) analyzed samples. All reported results for Co-60 and Cs-137 were below the Operational DCGLs. A few of the radionuclides listed in Table 2-12 of the LTP were identified with reported concentrations that met the accepted criterion for detection. All were below the Operational DCGLs. Tables 2.6 through 2.8 summarize the data.

**Table 2.6 – Basic statistical quantities for Sr-90 and tritium from the population data set associated with the exposed surface of the Service Building wall (the side facing the excavation).** Strontium-90 and tritium were both over the Operational DCGL.

| Quantity: | Sr-90[1] | Tritium[1] |
|---|---|---|
| Operational DCGL: | 1.51E-01 | 7.86E+00 |
| Minimum Value: | -1.40E-02 | 1.05E+00 |
| Maximum Value: | 2.66E-01 | 5.96E+01 |
| Mean: | 3.98E-02 | 2.33E+01 |
| Median: | 2.74E-02 | 2.00E+01 |
| Standard Deviation: | 5.84E-02 | 2.15E+01 |

[1] All concentration results in pCi/g

**Table 2.7 – Basic statistical quantities for Sr-90 and tritium from the population data set associated with the middle cores of the Service Building wall.** Tritium is over the Operational DCGL. The large difference between tritium mean and median indicate skewness in the data, which is conservative in this case.

| Quantity: | Sr-90[1] | Tritium[1] |
|---|---|---|
| Operational DCGL: | 1.51E-01 | 7.86E+00 |
| Minimum Value: | -1.61E-02 | -1.58E+00 |
| Maximum Value: | 5.50E-02 | 6.38E+00 |
| Mean: | 1.67E-02 | 1.44E+00 |
| Median: | 1.50E-02 | 1.40E-01 |
| Standard Deviation: | 2.58E-02 | 2.73E+00 |

[1] All concentration results in pCi/g

**Table 2.8 – Basic statistical quantities for Sr-90 and tritium from the population data set associated with the internal cores of the Service Building wall (the deepest cores).** Strontium-90 and tritium were both over the Operational DCGL.

| Quantity: | Sr-90[1] | Tritium[1] |
|---|---|---|
| Operational DCGL: | 1.51E-01 | 7.86E+00 |
| Minimum Value: | -3.52E-02 | -4.52E-01 |
| Maximum Value: | 3.60E-01 | 3.96E+01 |
| Mean: | 2.46E-02 | 1.34E+01 |
| Median: | 2.07E-03 | 8.84E+00 |
| Standard Deviation: | 9.79E-02 | 1.37E+01 |

[1] All concentration results in pCi/g

The average concentration for Sr-90 and tritium from each population data set is presented in Table 2.9. Table 2.9 takes these values from the mean values in Tables 2.6 through 2.8.

**Table 2.9 – Characterization results as a function of core location.**

| Core Population Set | Number of Wafers Analyzed | Sr-90[1] | Tritium[1] |
|---|---|---|---|
| Exposed Surface: | 22 | 3.98E-02 | 2.33E+01 |
| Middle: | 7 | 1.67E-02 | 1.44E+00 |
| Internal: | 14 | 2.46E-02 | 1.34E+01 |

[1] All concentration results in pCi/g

The data indicates that the Service Building wall will contain Sr-90 and tritium diffused volumetrically to some extent. The data support the assumption of residual contamination on either side of the Service Building wall that tapers off with increased depth into the concrete. Although the reported concentrations were above the Operational DCGLs for the characterization survey and sampling plan, the concentrations and the amount of total inventory available to be released are low enough to project meeting the acceptance criteria of the radiological assessment using the methodology described by Reference 4.19. A sufficient number of cores have been collected and analyzed to characterize the Service Building wall for radiological assessment.

## 2.12 Characterization of the PAB Wall Core

The survey and sampling plans specified concrete core collection through the remnant wall of the PAB. One (1) location was considered acceptable given the small area of the remaining PAB. Radionuclide specific analyses of concrete cores were performed to determine whether concrete concentrations met the Operational DCGLs. The Operational DCGLs for concrete were radionuclide specific and were set to achieve a dose limit of two (2) mrem in a year TEDE for tritium and ten (10) mrem in a year TEDE for Sr-90. The difference in the dose limits for the Operational DCGLs was based on the potential impact to future groundwater. These Operational DCGLs were well below the LTP and CTDEP criterion previously specified. The Operational DCGLs were used to determine radionuclide specific MDCs to ensure adequate sensitivity during analysis.

The location for the core was specified by the FSS Engineer using observation and process knowledge as the basis for professional judgment. The location was in the saturated zone about three (3) feet above bedrock level. Cores were the appropriate choice for characterization of the wall given the potential for tritium and Sr-90 diffusion. The cores would then be cut in wafers, analyzed as necessary to determine the radionuclide concentration and depth of contamination.

Mobile direct-rotary drilling was used to drill through the concrete. The survey and sampling plans specified one (1) core location. Direct-rotary drilling was to continue until soil was obtained, when possible. Total core length was approximately 24" at this location.

Cores were cut into 1" to 2" wafers on-site using a standard brick saw with a diamond tipped blade. Selected wafers were given special sample nomenclature and were designated for off-site analysis. The FSS Engineer selected a few wafers from the end of the core that was collected first (i.e., the top 0" to 4" from the exposed surface of the wall), a wafer from the middle of the core, and a few wafers from the end of the last core collected.

The wafers were maintained under COC until turned over to the off-site laboratory for analysis. The location of the core samples is denoted on Attachment 5.6.

A total of nine (9) wafer samples from the one (1) core location were processed off-site using approved procedures and methodologies. All samples were analyzed using gamma spectroscopy to the specified sensitivities required by the survey and sampling plan. All samples were analyzed for tritium and Sr-90 to the specified sensitivities required by the survey and sampling plan. The first wafer (i.e., the closest to the exposed surface) was analyzed for all the radionuclides listed in the LTP. Those radionuclides not specified by the survey and sampling plan were analyzed to the specific sensitivities specified by the SOW. Calculated results were reported instead of <MDC. The sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations ($2\sigma$), data qualifiers, and the required and observed MDC.

The off-site laboratory certificate of analysis reported tritium in concentrations above the Operational DCGL in three of the nine analyzed samples. Strontium-90 was reported in concentrations that met the accepted criterion for detection (i.e., a result greater than two (2) standard deviations uncertainty) in one (1) of the nine (9) analyzed samples. Cobalt-60 was reported in concentrations that met the accepted criterion for detection in three (3) of the nine (9) analyzed samples. Cesium-137 was reported in concentrations that met the accepted criterion for detection in two (2) of the (9) nine analyzed samples. All reported results for Sr-90, Co-60 and Sr-90 were below the Operational DCGLs. A few of the radionuclides listed in Table 2-12 of the LTP were identified with reported concentrations that met the accepted criterion for detection. All were below the Operational DCGLs. Tables 2.10 through 2.12 summarize the data.

**Table 2.10 – Basic statistical quantities for Sr-90 and tritium from the population data set associated with the exposed surface of the PAB wall remnant (the side facing the excavation).** Tritium was over the Operational DCGL.

| Quantity: | Sr-90[1] | Tritium[1] |
|---|---|---|
| Operational DCGL: | 1.51E-01 | 7.86E+00 |
| Minimum Value: | -1.99E-03 | 1.76E+00 |
| Maximum Value: | 4.81E-03 | 1.24E+01 |
| Mean: | 1.08E-03 | 8.85E+00 |
| Median: | 7.49E-04 | 1.06E+01 |
| Standard Deviation: | 2.99E-03 | 4.88E+00 |

[1] All concentration results in pCi/g

**Table 2.11 – Reported quantities for Sr-90 and tritium from the single sample associated with the middle core of the PAB wall remnant.** Tritium was below the Operational DCGL.

| Quantity: | Sr-90[1] | Tritium[1] |
|---|---|---|
| Operational DCGL: | 1.51E-01 | 7.86E+00 |
| Reported Value: | -2.45E-03 | 4.36E+00 |

[1] All concentration results in pCi/g

**Table 2.12 – Basic statistical quantities for Sr-90 and tritium from the population data set associated with the internal cores of the PAB wall (the deepest core).** Tritium was below the Operational DCGL.

| Quantity: | Sr-90[1] | Tritium[1] |
|---|---|---|
| Operational DCGL: | 1.51E-01 | 7.86E+00 |
| Minimum Value: | -2.45E-03 | 1.83E+00 |
| Maximum Value: | 2.69E-03 | 5.08E+00 |
| Mean: | 1.30E-04 | 2.69E+00 |
| Median: | 1.40E-04 | 1.92E+00 |
| Standard Deviation: | 2.48E-03 | 1.60E+00 |

[1] All concentration results in pCi/g

The average concentration for Sr-90 and tritium from each population data set is presented in Table 2.13. Table 2.13 takes these values from the mean values (or the reported value in the case of the middle core) in Tables 2.10 through 2.12.

**Table 2.13 – Characterization results as a function of core location.** Generally, Sr-90 and tritium concentrations appear to decrease with increasing depth into the concrete wall.

| Core Population Set | Number of Wafers Analyzed | Sr-90[1] | Tritium[1] |
|---|---|---|---|
| Exposed Surface: | 4 | 1.08E-03 | 8.85E+00 |
| Middle: | 1 | -2.45E-03 | 4.36E+00 |
| Internal: | 4 | 1.30E-04 | 2.69E+00 |

[1] All concentration results in pCi/g

The data indicates that the PAB wall will contain Sr-90 and tritium diffused volumetrically to some extent. The data support the assumption of residual contamination on either side of the PAB wall that tapers off with increased depth into the concrete. Although the reported concentrations were above the Operational DCGL for the characterization survey and sampling plan, the concentrations and the amount of total inventory available to be released are low enough to project meeting the acceptance criteria of the radiological assessment using the methodology described by Reference 4.19. The small amount of PAB wall remaining, and the low levels of residual contamination (as compared to the Service Building wall) justifies the conclusion that a sufficient number of cores have been collected and analyzed to characterize the PAB wall for radiological assessment.

## 2.13 Characterization of Cable Vault Footing Wall Cores

The survey and sampling plans specified concrete core collection through the Cable Vault Footing wall. Radionuclide specific analyses of concrete cores were performed to determine whether concrete concentrations met the Operational DCGLs. The Operational DCGLs for concrete were radionuclide specific and were set to achieve a dose limit of two (2) mrem in a year TEDE for tritium and ten (10) mrem in a year TEDE for Sr-90. The difference in the dose limits for the Operational DCGLs was based on the potential impact to future groundwater. These Operational DCGLs were well below the LTP and CTDEP criterion previously specified. The Operational DCGLs were used to determine radionuclide specific MDCs to ensure adequate sensitivity during analysis.

The locations for the cores were specified by the FSS Engineer using observation and process knowledge as the basis for professional judgment. The locations were in the saturated zone about three (3) feet above bedrock level. Cores were the appropriate choice for characterization of the walls given the potential for tritium and Sr-90 diffusion. The cores would then be cut in wafers, analyzed as necessary to determine the radionuclide concentration and depth of contamination.

Mobile direct-rotary drilling was used to drill through the concrete. The survey and sampling plans specified two (2) core locations. Direct-rotary drilling was to continue until soil was obtained, when possible. Total core length was approximately 18" at these locations.

Cores were cut into 1" to 2" wafers on-site using a standard brick saw with a diamond tipped blade. Selected wafers were given special sample nomenclature and were designated for off-site analysis. The FSS Engineer selected a few wafers from the end of the core that was collected first (i.e., the top 0" to 2" from the exposed surface of the wall) and a few wafers from the end of the last core collected.

The wafers were maintained under COC until turned over to the off-site laboratory for analysis. The location of the core samples is denoted on Attachment 5.6.

A total of eight (8) wafer samples from two (2) core locations were processed off-site using approved procedures and methodologies. All samples were analyzed using gamma spectroscopy to the specified sensitivities required by the survey and sampling plan. All samples were analyzed for tritium and Sr-90 to the specified sensitivities required by the survey and sampling plan. The first wafer (i.e., the closest to the exposed surface) was analyzed for all the radionuclides listed in the LTP. Those radionuclides not specified by the survey and sampling plan were analyzed to the specific sensitivities specified by the SOW. Calculated results were reported instead of <MDC. The sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations ($2\sigma$), data qualifiers, and the required and observed MDC.

The off-site laboratory certificate of analysis reported tritium in concentrations above the Operational DCGL in one (1) of the eight (8) analyzed samples. Strontium-90 was reported in concentrations above the Operational DCGL in two (2) of the eight (8) analyzed samples. Cobalt-60 was reported in concentrations that met the accepted criterion for detection in one (1) of the eight (8) analyzed samples. Cesium-137 was not identified in any of the samples. All reported results for Co-60 were below the Operational DCGL. A few of the radionuclides listed in Table 2-12 of the LTP were identified with reported concentrations that met the accepted criterion for detection. All were below the Operational DCGLs. Tables 2.14 and 2.15 summarize the data.

**Table 2.14 – Basic statistical quantities for Sr-90 and tritium from the population data set associated with the exposed surface of the Cable Vault Footing wall (the side facing the excavation).** Stronium-90 was over the Operational DCGL.

| Quantity: | Sr-90[1] | Tritium[1] |
|---:|:---:|:---:|
| Operational DCGL: | 1.51E-01 | 7.86E+00 |
| Minimum Value: | -5.32E-03 | 8.91E-01 |
| Maximum Value: | 1.09E+00 | 3.76E+00 |
| Mean: | 5.40E-01 | 1.76E+00 |
| Median: | 5.37E-01 | 1.19E+00 |
| Standard Deviation: | 6.24E-01 | 1.34E+00 |

[1] All concentration results in pCi/g

**Table 2.15 – Basic statistical quantities for Sr-90 and tritium from the population data set associated with the internal cores of the PAB wall (the deepest core).** Tritium was over the Operational DCGL.

| Quantity: | Sr-90[1] | Tritium[1] |
|---|---|---|
| Operational DCGL: | 1.51E-01 | 7.86E+00 |
| Minimum Value: | -2.12E-02 | 3.99E+00 |
| Maximum Value: | 3.03E-02 | 1.46E+01 |
| Mean: | -2.17E-03 | 7.88E+00 |
| Median: | -8.90E-03 | 6.47E+00 |
| Standard Deviation: | 2.24E-02 | 4.74E+00 |

[1] All concentration results in pCi/g

The average concentration for Sr-90 and tritium from each population data set is presented in Table 2.16. Table 2.16 takes these values from the mean values in Tables 2.14 and 2.15.

**Table 2.16 – Characterization results as a function of core location.** Strontium-90 maximum concentrations were identified with the exposed surface of the footing, while tritium concentrations tended to be higher on the other side of the footing wall.

| Core Population Set | Number of Wafers Analyzed | Sr-90[1] | Tritium[1] |
|---|---|---|---|
| Exposed Surface: | 2 | 5.40E-01 | 1.76E+00 |
| Internal: | 2 | -2.17E-03 | 7.88E+00 |

[1] All concentration results in pCi/g

The data indicates that the Cable Vault Footing wall will contain Sr-90 and tritium diffused volumetrically to some extent. The data support the assumption of residual contamination on either side of the Cable Vault Footing wall that tapers off with increased depth into the concrete. Although the reported concentrations were above the Operational DCGL for the characterization survey and sampling plan, the concentrations and the amount of total inventory available to be released are low enough to project meeting the acceptance criteria of the radiological assessment using the methodology described by Reference 4.19. The small amount of Cable Vault Footing remaining, and the low levels of residual contamination (as compared to the Service Building wall) justifies the conclusion that a sufficient number of cores have been collected and analyzed to characterize the Cable Vault Footing wall for radiological assessment.

The Cable Vault Outer wall above the footing will also remain. Two (2) sample locations were evaluated from the Cable Vault Outer wall as previously discussed. The radionuclides and levels of residual contamination between the Cable Vault Outer wall and the Service Building wall are similar. The Cable Vault Outer wall surface area is about half the Service Building surface area. The combination of similar activity and less surface area justifies the conclusion that a sufficient number of cores have been collected and analyzed to characterize the Cable Vault Outer wall for radiological assessment.

## 2.14 Characterization of Bedrock Cores

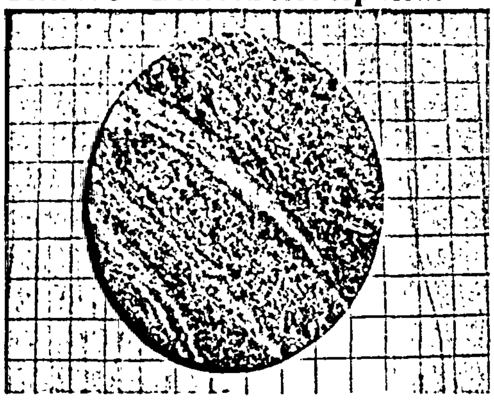
The survey and sampling plans specified core collection into the exposed bedrock. Radionuclide specific analyses of bedrock cores were performed to determine whether bedrock concentrations met the Bedrock Activity Target Concentrations. The Bedrock Activity Target Concentrations were radionuclide specific and were derived using an inventory approach to yield a future groundwater concentration of one-half (½) the USEPA MCL. These Bedrock Activity Target Concentrations were well below the LTP and CTDEP criterion previously specified. The Bedrock Activity Target Concentrations were used to determine radionuclide specific MDCs to ensure adequate sensitivity during analysis.

The locations for the cores were specified by the FSS Engineer using observation and process knowledge as the basis for professional judgment. Locations were in the saturated zone. Cores, collected from each location, were the appropriate choice for characterization of the bedrock given the potential for tritium and Sr-90 diffusion. The cores would then be cut in wafers, analyzed as necessary to determine the radionuclide concentration and depth of contamination.

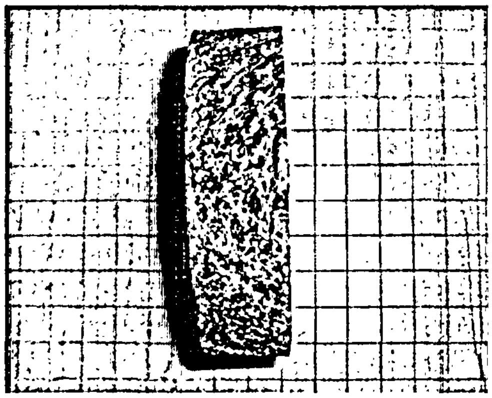
Mobile direct-rotary drilling was used to drill through the concrete. The survey and sampling plans specified thirteen (13) core locations. Direct-rotary drilling was to continue to full core-bit depth. One (1) core was collected at each location. Total core length was about 12" at these locations.

Cores were cut into 1" to 2" wafers on-site using a standard brick saw with a diamond tipped blade. Selected wafers were given special sample nomenclature and were designated for off-site analysis. The FSS Engineer selected three (3) wafers from the top 0" to 6" from the exposed surface of the bedrock. Pictures 5 and 6 show views of a typical core after cutting.

**Picture 5 – Bedrock core top view.**



**Picture 6 – Bedrock core side view.**



The wafers were maintained under COC until turned over to the off-site laboratory for analysis. The location of the core samples is denoted on Attachment 5.6.

A total of thirty-nine (39) wafer samples from thirteen (13) core locations were processed off-site using approved procedures and methodologies. All samples were analyzed using gamma spectroscopy to the specified sensitivities required by the survey and sampling plan. All samples were analyzed for tritium and Sr-90 to the specified sensitivities required by the survey and sampling plan. The first wafer (i.e., the closest to the exposed bedrock surface) from three (3) of the thirteen (13) core locations was analyzed for all the radionuclides listed in the LTP. Those radionuclides not specified by the survey and sampling plan were analyzed to the specific sensitivities specified by the SOW. Calculated results were reported instead of <MDC. The sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations (2$\sigma$), data qualifiers, and the required and observed MDC.

The off-site laboratory certificate of analysis reported Sr-90 in concentrations that met the accepted criterion for detection (i.e., a result greater than two (2) standard deviations uncertainty) in thirty-three (33) of the thirty-nine (39) analyzed samples. Tritium was identified in one (1) of the thirty-nine (39) analyzed samples. Cobalt-60 was reported in concentrations that met the accepted criterion for detection in five (5) of the thirty-nine (39) analyzed samples. Cesium-137 was reported in concentrations that met the accepted criterion for detection in three (3) of the thirty-nine (39) analyzed samples. All reported results for Co-60 and Cs-137 were below the Bedrock Activity Target Concentrations. A few of the radionuclides listed in Table 2-12 of the LTP were identified with reported concentrations that met the accepted criterion for detection. All were below the Bedrock Activity Target Concentrations. Table 2.17 summarizes the data.

**Table 2.17 – Basic statistical quantities for Sr-90 and tritium from the population data set associated with the exposed surface of the bedrock (the side facing the excavation).**

| Quantity: | Sr-90[1] | Tritium[1] |
|---|---|---|
| Operational DCGL: | 3.15E+01 | 4.41E+02 |
| Minimum Value: | -3.40E-02 | -5.07E+00 |
| Maximum Value: | 1.24E-01 | 7.39E+00 |
| Mean: | 5.43E-02 | 1.05E+00 |
| Median: | 5.70E-02 | 1.08E+00 |
| Standard Deviation: | 3.52E-02 | 3.14E+00 |

[1] All concentration results in pCi/g

The data indicates a positive trend with regards to Sr-90. Strontium-90 was identified in nearly all analyzed bedrock samples. If Sr-90 was present to depth in the bedrock, then tritium would be expected also, and in higher concentrations given the rapid ability of tritium to diffuse. Tritium was identified in only one (1) of the thirty-nine (39) samples, and at reported concentrations just above the accepted criterion for detection (i.e., a result greater than two (2) standard deviations uncertainty. A review of the analytical Method Blanks (MB) for Sr-90 finds them to be positive and a significant percentage of the analysis MDC.

The data support the assumption that residual radioactivity does not reside in bedrock. However, the radiological assessment will assume Sr-90 and tritium to be present for calculation purposes and to demonstrate compliance with the acceptance criteria using the methodology described by Reference 4.19. A sufficient number of cores have been collected and analyzed to characterize the bedrock for radiological assessment for this area. More cores may need to be collected and analyzed as the excavation exposes more bedrock.

## 2.15 In-situ Measurements using ISOCS®

Survey and sampling between January 11, 2005 and January 13, 2005 consisted of numerous data points being collected using the ISOCS® system. This method of measurement was intended to augment scanning that was being performed in the excavation. Bedrock cores were also obtained for the purpose of establishing a correlation between ISOCS® analysis results and laboratory analytic results (from the analysis of the cores). The primary data collection revolved around one (1) area of the excavation with exposed bedrock and little or no soil within the defined Field Of View (FOV) for the ISOCS®. Four (4) other locations were selected for evaluation by the field FSS Engineer using observation and process knowledge as the basis for professional judgment.

The locations for the cores were specified by the FSS Engineer using observation and process knowledge as the basis for professional judgment. Locations were in the saturated zone. Cores, collected from each location in the FOV, were the appropriate choice for the in-situ study given the lower potential for Co-60 and Cs-137 to diffuse in bedrock relative to tritium. The cores would then be cut in wafers, analyzed as necessary to determine the radionuclide concentration and depth of contamination.

Mobile direct-rotary drilling was used to drill through the concrete. The survey and sampling plans specified eight (8) core locations in a 120" radius. These eight (8) cores locations are included with the thirteen (13) bedrock locations previously discussed. Direct-rotary drilling was to continue to full core-bit depth. One (1) core was collected at each location. Total core length was about 12" at these locations.

Cores were cut into 1" to 2" wafers on-site using a standard brick saw with a diamond tipped blade. Selected wafers were given special sample nomenclature and were designated for off-site analysis. Although these cores are discussed in a previous section, for the purposes of this survey, the top three (3) wafers were to be assessed for the in-situ evaluation.

The wafers were maintained under COC until turned over to the off-site laboratory for analysis. The location of the core samples is denoted on Attachment 5.6.

All samples were analyzed using gamma spectroscopy to the specified sensitivities required by the survey and sampling plan. All samples were analyzed for tritium and Sr-90 to the specified sensitivities required by the survey and sampling plan. One (1) wafer, the first wafer (i.e., the closest to the exposed bedrock surface) was analyzed for all the radionuclides listed in the LTP. Those radionuclides not specified by the survey and sampling plan were analyzed to the specific sensitivities specified by the SOW. Calculated results were reported instead of <MDC. The sample report summaries included unique sample identification, analytical method, radionuclide, calculated result and uncertainty of two (2) standard deviations ($2\sigma$), data qualifiers, and the required and observed MDC.

The principle radionuclides, Co-60 and Cs-137 were below the MDCs of the ISOCS® and the cores samples. There was insufficient data to correlate ISOCS® measurements with bedrock core analytic results for this area. The data did demonstrate that gamma emitting radionuclides would not be present in significant quantities relative to future Bedrock Activity Target Concentrations. Additional details pertaining to the use of ISOCS® to assess the bedrock, and accompanying data, are provided in Attachment 5.7.

## 3. CONCLUSION

Survey and sampling for characterization in the excavation has identified residual activity on exposed surfaces of concrete and footings, in remaining soil and in concrete and footings to some depth. The radionuclide mix varies with sample media. Gamma emitting radionuclides, principally Co-60 and Cs-137 have been identified in soil. Beta emitting radionuclides, Sr-90 and tritium have been identified in and on concrete. No residual contamination is believed to be present in bedrock; however, Sr-90 and tritium will be assumed to be present at reported quantities for the purposes of characterization and future radiological assessment as previously discussed.

Characterization is complete for this portion of the west end of the excavation with the exception of soil. Soil analytic results demonstrate that additional remediation and subsequent sampling and evaluation is required in the northwest corner of the excavation. Survey and sampling of the remaining portion of the west excavation is necessary to fully characterize the excavation prior to the radiological assessment planned for March 2005. The characterization should include remaining soils, exposed concrete structures and footings of the former WDB and RHR pit as necessary.

## 4. REFERENCES

4.1 Haddam Neck License Termination Plan (LTP). Revision 2. August 2004.

4.2 Working files maintained under Station Administrative Control Procedure (ACP) 1.2-2.87 "NRC Regulation 10CFR50.75 (g) Decommissioning Records."

4.3 Historical Review Team Report. March 1998.

4.4 Results of Scoping Surveys. September 1998.

4.5 Augmented Characterization Survey Report. January 1999.

4.6 Characterization Report. January 2000.

4.7 Historical Site assessment. June 2000.

4.8 Survey and Sampling Work Plan SSWP BCY-04-01-001.

4.9 Health Physics Radiation Protection Manual (RPM) 5.1-00 "Final Status Survey Program".

4.10 RPM 5.1-2 "Preparation of Characterization, Remedial Action And Turnover Survey Plans".

4.11 RPM 5.1-3 "Collection Of Surface And Subsurface Soil, Shoreline Sediment, Asphalt And Liquid Samples For Scoping, Characterization And Final Status Surveys".

4.12 RPM 5.1-4 "Control And Accountability Of Portable Survey Instruments For Scoping, Characterization And Final Status Surveys".

4.13 RPM 5.1-5 "Chain of Custody for Final Status Survey Samples".

4.14 RPM 5.2-1 "Setup And Operation of The E-600 Digital Survey Instrument For Scoping, Characterization and Final Status Surveys".

4.15 RPM 5.2-2 "Operation of the Exploranium GR-130 Minispec".

4.16 RPM 5.2-3 "Operation of the Canberra Portable Gamma Spectroscopy System".

4.17 RPM 5.4-00 "Site Closure Training Program".

4.18 Initial Target Operational DCGLs for CY. CY memo ISC 04-049. November 23 2004.

4.19 Technical Basis for Performing Radiological Assessments of Bedrock, Soil and Subsurface Concrete Using an Inventory-Based Approach. Revision 1. February 15 2005.

4.20 Kd Values of Backkfill Material for Connecticut Yankee. HP TSD CY-HP-0185. November 17 2004.

4.21 Survey and Sampling Work Plan SSWP BCY-04-10-005 with Addendums.

4.22 Survey and Sampling Work Plan SSWP BCY-04-11-005.

4.23 Survey and Sampling Work Plan SSWP BCY-04-12-001.

4.24 The Gamma Rays of the Radionuclides. Erdtmann and Soyka. 1979.

4.25 Statement of Work (SOW) for Environmental, Bioassay and Waste Characterization Analytical Services. CY-ISC-SOW-001. August 27 2003.

## 5. ATTACHMENTS

5.1 Overall Excavation Footprint (2 pages including cover)

5.2 Extent of Characterization Effort as of February 2005 (2 pages including cover).

5.3 General Location of Drumming Room Leaks (2 pages including cover)

5.4 Initial Target Operational DCGLs for CY (7 pages including cover)

5.5 Technical Basis for Performing Radiological Assessments of Bedrock, Soil and Subsurface Concrete Using an Inventory-Based Approach (8 pages including cover)

5.6 Sample and Measurement Locations (2 page including cover)

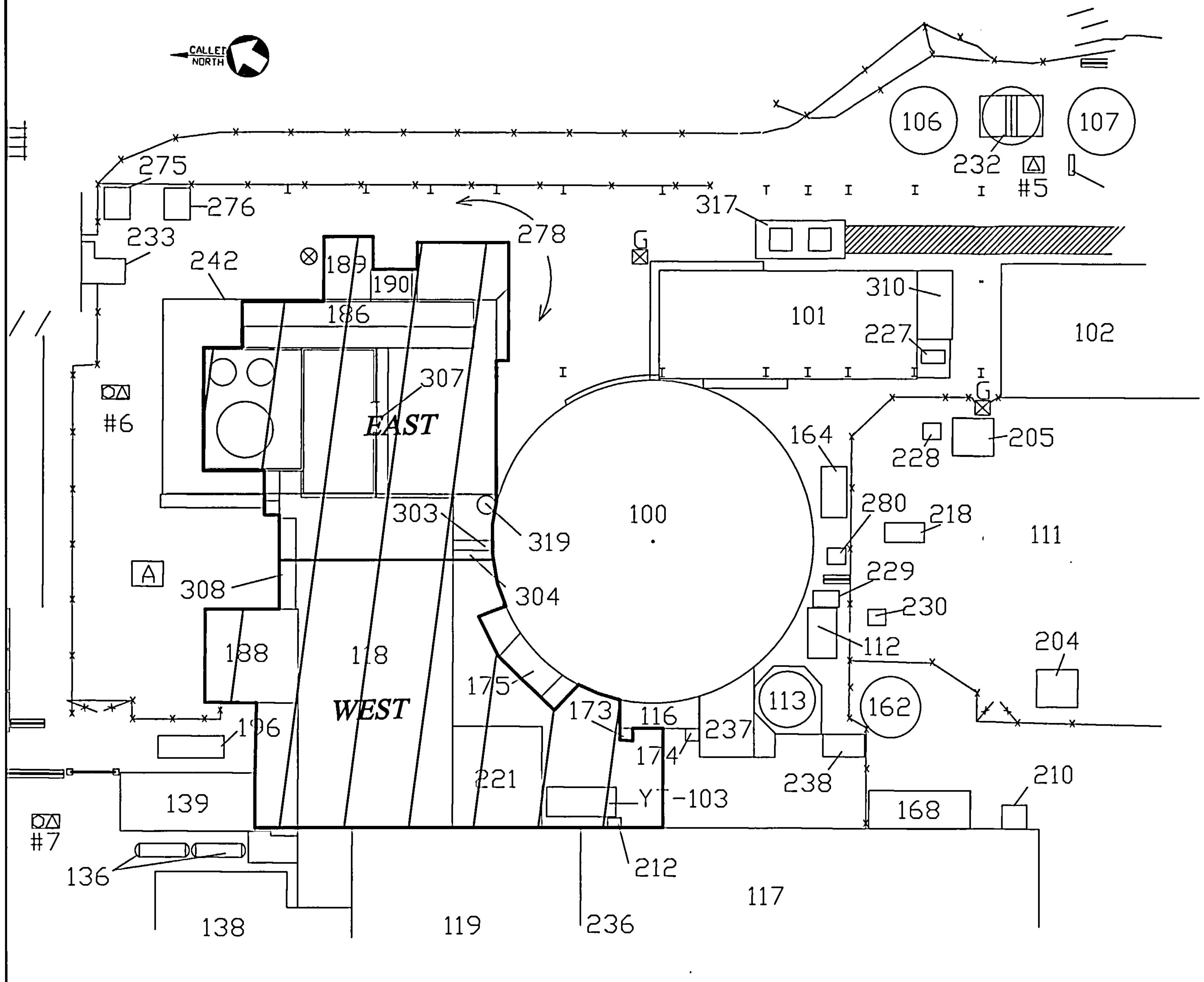
5.7 ISOCS® Assessment of West Section of the Excavation Associated with the Northeast Protected Area (8 pages including cover)

5.8 Photographs of Project and Survey and Sampling (17 pages including cover)

# Attachment 5.1

# Overall Excavation Footprint

(2 pages including cover)




NOTES:
1) THIS DRAWING IS FOR GENERAL INFORMATION ONLY. DIMENSIONS ARE APPROXIMATE. FOR DETAILS, SEE REFERENCE AND OTHER DRAWINGS AS NECESSARY.

100) CONTAINMENT BUILDING
101) SPENT FUEL BUILDING
102) RADWASTE REDUCTION FACILITY (RRF)
103) BUTLER BLDG. MAINT.STORAGE (HOT SIDE) PAD
105) MYROCK FABRICATION SHOP PAD
106) "A" MAKE-UP WATER TANK(TK-20-1A)
107) "B" MAKE-UP WATER TANK (TK-62-1A)
108) FABRICATION SHOP PAD
109) HYDROGEN STORAGE TK. CONCRETE SADDLES
110) LIQUID NITROGEN PAD
111) 115-KV SWYD. STRUCTURES & PADS
112) ELECTRIC FEED PUMP HOUSE
113) DEMIN. WATER STORAGE TANK (TK-25-1A)
114) DIESEL FUEL STOR. TANK (TK-33)
115) DISCHARGE CANAL
116) TERRY TURBINE BUILDING
117) SERVICE BUILDING
118) PRIMARY AUXILIARY BUILDING (PAB)
119) CONTROL ROOM (OLD)
120) 4.16KV BUS 12
121) TURBINE BUILDING
122) AUXILIARY BOILER ROOM
122A) AUX BOILER STACK
124) 345-KV TRANSFORMER AREA PADS & SUMP
125) SPARE GSU 12R6X TRANSFORMER PAD
127) PENINSULA / BONE YARD
128) CONTROL BUILDING FOR SEPTIC
129) SEPTIC SAND FILTERS
130) 345-KV TOWER
131) INTAKE / SCREENHOUSE
133) NEW CONTROL ROOM SEPTIC TK-132-1A & PUMPS
134) ADMINISTRATION BUILDING
135) SECURITY OFFICE PAD
136) DIESEL GEN. 5,000 GAL USTs.(2)
138) DIESEL GENERATOR BUILDING W/ USTs.
139) "B" SWITCHGEAR BUILDING
140) OFFICE BUILDING #3 & PAP
141) WAREHOUSE #2 (MAIN)

245) BUS 15 PAD
246) WARMING / DE-ICING LINE ISOLATION VALVE
247) STORM WATER ACCUMULATION BASIN
248) CONCRETE DIKE WALLS / PAD & EQUIP.
249) GATEHOUSE
250) 27K GAL. TK. 345KV XFMR FAIL OIL OVRFLO
251) 1K GAL SUMP/ Pps. 345KV XFMR FAIL OIL XFR.
252) 3500 GAL. SEPTIC TK.( AT GRADE)
253) 14,000 GAL. SEPTIC TK. - (BELOW GRADE)
254) ABANDONED SEPTIC FIELD
255) 18K GAL. RECIRCULATION TK. - (BELOW GRADE)
256) EFFLUENT PUMP STATION - (BELOW GRADE)
257) OLD OFFICE TRAILER
258) SHED ON BLOCKS
259) "TRAILER CITY" POWER DIST. HOUSE
260) 3-PHASE POWER CENTERS
261) SEPTIC SYSTEM PUMP PIT / PUMPS
262) MET. TOWER SHACK
263) WELL & PUMP P3-1C
264) WELL & PUMP P3-1D
265) WELL & PUMP P3-1A
266) WELL & PUMP P3-1B
267) YT-101 ASB. RESP. CLEAN TRAILER
268) ASB. DECON TRAILER
269) VALVE CHAMBER
270) OLD CANAL DREDGE SPOILS PILE
271) CW/ SW LINES ,SCRNHSE. TO TB (BELOW GRADE)
272) PWR &TEL DUCT BANKS/MANHOLES TO 345KV SWYD (BELOW & ABOVE GRADE)
273) WARMING / DE-ICING LINE (BELOW GRADE)
274) CONTAINMENT WING WALL BLOCKS
275) HOT TOOL CRIB
276) SEAVAN STORAGE SHED
277A) HILLSIDE STAIRS
277B) HILLSIDE STAIRS
278) LARGE CONCRETE PAD-RPV TRANSFER
279) YARD CRANE
280) CONTAINMENT MAT SUMP/ PUMPS
281) SEWAGE EJECTOR
282) GREASE PIT

142) WAREHOUSE #1
143) RCA ACCESS POINT (NEW)
144) LOADING DOCK
152) INFORMATION CENTER
153) MOTORCYCLE PARKING PAD
155) PARKING AREA
156) ADDITIONAL PARKING
157) EMERGENCY OPERATIONS CENTER
158) POND
159) ACCESS ROAD
160) CHEMICAL STORAGE WAREHOUSE
161) BLDG MAINTENANCE EQUIP WAREHOUSE
162) CONDENSATE STORAGE TANK (TK-25-1B)
164) UNCONDITIONAL REL. FAC. (URF) PAD
165) SECURITY DIESEL PAD
166) STORAGE SHED W/PAD
168) MATERIAL STORAGE (SEA-VAN) W/PAD
169) ACCESS ROAD (SOUTH)
170) FITNESS CENTER PAD
173) AUX FEED PUMP ENCLOSURE 1A
174) AUX FEED PUMP ENCLOSURE 1B
175) CABLE VAULT
176) HP PROJECT TRAILER
177) MAIN ENTRANCE BYPASS ROAD
178) BOTTLED GAS STORAGE AREA
179) P.E.G. BUILDING
180) RECORDS
181) STEAM GEN MOCKUP BLDG
182) FIRE ARMS TRAINING SIM.TRAILER W/ PAD
183) SECURITY DIESEL TRANSFORMER PAD
184) COMPACTOR & DUMPSTER W/ PAD
185) DUMPSTER W/ PAD
186) ION EXCHANGE STRUCTURE
187) 309 TRANSFORMER W/ PAD
188) WASTE DISPOSAL BLDG
189) SPENT RESIN STORAGE
190) SPENT RESIN FACILITY
192) BUTLER BLDG. INSUL. SHOP & STORAGE W / PAD

283) DIESEL FUEL TK. & PORTABLE DIKE
284) GASOLINE TK. & PORTABLE DIKE
285) HELIPAD LIGHTS
286) EOF SEPTIC FIELD
287) WELL
288) SODAR CONES & PAD
289) COMMUNICATIONS SHED & PAD
290) WELL
291) TRANSFORMER
292) POWER & COMMUNICATIONS PANELS
293) EOF SEPTIC TANK (BELOW GRADE)
294) DOCK
295) DISCHARGE STRUCTURE
296) FISH POND & FOUNTAIN
297) VCC FABRICATION PAD
298) SEWAGE EJECTOR
299) OLD RCA SHOWER DRAINS COLLECTION TK.
300) OLD RCA SHOWER DRAINS OVERFLOW TK.
301) UNIT 1 DISCHARGE TUNNEL(BELOW GRADE)
302) UNIT 2 DISCHARGE TUNNEL(BELOW GRADE)
303) CONTAMINATED (E) PIPE TRENCH
304) NON CONTAMINATED (W) PIPE TRENCH
305) SEPTIC TANK (BELOW GRADE)
306) FLOW SPLITTER BOX (AT GRADE)
307) PAB-ION EXCHANGE PIPE TRENCH
308) WDB- TANK FARM PIPE TRENCH
309) EXISTING SEPTIC FIELD
310) SPENT FUEL AUXILIARY BLDG.
311) CL&P POWER POLES (TYP.)
312) CL&P 23 KV OVERHEAD LINES
313) SEISMIC MONITOR
314) ABANDONED 1K GAL. SEPTIC TK.(BELOW GRADE)
315) ABANDONED SEPTIC FIELD
316) RIP-RAP WEIR
317) TFR PREP BUILDING
318) HEAVY HAUL ROUTE-To Barge Landing And South Access Road
319) PRIMARY VENTILATION STACK

196) AIR COOLED DIESEL GEN. PAD
197) 480 V BUS 10 SWGR W/PAD
199) 115-KV TOWER
200) 4 PLEX MODULAR
201) PBX ROOM
202) PBX TRANSFORMER
203) DIESEL FUEL STOR. TANK (TK-126-1A)
204) STATION SERVICE XFMR 12R-22S (399)
205) STATION SERVICE XFMR 12R-21S (389)
207) TRASH STORAGE
210) LIQUEFIED PETROLEUM GAS TANK PAD
212) CHEM LAB GAS STORAGE BOTTLES PAD
213) STORAGE SHED
214A) HTG. OIL TK. 275 GAL.W/ PORTABLE DIKE
214B) HTG. OIL TK. 275 GAL.W/ PORTABLE DIKE
215) WASTE OIL TANK W/ PORTABLE DIKE
218) 115-KV TIE BKR.
221) HP FACILITY
224) BARGE LANDING
225) POWER DIST. EQUIP.PAD SOUTH EAST YARD
226) POWER DIST. EQUIP SOUTH YARD W/ PAD
227) EG-8 W/ PAD
228) BUS 13 TRANSFORMER
229) 480V BUS 16
230) BUS 16 TRANSFORMER
231) HEAT TRACE PANEL W/ PAD
232) RAD WASTE "C" VAN W/W.G.SPHERE PAD
233) RCA ACCESS POINT
234) BICRON TRUCK RAD MONITOR
235) WEATHER STATION
236) AUXILIARY BAY
237) ALT. CONTAINMENT ACCESS
238) CLEAN STORAGE SHED W/ PAD
239) HTG. OIL TK. 1000 GAL. W/PORTABLE DIKE
240) 8 - PLEX OFFICES
241A) SHELTER
241B) SHELTER
242) TANK FARM TEMPORARY SHELTER
243) ABANDONED SEPTIC TANK(BELOW GRADE)
244) SEWAGE EJECTOR

LEGEND

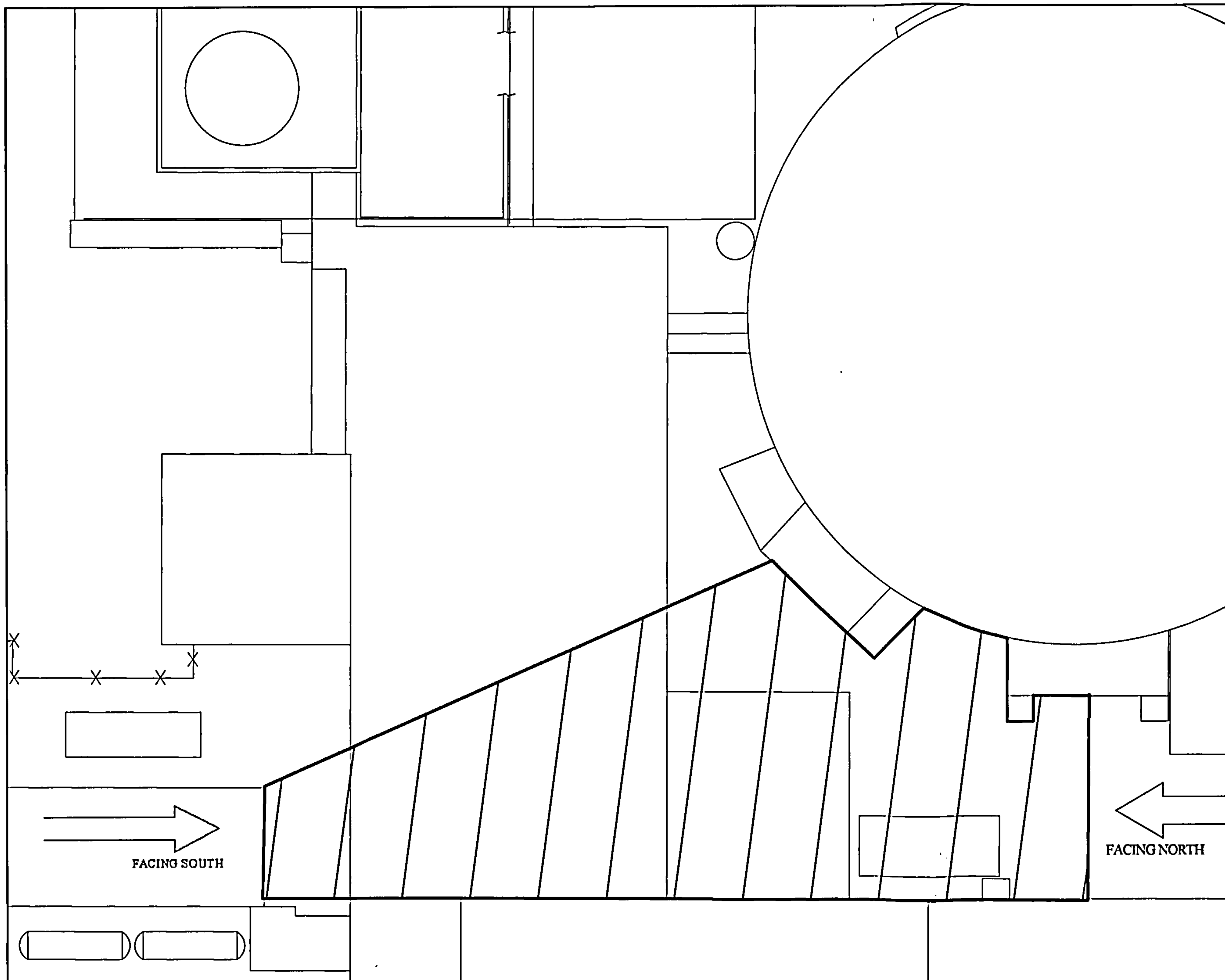
Extent of excavation

| Attachment 5.1 | Connecticut Yankee Atomic Power Company<br>Overall Excavation Footprint | Date | By |
|---|---|---|---|
| | | February 2005 | J McCarthy |

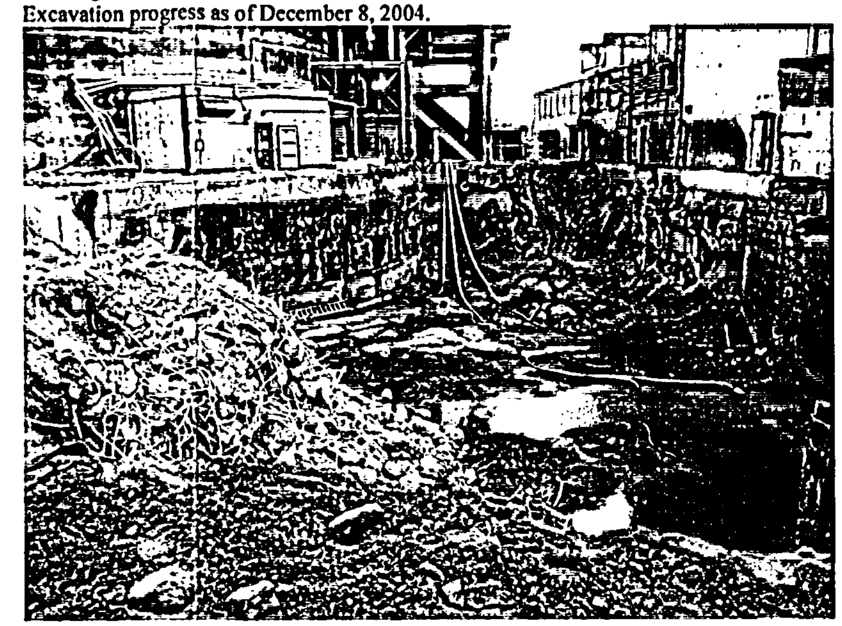
# Attachment 5.2

## Extent of Characterization Effort as of February 2005

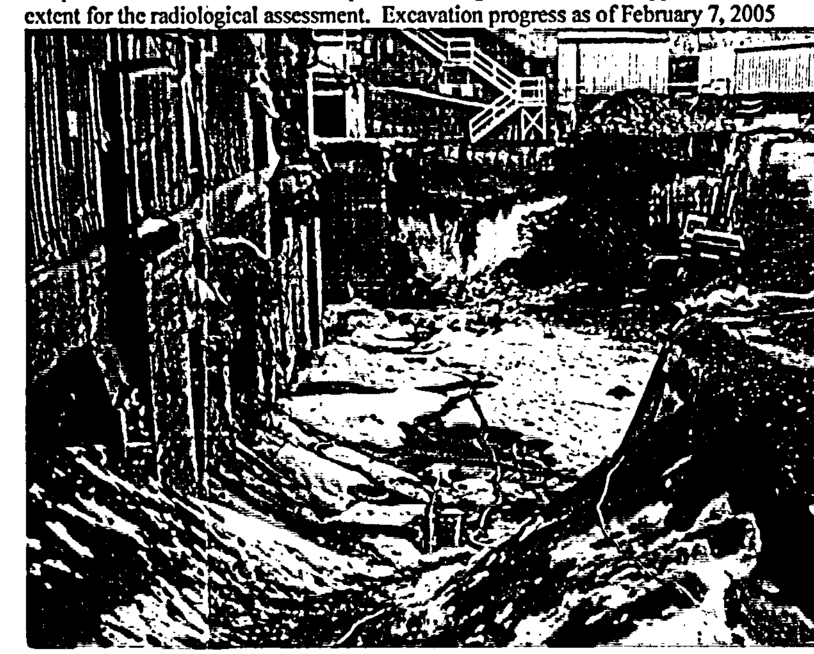
(2 pages including cover)




**Picture 1 – Earlier view of the excavation facing south.** The Cable Vault and Service Building remaining structures can be seen to the left and right of the picture, respectively. Excavation progress as of December 8, 2004.



**Picture 2 – Current view of the excavation facing south.** The excavator, located in the footprint of the former Waste Disposal Building, delineates the approximate northern extent for the radiological assessment. Excavation progress as of February 7, 2005



| | | Date | By |
|---|---|---|---|
| *Attachment 5.2* | *Connecticut Yankee Atomic Power Company*<br>*Extent of Characterization to February 2005* | *February 2005* | *J McCarthy* |

# Attachment 5.3

## General Location of Drumming Room Leaks

(2 pages including cover)

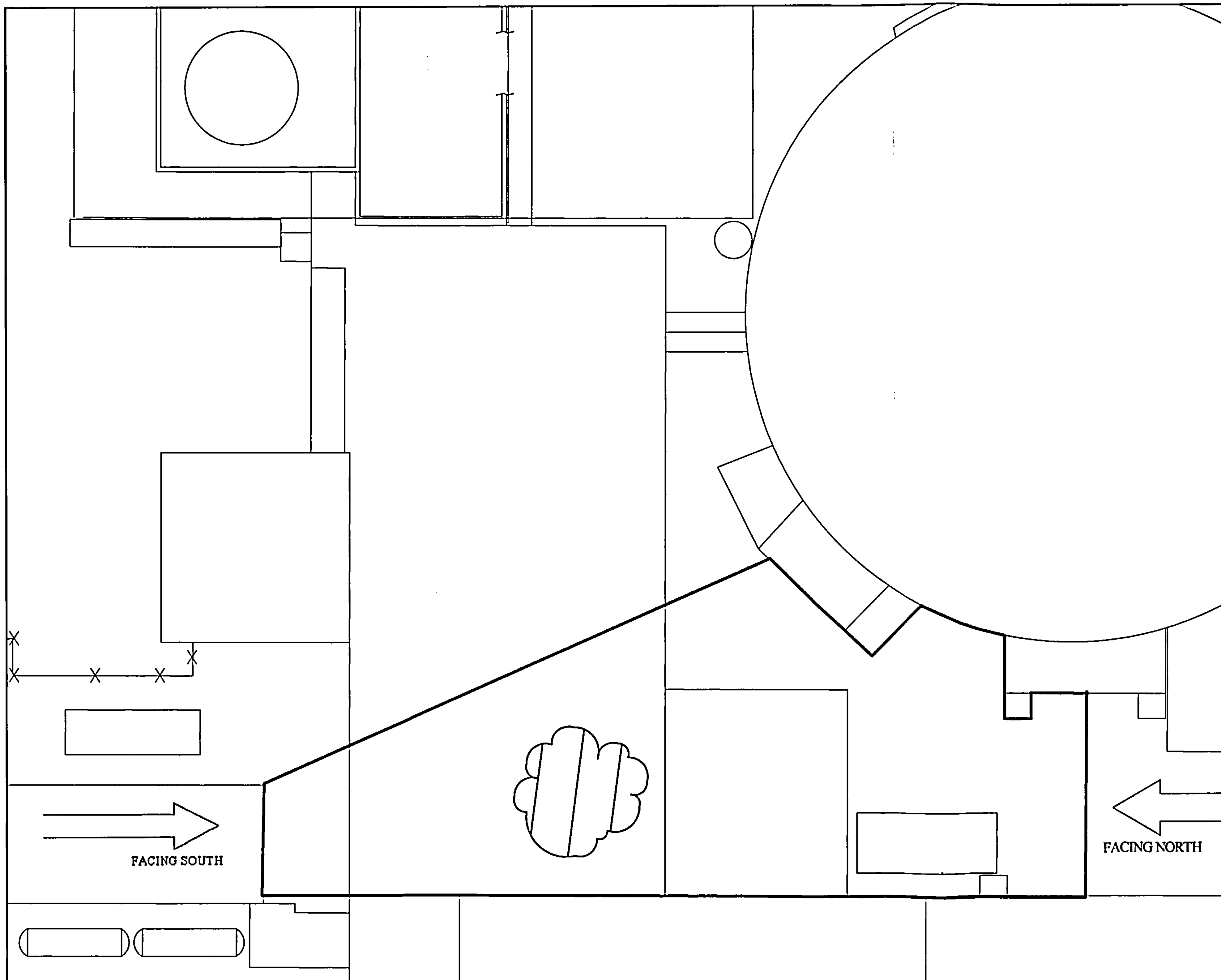



**Figure 1 –Configuration of the PAB Drumming Room in 1980.** The layout shows a compactor and two rollers used to mix cement and liquid waste for solidification. A waste storage area is shown adjacent to Drumming Room and was added as a building under DCR-CY-5520-79. The waste storage building was converted into the former Health Physics Facility sometime in the 1980's. The rollup door provided access and egress to the PAB and Service Building alleyway. The revision cloud on the drawing shows the approximate location of the leaks.

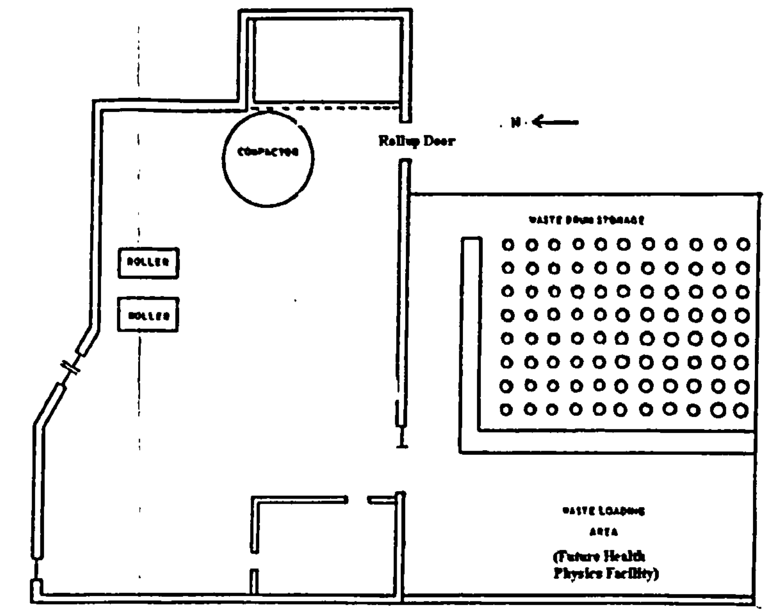



**Figure 2 –Aerial view of the RCA from July 2003.** The numbers point out specific locations of interest as reference points to this radiological characterization report. The number 1 denotes the tank farm area which is covered by a tent. The number 2 denotes the PAB and the number 3 denotes the Health Physics Facility.


| | | Date | By |
|---|---|---|---|
| *Attachment 5.3* | *Connecticut Yankee Atomic Power Company*<br>*General Location of Drumming Room Leaks* | *February 2005* | *J McCarthy* |

# Attachment 5.4

# Initial Target Operational DCGLs for CY

(7 pages including cover)

CONNECTICUT YANKEE ATOMIC POWER COMPANY
HADDAM NECK PLANT
362 INJUN HOLLOW ROAD • EAST HAMPTON, CT 06424-3099

November 23, 2004
ISC 04-049

TO: File

FROM: R. McGrath

SUBJECT: Initial Target Operational DCGLs for CY

The purpose of this memo is to establish Initial Target Operational Derived Concentrations Guideline Levels (DCGLs) for use in Final Status Survey (FSS) design at CY. Operational DCGLs are needed when a Final Status Survey is to be conducted in areas potentially affected by existing groundwater contamination or containing contaminated structures in addition to soil contamination. All Final Status Surveys conducted on site to date have been in areas with the potential of only soil contamination. As the near term FSS schedule includes areas potentially affected by existing groundwater and contaminated structures, target operational DCGLs are needed at this time.

<u>Background</u>

Equation 5-1 of the CY LTP presents the framework for setting operational DCGLs. Equation 5-1 is as follows ("Dose" substituted for "H" in this equation as discussed in the text of LTP section 5.4.7.1):

$$\text{Dose}_{\text{Total}} = \text{Dose}_{\text{Soil}} + \text{Dose}_{\text{Existing Groundwater}} + \text{Dose}_{\text{Future Groundwater}}$$

Definitions of the terms of equation 5-1 are as follows:

$\text{Dose}_{\text{Total}}$: The combined TEDE dose to the average member of the critical group due to residual radioactivity above background from all dose pathways.

$\text{Dose}_{\text{Soil}}$: The portion of the dose from all pathways that is contributed by the soil related pathways.

$\text{Dose}_{\text{Existing Groundwater}}$: The portion of the dose from all pathways that is due to residual radioactivity currently in groundwater on site that is still present within the capture zone of a survey area at the time of release of the survey area from the CY NRC license. In the case of the CT DEP criteria discussed below, this criteria applies at the time of property transfer.

Dose $_{\text{Future Groundwater}}$: The portion of the dose from all pathways due to residual radioactivity that is projected to be present in groundwater due to the leaching of residual radioactivity from concrete debris from CY buildings that are buried on site at the time of removal of a survey area from the license. Although the LTP is silent on the topic of buried concrete basements and foundations this criteria is being taken to apply to this type of buried concrete in addition to "debris".

Per the NRC regulation and the CY LTP, "Dose Total" can be no more than 25 mrem/yr TEDE from all pathways. This requirement applies at the time of removal of a survey area from the CY part 50 license. Additionally, the Connecticut DEP in a letter dated November 20, 2002 adopted a radiological remediation standard of 19 mrem/yr TEDE to the average member of the critical group. Whereas, the CT DEP standards are applicable under the Connecticut Property Transfer Act, they apply at the time of property transfer. Both the NRC and CT DEP standards require an ALARA evaluation to determine if additional remediation is justified. It should be noted that CY has proposed to the CT DEP that the Maximum Contaminated Levels (MCLs) defined by the US Environmental Protection Agency for groundwater contamination will be achieved at CY at the time of property transfer.

There is an additional operational DCGL that needs to be set for use in the FSS of buildings that will remain after license termination. This is the Building Occupancy DCGL. This DCGL is independent of soil, existing groundwater and future groundwater.

<u>Effect of Changes to Demolition Approach</u>

CY has changed its planned approach to decommissioning from that described in CY LTP Revision 1a. We no longer plan on using concrete debris from the demolition of buildings to backfill the basements of those buildings. All concrete above 4 ft below grade will be removed from site and dispositioned at an appropriate disposal site. Many internal concrete structures such as those inside the containment liner will also be disposed of off-site. In lieu of concrete debris, clean backfill material from off-site locations will be utilized for backfill of basements.

To define further the earlier compliance equation, the different dose components that make up the total dose from the applicable pathways are stated as follows in the CY LTP.

$$1 \geq f^{i}_{Soil} + f^{i}_{ExistingGW} + f^{i}_{w} f^{i}_{ConcreteDebris}$$

Where each of the terms (or product) represents a fraction of the 25mrem/yr maximum dose. The term $f_w$ represents the fraction of the total dose determined to apply to "concrete debris" that is from water dependent pathways as described in the CY LTP. The last term of the above equation is also called "future groundwater".

Whereas concrete debris will no longer be used for building backfill, a proposed license amendment to the CY LTP has been prepared to reflect the current decommissioning approach. The last factor of above equation is being replaced with a dose component that corresponds to the dose from water dependent pathways that are the result of the leaching of radionuclides from basement concrete, footings, liner surfaces and any piping that contacts groundwater after the completion of remediation and the backfilling of basements. Based on characterization data and expected modeling results the estimated dose for the last factor in the above equation has been calculated. For the expected worst case area (containment basement) the estimated dose is approximately 1 mrem/yr. Characterization results also indicate that if the difficult remediation of the concrete surrounding the In Core Sump is performed, a 2 mrem/yr future groundwater dose is achievable without the changes contained in the proposed LTP amendment. Although no data has been obtained for the spent fuel pit, options are available to achieve a 2 mrem future groundwater dose for that building. It is therefore proposed that the initial value for the trigger for the "future groundwater" dose term of the compliance equation be set at 2 mrem/yr until additional characterization data is available and the status of NRC approval of the proposed license amendment is clarified at which point this target value possibly could be reduced slightly.

Additional Considerations

**Soil:** CY has been utilizing an administrative value 10 mrem/yr for the conduct of all final status surveys to date. As mentioned before, none of the FSS survey areas to date have been impacted by existing groundwater or contaminated concrete. It is desirable to maintain this administrative value for the sum of the soil and future groundwater dose components of the compliance equation (i.e. Dose for Existing Groundwater not included). With this in mind the Target Operational DCGL for soil is set at 8 mrem/yr when the future groundwater target value is subtracted from the administrative dose value (i.e. 10 minus 2).

Note that remediation to lower levels may be required in areas that exhibit measurable concentrations of certain relatively mobile radionuclides. This is due to the commitment to CT DEP to achieve the EPA MCLs as discussed above. Sr-90 is the primary radionuclide to which this will apply. Sr-90 is relatively mobile and has a relatively low MCL. After characterization has been performed and remediation or final status survey planning is in progress, the Site Closure Technical Support Manager (SCTSM) and/or the Groundwater Program Lead (GPL) should be consulted to review the characterization data. The consultation will determine the need for lower remediation target levels in order to facilitate the achievement of the MCLs in the future.

**Existing Groundwater:** Another consideration is the criteria for dose from "existing groundwater" contamination. The value to be used in the compliance equation will depend on the groundwater monitoring sample results at the time that release is being requested. In the case of the NRC license, that time is the submittal of the request to remove the affected area from the NRC license. Concerning the State of Connecticut this time is when CY requests approval to transfer the property.

There is no need to set a target value for groundwater for NRC compliance as this value could be impacted downward by a revision of soil and future groundwater target values. It is also possible that the request to NRC for release of an area from the NRC license will need to wait for groundwater monitoring sample results to show compliance with the NRC release criteria.

CY has proposed a standard of the EPA Maximum Contaminant Levels (MCLs) for compliance with the State of Connecticut Remedial Standard Regulations. Compliance with the EPA MCLs would result in a "Existing Groundwater" dose component factor slightly less then 1 mrem/yr TEDE. Compliance with the MCL would be required at the time of property transfer which occurs after termination of the NRC license for a particular area of the site. Although this is the final target for "existing groundwater" dose component, it need not be required as a near term target except for determining the maximum target for the other dose components. That is to say, CY also needs to comply with the State of Connecticut Radiological Remediation Standard of 19 mrem/yr TEDE from all pathways. Therefore, the sum of the soil and future groundwater components can *be no more then 18 mrem/yr at the time of property transfer, allowing for an existing* groundwater dose of 1 mrem/yr. Considering the above discussion, the establishment of a groundwater target is not required.

Implementation Considerations

Among the challenges involved in the use or the proposed operational DCGLs are those associated with the implementation of the target soil DCGL in areas potentially impacted by "future groundwater". The attached graph illustrates the effect of area background radiation levels on the number of soil samples that are required in Class 1 areas. The number of samples required when background is at or below approximately 10,000 cpm corresponds to the normal level of approximately 20 samples. It can also be seen that when background levels exceed 20,000 cpm, the number of samples required increases dramatically even at our administrative dose level of 10 mrem/yr for some background levels. Recently conducted background surveys of the CY industrial area have shown the background levels currently exceed 10,000 cpm even in areas well away from areas containing source material. Although these levels will decrease as decommissioning continues, it is likely that remedial action surveys and final status surveys will need to be conducted in areas where background exceed 10,000 cpm. In order to achieve the target operational soil DCGLs of 8 mrem/yr in these areas, the use of alternative methods such as In-situ Gamma Spectroscopy (Trade name "ISOCS") may be justified to offset the expense of a greater number of soil samples. Experience at Maine Yankee has shown that the use of ISOCS in lieu of scanning in high background areas has kept the number of soil samples required in Class 1 areas to a reasonable value.

Initial Operational DCGL Target Values

The following summarizes the various initial target values:

Dose $_{Soil}$ (Areas with no potential for future groundwater dose): 10 mrem/yr (Subject to SCTSM and/or GPL review of characterization data)

Basis: To continue current 10 millirem/yr administrative limit when "existing" groundwater dose is not included.

Dose Soil (Areas potentially impacted by "future" groundwater): 8 mrem/yr (Subject to SCTSM and/or GPL review of characterization data)

Basis: To continue current 10 millirem/yr administrative limit when "existing" groundwater dose is not included and the future groundwater target has been subtracted.

Dose Existing Groundwater: Not required until time of Release from NRC license or Property Transfer

Basis: The allowable dose for "existing" groundwater will be determined by the actual dose values for the other components of the compliance equation.

Dose Future Groundwater: 2 mrem/yr

Basis: Conservatively set at this value to cover potential dose from contamination leaching from concrete and released from metal surfaces that will be in contact with groundwater after site closure

Dose Soil plus Future Groundwater: 18 mrem/yr (**Note:** Exceeding individual target values is allowed only after Site Closure Manager Approval as described below)

Basis: Assures achievement of State of Connecticut release limit of 19 mrem/yr when groundwater dose due to MCL concentrations are included.

Operational Building Occupancy DCGL: 10 mrem/yr

Basis: As this criteria is independent of the soil, existing and future groundwater criteria, it can be set at the administrative site release dose limit.

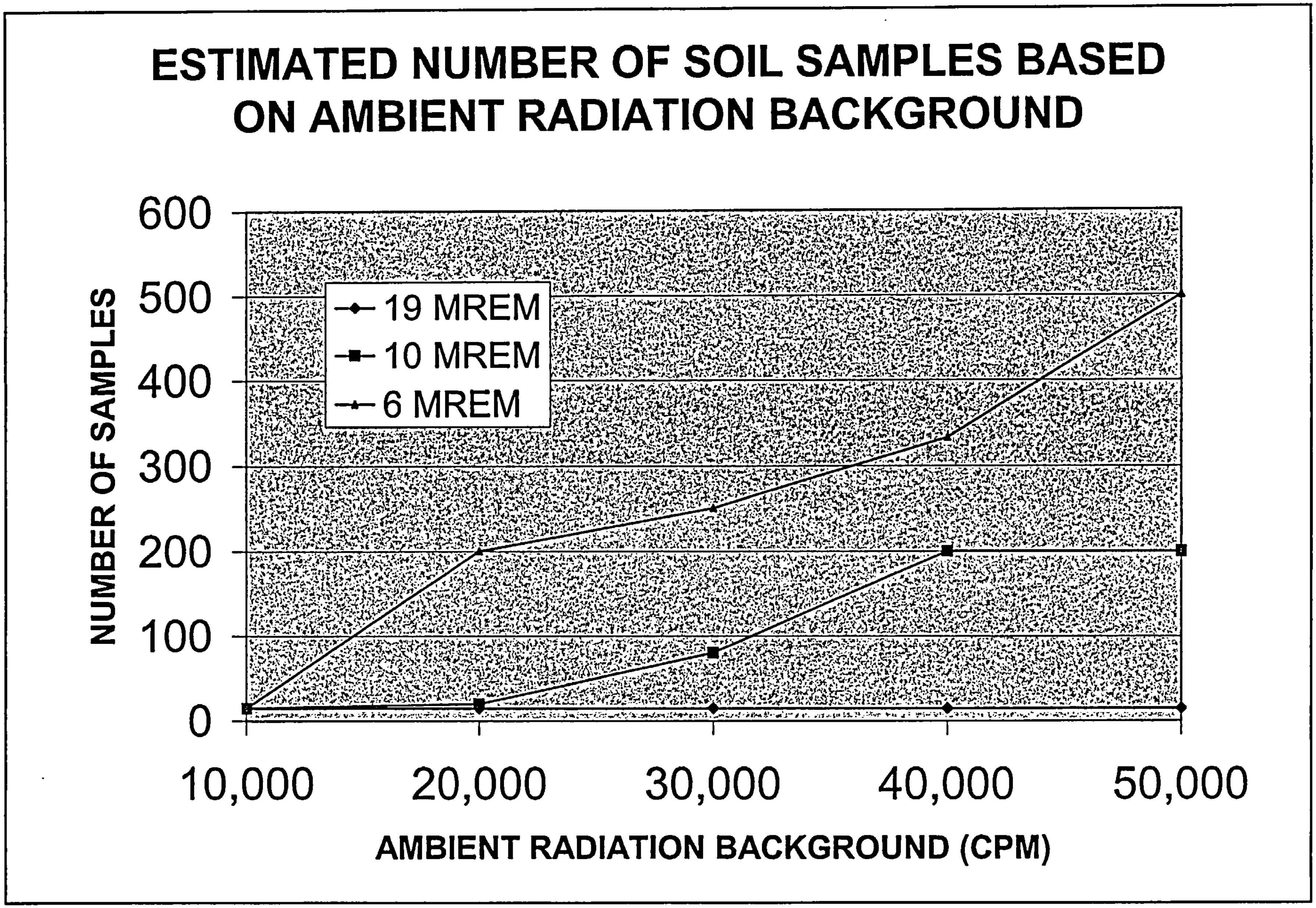
Approval Process for Exceeding Individual Operational DCGL Target Values

The Target Operational DCGLs for soil and future groundwater can be used at this time without further approvals. There is some flexibility in the target values that can be set for individual DCGLs while maintaining compliance with the applicable regulations and requirements. This flexibility is due to the differences in the applicable criterion and in the point in time that the criterion is invoked.

Considering the above, changes can be made to the Operational DCGL Target Values with the approval of the Site Closure Manager. This approval will be based on a cost analysis sufficient to justifying the higher DCGL target. Situations may emerge where significant costs could be saved on certain activities by using a higher administrative operational DCGL for soil plus future groundwater than 10 mrem/yr. Use of higher values should be evaluated on a case-by-case basis, considering project economics. The target level may be increased based on economic evaluations that consider the savings from reduced labor, waste disposal, and final status survey costs.

Please contact me with any questions on this memo.

Cc: T. Peacock
C. Miller
R.Yetter
G. van Noordennen
E. Darois
C. Newson
B. Couture
M. Atkins
J. McCarthy
P. Hollenbeck
K. Heider


ESTIMATED NUMBER OF SOIL SAMPLES BASED ON AMBIENT RADIATION BACKGROUND
600
500
400
300
200
100
0
NUMBER OF SAMPLES
19 MREM
10 MREM
6 MREM
10,000
20,000
30,000
40,000
50,000
AMBIENT RADIATION BACKGROUND (CPM)

# Attachment 5.5

## Technical Basis for Performing Radiological Assessments of Bedrock, Soil and Subsurface Concrete Using an Inventory-Based Approach

(8 pages including cover)

CY HEALTH PHYSICS
TECHNICAL SUPPORT DOCUMENT

# Connecticut Yankee Decommissioning

## Health Physics Department Technical Support Document

**HP Number: CY-HP-0190 Revision #: 1**

**Subject: Technical Basis for Performing Radiological Assessments of Bedrock, Soil and Subsurface Concrete Using an Inventory-Based Approach**

**Date: 2/15/05**

Performed By: Richard N. McGrath Date: 2/15/2005

Reviewed By: Jack McGrath Date: 2/15/05

Approved By: Date: 2/15/05

# Technical Basis for Performing Radiological Assessments of Bedrock, Soil and Subsurface Concrete Prior to Performing Backfill

## 1.0 Introduction

The CY LTP requires that excavations resulting in exposed bedrock be radiologically assessed prior to backfill. The objective of this radiological assessment is to determine the potential impact of any residual radioactivity contained in or on bedrock or subsurface concrete surfaces to potential future groundwater contamination. An additional case is included to allow an assessment of excavations that include areas of soil in addition to bedrock. This document provides the basis and general methodology for performing these assessments.

## 2.0 Analysis and Discussion

### 2.1 General Model

The soil excavations performed to support the decommissioning will result in exposed bedrock areas. These areas may be in areas that have been designated as Radiological Control Areas (RCAs) and adjacent areas to the PAB, Containment, Tank Farm, and Spent Fuel Building. The CY LTP requires that these exposed bedrock areas be assessed to determine the magnitude of contamination and its potential impact on possible future groundwater contamination.

It should be noted that the general model describes the case where the excavation is bounded by only bedrock and/or concrete. There is an additional case where the surfaces of the excavation include bedrock and/or concrete and additionally a relatively shallow layer of soil covering the bedrock that has been shown to meet the CY LTP Derived Concentration Guideline Levels (DCGLs). The model for this additional case is described in Section 2.3.

Reference 4.1 represents a request from CY to the NRC to amend the LTP to use an inventory based approach to calculate the future groundwater dose component for buried subsurface structures. This approach employs the following expression to calculate the future groundwater concentration, $C_{w,i}$, from nuclide i:

$$C_{w,i} = \frac{B_i \sum_{s=1}^{N} CFR_{s,i} A_{s,i}}{R_i V} \quad \text{(eq 1)}$$

Where:

- $CFR_{s,i}$ is the cumulative fractional release for each subsurface structure, s, and each radionuclide, i,

- $A_{s,i}$ is the total activity of radionuclide i contained in the subsurface structure s,
- $B_i$ is the radionuclide specific buildup factor (this accounts for an increase in groundwater concentrations for radionuclides that diffuse relatively slowly compared to tritium) in years following the first,
- V is the dilution water volume, and
- $R_i$ is the radionuclide specific retardation factor (this accounts for the adsorption of radionuclides on the backfill material in the basement, calculated from the backfill soil distribution coefficients ($K_d$) determined in the Brookhaven National laboratory study of $K_d$ values for CY backfill soil (Ref 4.2 except for H-3 and Eu-152 which are from the CY LTP and taken to apply to both soil B and soil A/B)), the soil density ($\rho$, from the CY LTP), and the soil porosity ($\eta$ also from the CY LTP) as:

$$R = 1 + \frac{\rho K_d}{n} \quad \text{(eq 2)}$$

From the CY LTP, the soil density and porosity have values of 1.56 and 0.35 respectively. The distribution coefficient values are dependent on the soil types used to backfill the excavations. Three types of soil have been evaluated for backfill in reference 4.2, soil A, soil B and soil A/B mix. This evaluation concludes that two possible backfill soils will be used: Type B, and Type A/B mix. The values of Kd for selected radionuclides are taken from reference 4.3 and are provided in Table 1.

In order to determine the dilution water volume, V, in equation 1, the physical volume of an area, $V_a$, must be corrected using the soil porosity $\eta$ as:

$$V = \eta V_a \quad \text{(eq 3)}$$

Table 1 also provides values for the retardation factor, R, (from reference 4.3) for selected radionuclides.

Table 1: Parameter Values In Support of Excavation Assessments

| Radionuclide | $K_d$ (Soil B) | $K_d$ (Soil A/B) | R (Soil B) | R (Soil A/B) |
|---|---|---|---|---|
| H-3 | 0.06 | 0.06 | 1.26 | 1.26 |
| Fe-55 | 1200 | 1200 | 5350 | 5350 |
| Co-60 | 220 | 22 | 982 | 99 |
| Sr-90 | 10 | 44 | 46 | 197 |
| Cs-137 | 149 | 45 | 665 | 202 |
| Eu-152 | 825 | 825 | 3678 | 3678 |

## 2.2 Implementation of the General Model

In applying equation 1 to an area and volume of an excavation that contains exposed bedrock and/or concrete building foundations, the Cumulative Fractional Release (CFR) and Buildup factors (B) are set to 1.0. These conservative values assumes that 100% of the radioactivity will be liberated from all surfaces (bedrock and the subsurface concrete) and enter the groundwater volume in the first year following the backfill. A value of 1 for the buildup factor is appropriate for this assumption (CFR=1) since no activity would be available on the surface to increase groundwater concentrations in subsequent years.

The total activity available from the surfaces within the excavation, $A_i$, is calculated from the surface activity, $S_i$, and the area of the surface, a, as:

$$A_i(\text{pCi}) = \frac{S_i(\text{dpm/100cm}^2)\times a(\text{cm}^2)}{100\times 2.22\text{dpm/pCi}} \quad \text{(eq 4)}$$

Therefore, for CFR and B values of 1.0 and substituting equations 3 and 4 into equation 1, the future groundwater concentration from a single excavation area in pCi/L is expressed as:

$$C_{w,I} = 4.505\left(\frac{S_i}{R_I}\right)\frac{a}{V_a\eta} \quad \text{(eq 5)}$$

Where:

$S_i$ is the surface contamination level of nuclide i, in dpm/100cm,
R is the retardation factor for nuclide i,
$a/V_a$ is the surface area to volume ratio for the excavation in $cm^{-1}$, and
$\eta$ is the soil porosity.

In cases where $a/V_a$ is expressed in $ft^{-1}(ft^2/ft^3)$, equation 5 becomes:

$$C_{w,I} = 0.1478\left(\frac{S_i}{R_I}\right)\frac{a}{V_a\eta} \quad \text{(eq 6)}$$

## 2.2 Example Case

Consider an excavation where Sr-90 is the nuclide of concern, where the backfill will be a mix of A/B soil types and the area to volume ratio has a value of approximately 1/8 $ft^{-1}$. This example corresponds to the excavation that is currently being performed in the area of the west end of the former Primary Auxiliary Building. In this assessment it is assumed that the excavation is totally isolated from the flow of water. Actually there are areas on the sides of the excavation that consist of soil that meets the Operational DCGLs for soil. The calculation that follows is conservative as these openings allow the passing of radiologically clean groundwater through the excavation. Credit is not taken for the dilution that would result from this groundwater flow. The radionuclide content of any soil that makes up the side walls of the excavation does not need to be accounted for in the assessment of the excavation as all the dose from this material is accounted for in the separate land area final status survey performed for this soil.

This excavation has a total surface area (for bedrock, and concrete side walls) of approximately 11,000 $ft^2$ (bedrock surface area is approximately 8,900 $ft^2$). The average depth of groundwater above bedrock (with no area dewatering in progress) in this area is 10 feet. This yields a total volume of saturated soil/backfill of 89,000 $ft^3$. In this case, R has a value of 197.11, the soil porosity $\eta$ is 0.35, and the target groundwater concentration $C_{w,i}$ is ½ of the MCL for Sr-90, or 4.0 pCi/L. Substituting into Equation 6 is illustrated by the following:

$$4\text{pCi/L} = 0.1478\left(\frac{S_i}{197.11}\right)_{Sr-90} \frac{11{,}000\text{ft}^2}{89{,}000\text{ft}^3 \text{x} 0.35}$$

Solving for $S_i$ results in an average surface activity concentration of 15,106 dpm/100$cm^2$ for Sr-90.

This areal activity concentration is converted into a volumetric concentration by assuming that the density of the bedrock is 2.65 g/$cm^3$ and that 100% of the activity is contained within the first inch. Assuming the target contamination on a 100 $cm^2$ surface area is converted to an equivalent activity for a volume of 100 $cm^2$ area to a volume of 1 inch is performed as follows(this calculation results in the lowest target concentration as concrete, having a lower density, would result in a higher concentration):

- The volume corresponding to a 100 $cm^2$ surface area is:

  100 $cm^2$ X 2.54 cm(for the 1 inch depth) = 254 $cm^3$

- The mass corresponding to this volume of bedrock is:

  254 $cm^3$ X 2.65 g/$cm^3$ = 673.1 g

- The total activity corresponding to the target contamination level is:

  15,106 dpm/100$cm^2$ X 0.45 pCi/dpm = 6,798 pCi

- The concentration in the volume of bedrock at the target contamination level is:

  6,798 pCi/673.1g = 10.1 pCi/g

With these assumptions, the volumetric activity concentration from an areal activity concentration of 15,106 dpm/100cm$^2$ is 10.1, or 10 pCi/g.

## 2.3 Special Case – Soil Remaining in the Excavation

There may be situations where the excavation being evaluated has areas of exposed bedrock along with areas that contain soil above the bedrock that has been shown to have radionuclide concentrations that are below the Operational DCGLs for soil for the survey unit. It is expected that the total soil volume to remain above will be less then 10% of the total saturated volume of the excavation. In this case, the assessment of the excavation will be performed as follows:

2.3.1 Using soil characterization sample results, a depth and radionuclide concentration profile will be determined for the soil that is above the bedrock that is to remain inside the excavation (this area does not include any side walls of the excavation as discussed in the Section 2.2 example).

2.3.2 These profiles will be used to calculate the volume of contaminated soil above the bedrock and the total radionuclide inventory of that soil.

2.3.3 This radionuclide inventory in the soil will be assumed to be released to the groundwater in the excavation within the first year after backfilling and securing of excavation dewatering.

2.3.4 Equation (1) above will now be used to calculated the resulting groundwater concentration due to the soil as follows (B, Buildup Factor and CFR are assumed to be 1 as previously discussed for bedrock):

$$C_{w,i} = \frac{\sum_{s=1}^{N} A_{soil,i}}{R_i V} \quad \text{(eq 7)}$$

Where:

- $A_{soil,i}$ is the total activity of radionuclide i contained in the soil s,
- V is the dilution water volume, and
- $R_i$ is the radionuclide specific retardation factor (shown in Table 1)

To continue the example in Section 2.2, it is assumed that 50% of the bedrock in the floor of the excavation (4,450 ft$^2$) is covered with 2 foot of soil (this volume is 10 % of the saturated volume of the excavation). The radionuclide concentration of the soil is assumed to be the CY LTP Soil DCGL concentration for Sr-90 (1.55 pCi/g). The components of Equation 7 are calculated as follows:

- The total activity, $A_{soil,Sr\text{-}90}$ is calculated as follows:

  4,450 ft$^2$ x 2 ft x 28317 cm$^3$/ft$^3$ x 1.56 g/ cm$^3$ x 1.55 pCi/g = 6.09 E8 pCi

- V, dilution volume is calculated by:

  89,000 $ft^3$ x 28.317 liters/$ft^3$ x 0.35 (porosity) = 8.82 E5 liters

- Substituting into Equation (7):

$$C_{w,Sr-90} = \frac{6.09E8}{(197)(8.82E5)} = 3.5 pCi/L$$

This resulting groundwater concentration will be added to the other groundwater concentrations calculated for the other sources to the excavation (bedrock and/or concrete) to determine the total future groundwater concentration.

There is a difference in the Kd factors for the soil to be used for backfill (Kd = 44 for Sr-90) compared to the Kds used in the CY LTP table F-1(Kd = 32 for Sr-90) for the soil at the site. The use of the above method is limited to cases where the quantity of soil to remain in the excavation is no more then 10 % of the saturated soil volume to ensure that the effect of difference in soil Kds is within the error of the calculation. If larger quantities of original soil are to remain in the excavation, a more detailed equilibrium calculation will be performed.

## 3.0 Conclusions:

The methodology for determining a basis for performing radiological assessments of excavations bounded by various combinations of bedrock, concrete and/or contaminated soil using an inventory-based approach has been provided. This method provides for determining a conservative estimate of potential future groundwater concentration from bedrock and/or subsurface concrete surface activity, with any soil that may remain in the excavation above the bedrock. Also provided is the general approach for determining the volumetric concentration from the area concentration.

## 4.0 References

4.1 Letter from CY to NRC re: LTP Amendment Dated December 1, 2004, CY Letter CY-04-131

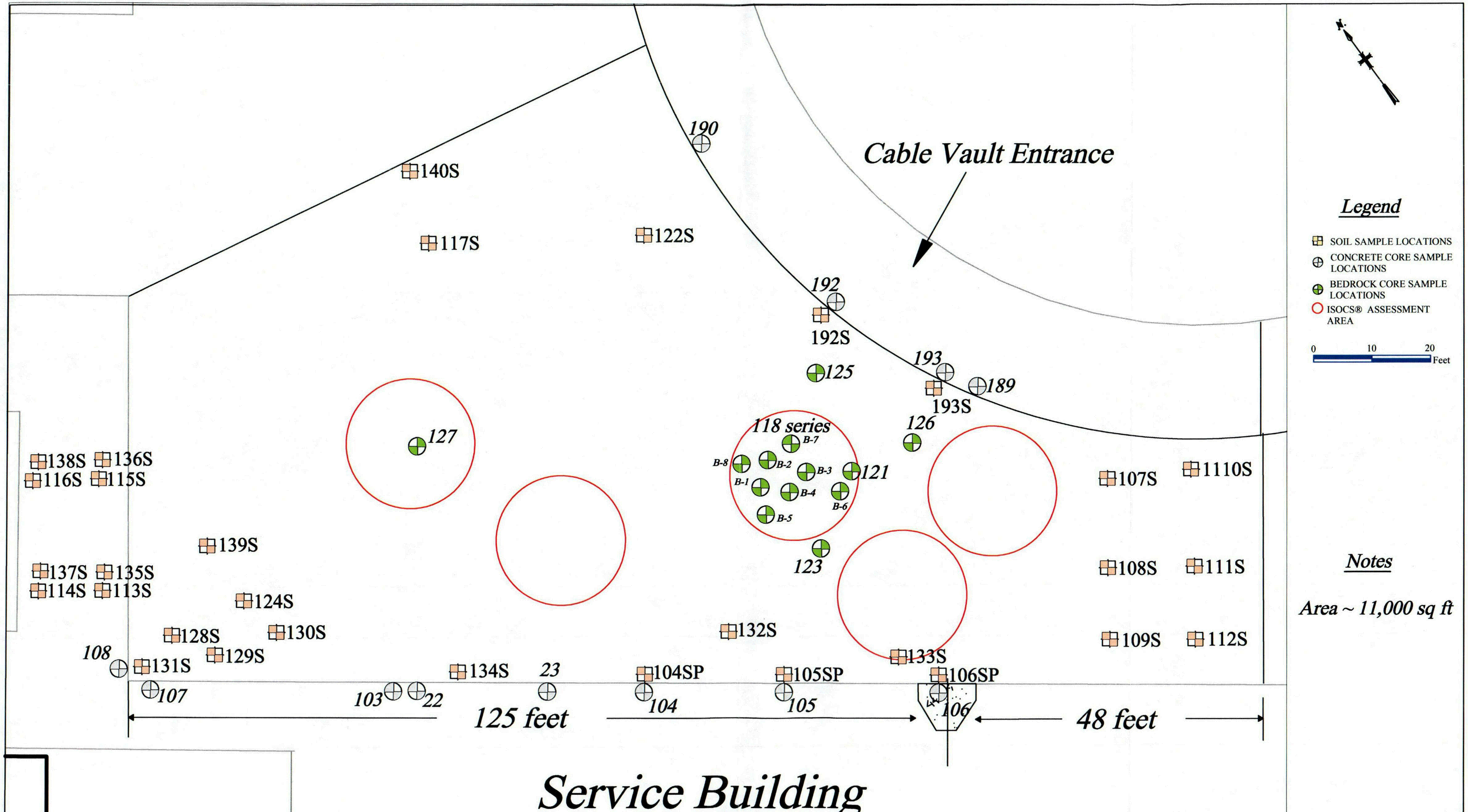
4.2 Technical Support Document CY- HP - 0185, Rev. 0, Kd Values of Backfill Material for Connecticut Yankee; Mark Fuhrmann Environmental Sciences Dept. Brookhaven National Laboratory, November 9, 2004

4.3 Technical Support Document CY - HP - 0184, Rev. 0, Estimates for Release of Radionuclides from Potentially Contaminated Concrete At the Haddam Neck Nuclear Plant; Terry Sullivan, Brookhaven National Laboratory Environmental Waste Technology Center, October 2004.

# Attachment 5.6

## Sample and Measurement Locations

(2 pages including cover)




| Attachment 5.6 | Connecticut Yankee Atomic Power Company<br>Sample and Measurement Locations | Date<br>February 2005 | By<br>P Hollenbeck<br>J McCarthy |
|---|---|---|---|

C02

# Attachment 5.7

## ISOCS® Assessment of West Section of the Excavation Associated with the Northeast Protected Area

(8 pages including cover)

# ISOCS® Assessment of West Section of the Excavation Associated with the Northeast Protected Area

## 1. System Description

The ISOCS® system is a commercially available combination hardware/software system that is designed to perform gamma spectroscopy analysis of various media utilizing a geometric relationship to the detector. The Connecticut Yankee ISOCS® hardware is comprised of a 40% relative efficiency P-type coaxial germanium detector, (serial number 3953,) in a dual port MAC dewar, an analog Inspector and two Panasonic Toughbook notebook computers connected using Windows Remote Desktop software and Linksys wireless PC Cards (802.11g protocol). The software is the Genie 2000 Gamma Acquisition and Analysis version 3.0 and Geometry Composer version 4.1a.

Collimation is available for the ISOCS® system in two thicknesses, 25-mm and 50-mm. The collimation used for this assessment was 50-mm thick stainless-steel encased lead that provides a choice of end apertures. This collimation effectively provides approximately two tenth-value layers of attenuation. Available apertures are 0°, 30°, 90°, and 180°. The collimation ratings correspond to the available, unobstructed field of view (FOV) for the detector when that collimation is in place. For example, the 50-mm thick, 0° collimator has no end opening, thus totally encasing the detector in 50-mm of shielding whereas the 90° collimator has an end opening that allows the detector face an unobstructed conical field of view that totals 90°.

The ISOCS® detector undergoes a process known as detector characterization at the Canberra facility. This process uses a Monte Carlo modeling code to generate a characterization file which contains the detector point source efficiency data and validation data. The detector efficiency is plotted and evaluated for various gamma energies and circular source diameters. Numerous counts are then performed to validate the Monte Carlo derived efficiency values which are evaluated for accuracy. This characterization data is loaded into the local computer system for use in the field. The characterization data encompasses the standard Canberra collimation equipment.

A software component of the ISOCS® system, 'Geometry Composer', combines the detector characterization with parameters of individual measurements taken with the ISOCS® system to provide both qualitative and quantitative results for that measurement.

A detailed explanation of the setup and operation of the ISOCS® system is available in procedures RPM 5.2-4 'Calibration of the ISOCS' and RPM 5.2-3 'Operation of the Canberra Portable Gamma Spectroscopy System'.

## 2. Data Collection and Analysis Method

The Geometry Composer software works by entering the physical dimensions of the count, the orientation of the detector (including the attitude, distance, and collimation), and data for the media such as contamination depth, material density and absorbers. Geometry Composer then calculates and creates an efficiency file based upon the detector-specific parameter file generated via the characterization process. This efficiency is then applied to the in-Situ count data which provides both comprehensive results as the final output. The results are valid for the area specified during the geometry input. The output results are in activity per unit or in gross activity, depending on settings chosen during creation of the calibration file.

## 3. ISOCS® Measurements and Core Locations

On the 11th and 13th of January 2005, numerous data points were collected using the ISOCS® system and 3-inch core boring equipment for the purpose of establishing a correlation between the two analysis methods. The primary data collection revolved around one area of the excavation with exposed bedrock and little or no soil within the defined FOV for the ISOCS®. To determine optimum survey coverage for the project, several measurements with varying distances and collimators were performed throughout the excavation. A summary of these measurements is provided in Table 1. The radius of the FOV for the measurements is determined by the formula:

$$radius = h * \frac{c}{90}$$

Where:

h = height of the detector
c = collimation in degrees

This formula yields a conservative detection radius that is simple to determine, yet maintains accuracy for analysis purposes. Generally, the true FOV is slightly larger and dependent upon the actual count parameters.

**Table 1**
**ISOCS® Measurement Location Summary**

| Measurement ID | Date | Location | Distance | Collimation |
|---|---|---|---|---|
| 06-00121 | 1/11/2005 | Excavation Center | 2-m | 0° |
| 06-00122 | 1/11/2005 | Excavation Center | 2-m | 90° |
| 06-00123 | 1/11/2005 | Excavation Center | 3-m | 90° |
| 06-00124 | 1/11/2005 | Excavation Center | 5.13-m | 90° |
| 06-00125 | 1/13/2005 | Excavation Center | 4-m | 0° |
| 06-00126 | 1/13/2005 | North | 50-cm | 90° |
| 06-00127 | 1/13/2005 | North | 3-m | 90° |
| 06-00128 | 1/13/2005 | West | 3-m | 90° |
| 06-00129 | 1/13/2005 | South | 3-m | 90° |

# ISOCS® Assessment of West Section of the Excavation Associated with the Northeast Protected Area

**Table 1**
**ISOCS® Measurement Location Summary**

| Measurement ID | Date | Location | Distance | Collimation |
|---|---|---|---|---|
| 06-00130 | 1/13/2005 | Southeast | 3-m | 90° |
| 06-00131 | 1/13/2005 | Excavation Center | 3-m | 180° |

Eight core locations were selected within the FOV of the primary ISOCS location (Figure 1). Each core was taken to a full bit depth (approximately 25-cm) and then was subsequently sliced into 2-cm thicknesses. The top three slices from each core location was sent for volumetric analysis which included, at a minimum, naturally-occurring radioactive material (NORM), and the plant-derived gamma emitting nuclides Co-60 and Cs-137. These two nuclides represent the primary plant-derived isotopes that are detectable by in-Situ gamma spectroscopy. Additionally, the core samples were analyzed for tritium and Sr-90.

## 4. Comparison of Samples

The goal in comparing volumetric sample data with ISOCS® count data measurements is to show that similar results are obtained via both analytical methods. This ensures that no significant residual contamination is left unaccounted. To accomplish this comparison, ISOCS® geometries were created that mimicked the depth of the core samples (approximately 2-cm). These geometries were applied to the ISOCS® measurements taken in the excavation. The output results, in pCi per square meter (pCi/m$^2$), were compared to the volumetric results, in pCi per gram (pCi/g), that were obtained from the off-site counting laboratory. Of the three ISOCS® measurements taken over the eight core locations, only one showed plant-derived activity greater than the measurement-specific minimum detectable activity (MDA). Measurement ID 06-00124 returned results of 27,875 pCi/m$^2$ of Co-60 and 21,539 pCi/m$^2$ of Cs-137. All eight of the core locations had results less than the measurement-specific MDA. Measurement ID 06-00124 was located where a portion of its' FOV comprised a bank of soil known to contain plant-derived activity. Absence of activity in the core samples and other ISOCS® measurements in the same location indicate that the activity detected in 06-00124 is exclusively in the bank of soil.

## 5. Conclusion

The plant-related residual gamma radioactivity within the excavation is less than the measurement specific MDA across the exposed bedrock for the ISOCS® and the volumetric analyses. Tables 2 and 3 provide summary analytical results for the ISOCS® measurements and core sample volumetrics respectively.

# ISOCS® Assessment of West Section of the Excavation Associated with the Northeast Protected Area

**Table 2 NISOCS® Results Summary**

| ISOCS Measurement ID | Co-60 Result pCi/m$^2$ | Co-60 Error pCi/m$^2$ | Co-60 MDA pCi/m$^2$ | Cs-134 Result pCi/m2 | Cs-134 Error pCi/m$^2$ | Cs-134 MDA pCi/m$^2$ | Cs-137 Result pCi/m$^2$ | Cs-137 Error pCi/m$^2$ | Cs-137 MDA pCi/m$^2$ |
|---|---|---|---|---|---|---|---|---|---|
| 06-00122 | | | 20300 | | | 24500 | | | 25400 |
| 06-00123 | | | 20000 | | | 24300 | | | 24900 |
| 06-00124 | 27875 | 7804 | 18200 | | | 27400 | 21539 | 8659 | 21300 |
| 06-00126 | | | 16300 | | | 20600 | | | 21100 |
| 06-00127 | | | 20500 | | | 23900 | 12937 | 6131 | 13600 |
| 06-00128 | | | 20700 | | | 25500 | 28270 | 9102 | 19500 |
| 06-00129 | 16456 | 7046 | 18900 | | | 28500 | 38403 | 10665 | 18400 |
| 06-00130 | | | 20300 | | | 21000 | | | 21000 |
| 06-00131 | | | 12900 | | | 14800 | 29153 | 6716 | 11500 |
| Average | 22165 | 7425 | 18678 | N/A | N/A | 23389 | 26060 | 8255 | 19633 |
| Standard Deviation | 8075 | 536 | 2592 | N/A | N/A | 4127 | 9480 | 1842 | 4635 |

# ISOCS® Assessment of West Section of the Excavation Associated with the Northeast Protected Area

**Table 3 - Core Volumetric Results Summary**

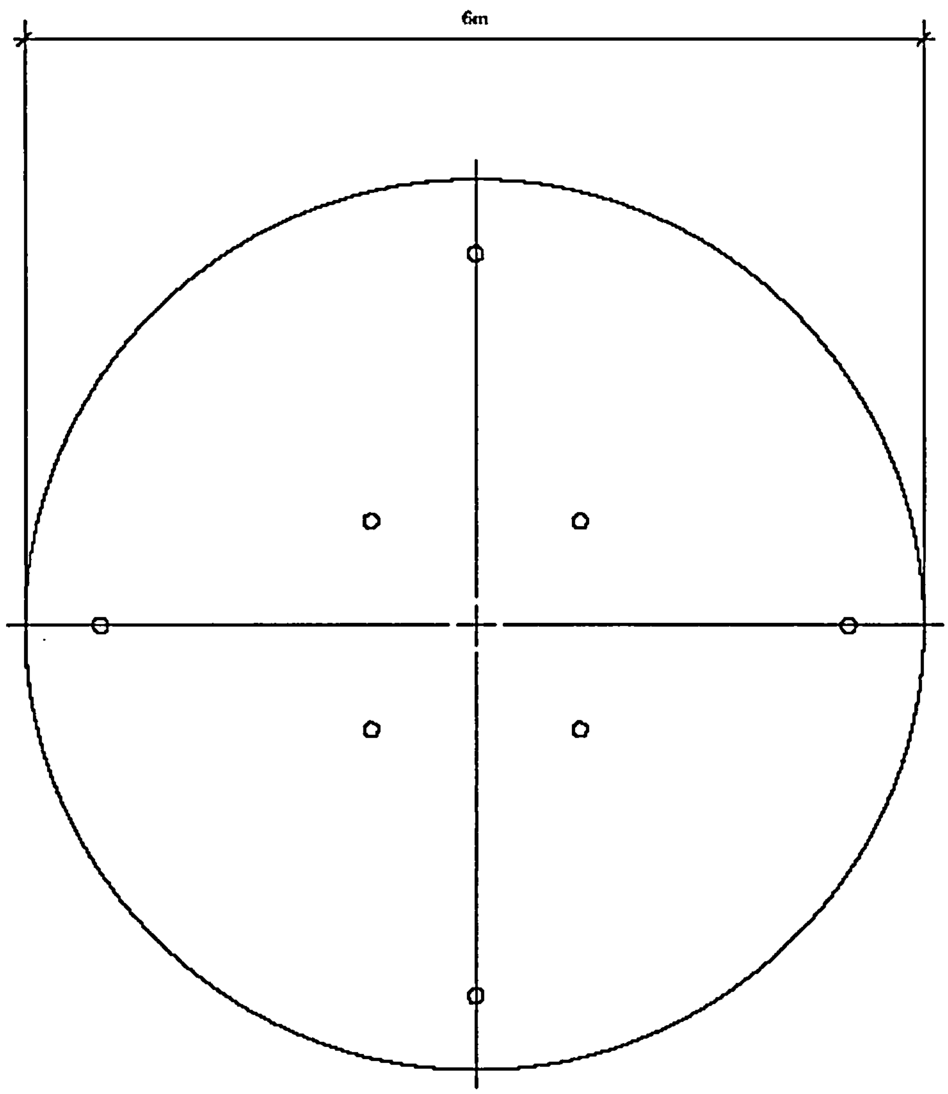
| Sample Designation | Co-60 Result pCi/g | Co-60 Error pCi/g | Co-60 MDA pCi/g | Cs-134 Result pCi/g | Cs-134 Error pCi/g | Cs-134 MDA pCi/g | Cs-137 Result pCi/g | Cs-137 Error pCi/g | Cs-137 MDA pCi/g | Sr-90 Result pCi/g | Sr-90 Error pCi/g | Sr-90 MDA pCi/g | Tritium Result pCi/g | Tritium Error pCi/g | Tritium MDA pCi/g |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 9801-0000-118-B1-01 | | | 0.0365 | | | 0.0342 | | | 0.0265 | 0.0868 | 0.0218 | 0.0238 | | | 7.20 |
| 9801-0000-118-B1-02 | | | 0.0390 | | | 0.0330 | | | 0.0250 | 0.0601 | 0.0181 | 0.0224 | | | 7.28 |
| 9801-0000-118-B1-03 | | | 0.0324 | | | 0.0294 | | | 0.0303 | 0.0978 | 0.0239 | 0.0231 | | | 7.23 |
| 9801-0000-118-B2-01 | | | 0.0310 | | | 0.0323 | | | 0.0251 | 0.0539 | 0.0229 | 0.0364 | | | 7.46 |
| 9801-0000-118-B2-02 | | | 0.0443 | | | 0.0407 | | | 0.0371 | 0.1240 | 0.0255 | 0.0197 | | | 7.15 |
| 9801-0000-118-B2-03 | | | 0.0213 | | | 0.0365 | | | 0.0249 | 0.0633 | 0.0201 | 0.0229 | | | 7.20 |
| 9801-0000-118-B3-01 | | | 0.0326 | | | 0.0373 | | | 0.0333 | 0.0739 | 0.0253 | 0.0352 | | | 7.12 |
| 9801-0000-118-B3-02 | | | 0.0340 | | | 0.0269 | | | 0.0276 | 0.0747 | 0.0224 | 0.0260 | | | 7.01 |
| 9801-0000-118-B3-03 | | | 0.0336 | | | 0.0303 | | | 0.0231 | 0.0455 | 0.0214 | 0.0318 | | | 7.27 |
| 9801-0000-118-B4-01 | | | 0.0375 | | | 0.0345 | | | 0.0316 | 0.1240 | 0.0285 | 0.0277 | | | 7.13 |
| 9801-0000-118-B4-02 | | | 0.0343 | | | 0.0302 | | | 0.0227 | 0.0433 | 0.0198 | 0.0286 | | | 7.16 |
| 9801-0000-118-B4-03 | | | 0.0248 | | | 0.0327 | | | 0.0265 | 0.0411 | 0.0221 | 0.0355 | | | 7.06 |
| 9801-0000-118-B5-01 | | | 0.0205 | | | 0.0270 | | | 0.0259 | 0.0495 | 0.0216 | 0.0321 | | | 7.21 |
| 9801-0000-118-B5-02 | | | 0.0337 | | | 0.0262 | | | 0.0278 | 0.0926 | 0.0262 | 0.0288 | | | 7.26 |
| 9801-0000-118-B5-03 | | | 0.0340 | | | 0.0325 | | | 0.0331 | 0.0502 | 0.0209 | 0.0302 | | | 7.20 |
| 9801-0000-118-B6-01 | | | 0.0253 | | | 0.0243 | | | 0.0211 | 0.1080 | 0.0272 | 0.0281 | | | 7.07 |
| 9801-0000-118-B6-02 | | | 0.0412 | | | 0.0458 | | | 0.0329 | 0.0864 | 0.0246 | 0.0283 | | | 7.06 |
| 9801-0000-118-B6-03 | | | 0.0293 | | | 0.0406 | | | 0.0293 | 0.0643 | 0.0219 | 0.0274 | | | 7.23 |
| 9801-0000-118-B7-01 | | | 0.0324 | | | 0.0291 | | | 0.0261 | 0.0298 | 0.0151 | 0.0239 | | | 9.89 |
| 9801-0000-118-B7-02 | | | 0.0341 | | | 0.0360 | | | 0.0304 | 0.0620 | 0.0178 | 0.0207 | | | 8.63 |
| 9801-0000-118-B7-03 | | | 0.0326 | | | 0.0317 | | | 0.0299 | 0.0684 | 0.0190 | 0.0234 | | | 9.82 |
| 9801-0000-118-B8-01 | | | 0.0378 | | | 0.0442 | | | 0.0295 | 0.0667 | 0.0185 | 0.0208 | | | 9.25 |
| 9801-0000-118-B8-02 | | | 0.0260 | | | 0.0300 | | | 0.0231 | 0.0577 | 0.0211 | 0.0320 | | | 8.85 |
| 9801-0000-118-B8-03 | | | 0.0247 | | | 0.0298 | | | 0.0309 | 0.0424 | 0.0145 | 0.0172 | | | 8.56 |
| 9801-0000-125-B-01 | | | 0.0311 | | | 0.0311 | | | 0.0257 | 0.0652 | 0.0181 | 0.0200 | | | 9.05 |
| 9801-0000-125-B-02 | | | 0.0517 | | | 0.0492 | | | 0.0800 | 0.0484 | 0.0173 | 0.0251 | | | 8.00 |
| 9801-0000-125-B-03 | | | 0.0382 | | | 0.0342 | | | 0.0359 | 0.0419 | 0.0153 | 0.0205 | | | 8.88 |
| 9801-0000-126-B-01 | | | 0.0446 | | | 0.0295 | | | 0.0297 | 0.0493 | 0.0163 | 0.0212 | | | 8.75 |

# ISOCS® Assessment of West Section of the Excavation Associated with the Northeast Protected Area

| Sample Designation | Co-60 Result pCi/g | Co-60 Error pCi/g | Co-60 MDA pCi/g | Cs-134 Result pCi/g | Cs-134 Error pCi/g | Cs-134 MDA pCi/g | Cs-137 Result pCi/g | Cs-137 Error pCi/g | Cs-137 MDA pCi/g | Sr-90 Result pCi/g | Sr-90 Error pCi/g | Sr-90 MDA pCi/g | Tritium Result pCi/g | Tritium Error pCi/g | Tritium MDA pCi/g |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 9801-0000-126-B-02 | | | 0.0334 | | | 0.0262 | | | 0.0288 | 0.0570 | 0.0157 | 0.0181 | | | 9.50 |
| 9801-0000-126-B-03 | | | 0.0263 | | | 0.0313 | | | 0.0213 | 0.0908 | 0.0300 | 0.0383 | | | 9.27 |
| 9801-0000-127-B-01 | | | 0.0383 | | | 0.0290 | | | 0.0261 | 0.0515 | 0.0163 | 0.0184 | | | 8.90 |
| 9801-0000-127-B-02 | | | 0.0303 | | | 0.0294 | | | 0.0342 | 0.0474 | 0.0164 | 0.0205 | | | 9.14 |
| 9801-0000-127-B-03 | | | 0.0332 | | | 0.0318 | | | 0.0260 | 0.0584 | 0.0207 | 0.0300 | | | 9.01 |
| Average: | N/A | N/A | 0.0333 | N/A | N/A | 0.0329 | N/A | N/A | 0.0297 | 0.0659 | 0.0208 | 0.0260 | N/A | N/A | 8.02 |
| Standard Deviation: | N/A | N/A | 0.0067 | N/A | N/A | 0.0058 | N/A | N/A | 0.0099 | 0.0237 | 0.0041 | 0.0058 | N/A | N/A | 0.99 |

## ISOCS® Assessment of West Section of the Excavation Associated with the Northeast Protected Area

## Core Locations Relative to the ISOCS® Field of View

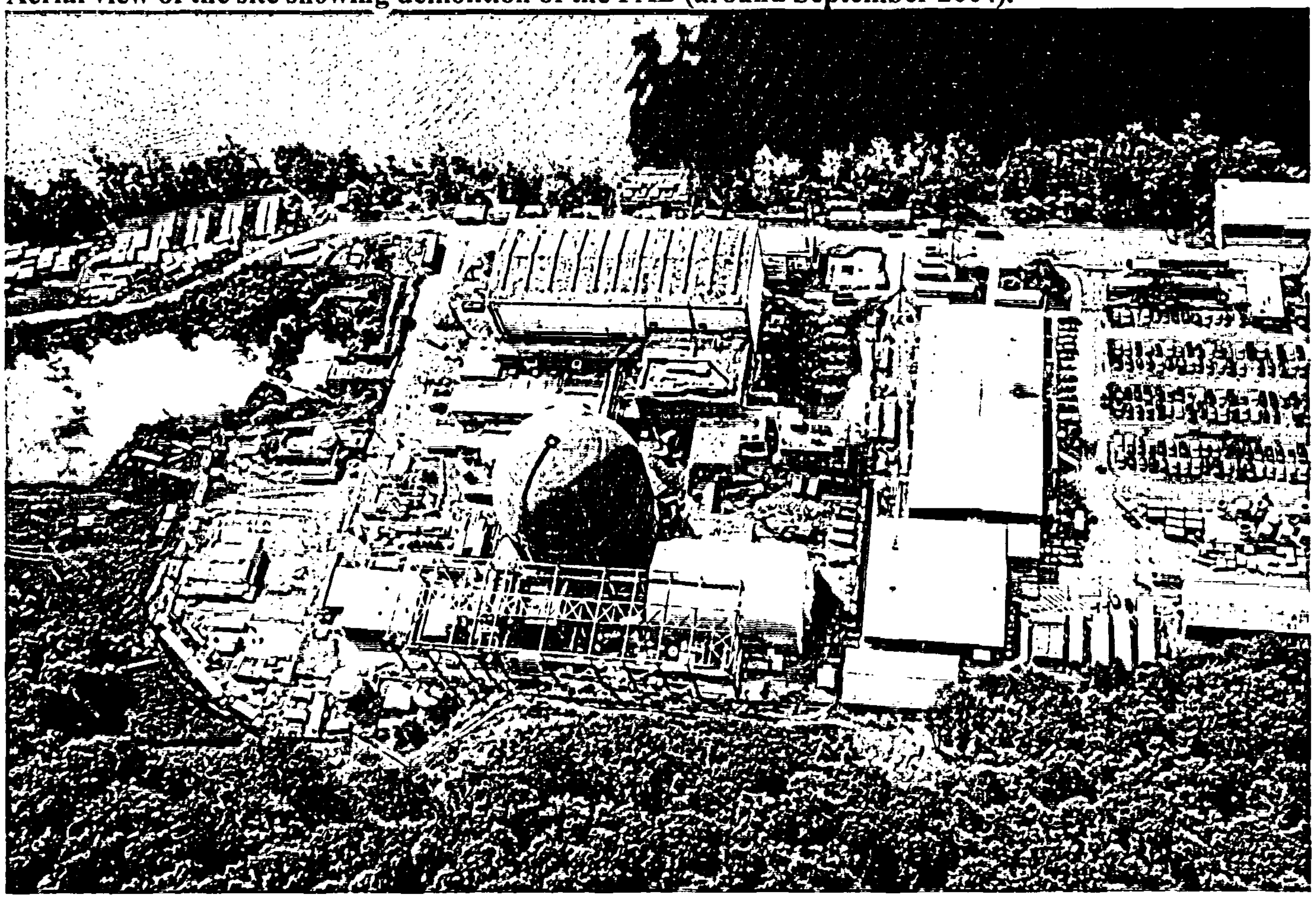


# Attachment 5.8

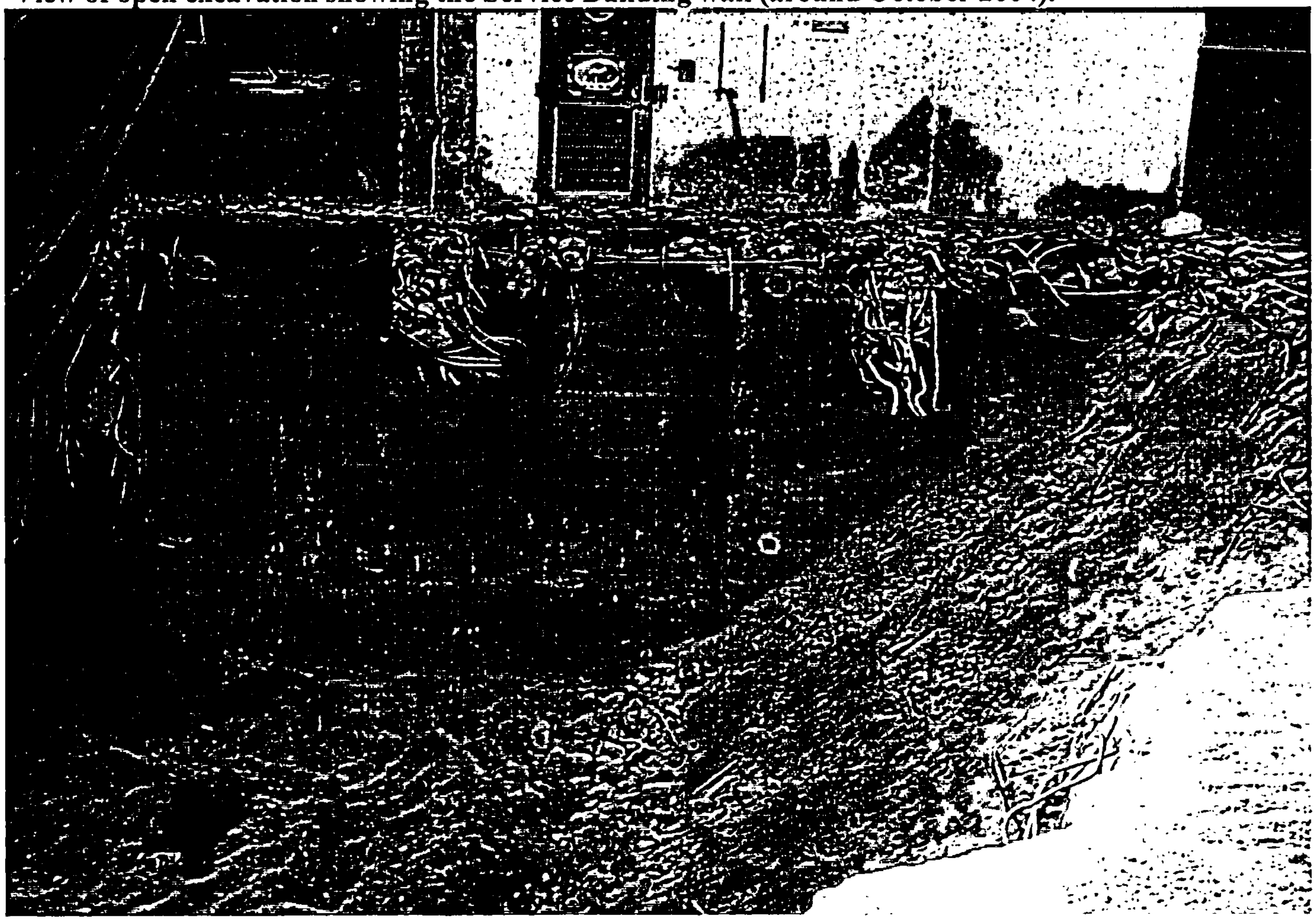
## Photographs of Project and Survey and Sampling

(17 pages including cover)

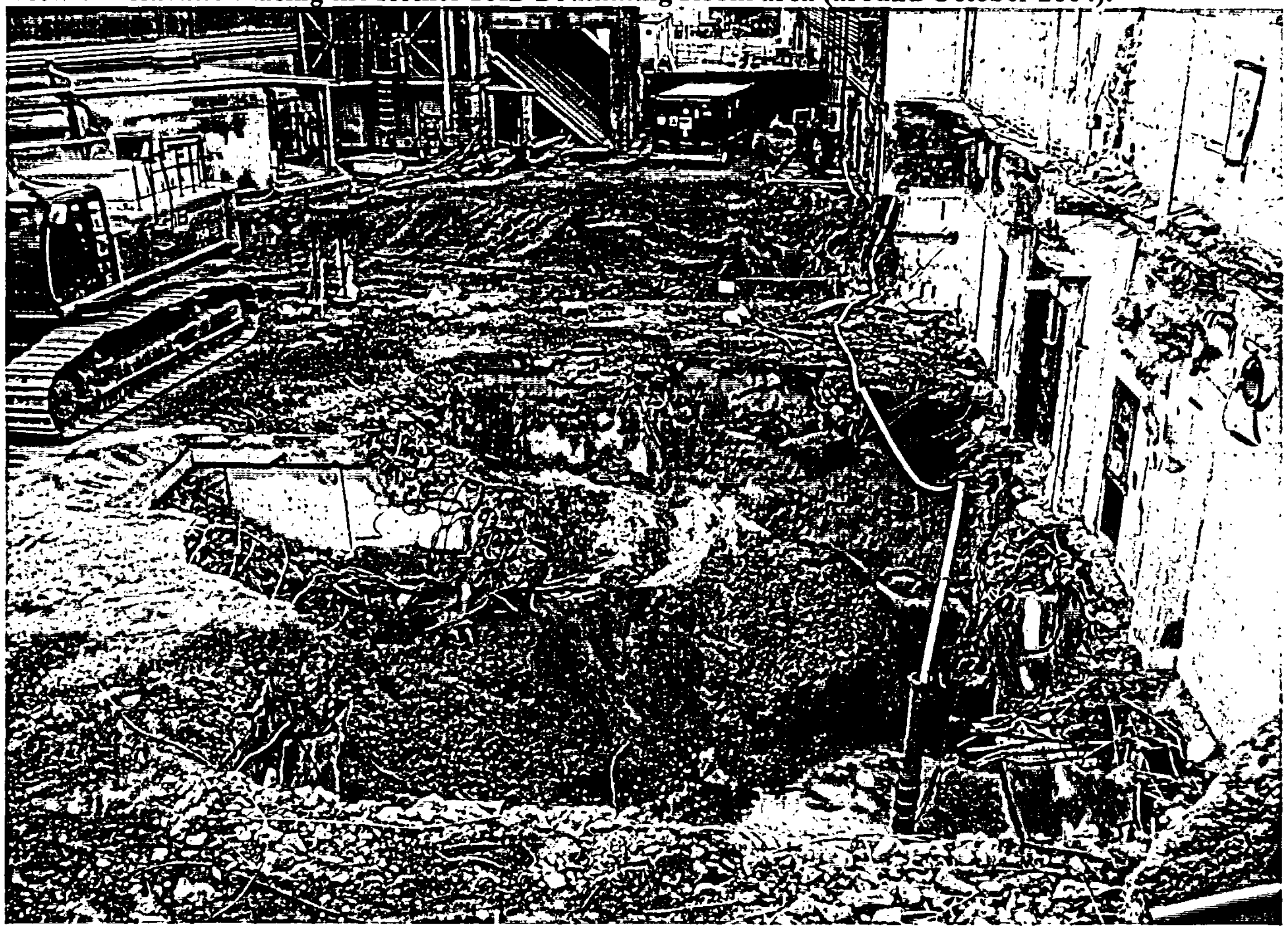
Aerial view of the site showing demolition of the PAB (around September 2004).



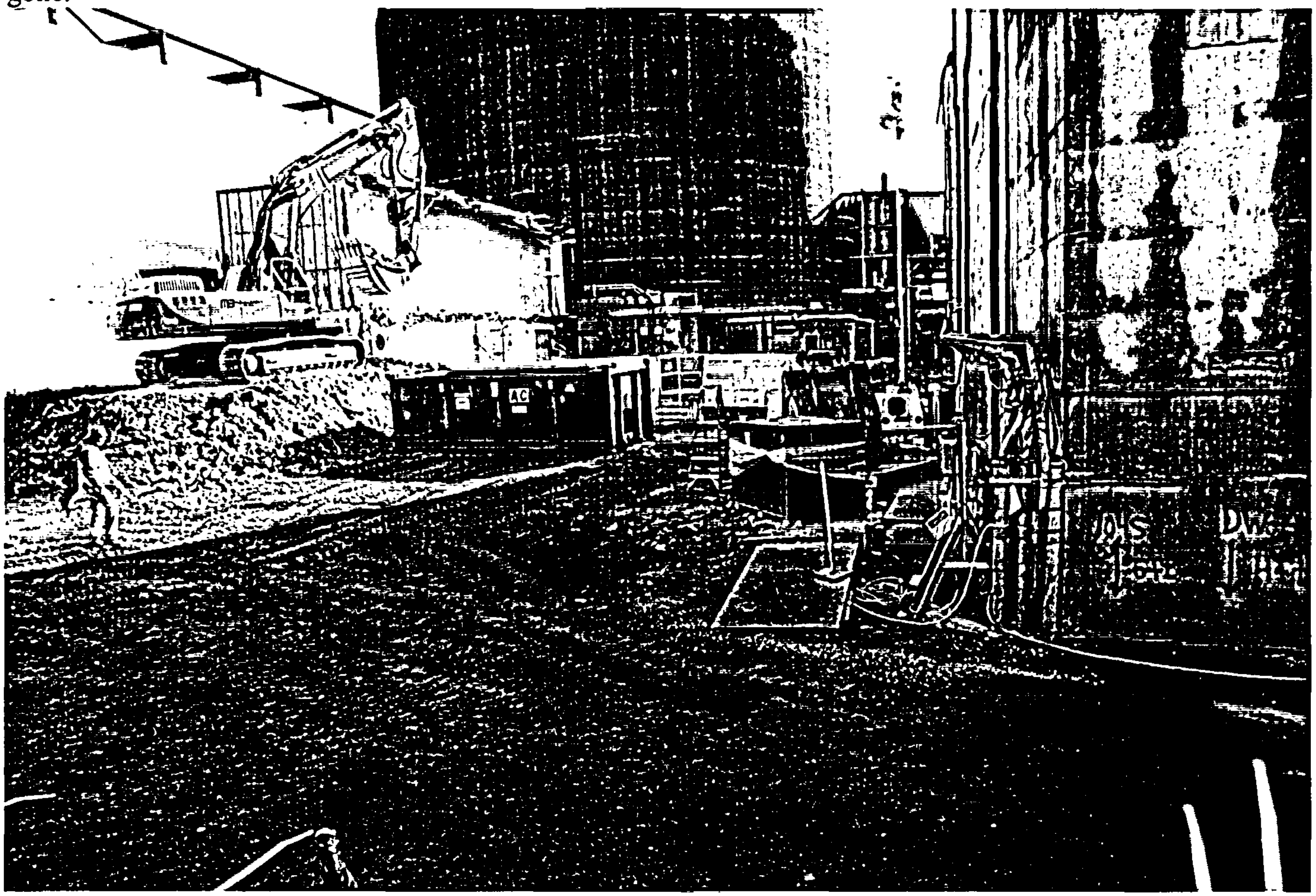
View of open excavation showing the Service Building wall (around October 2004).



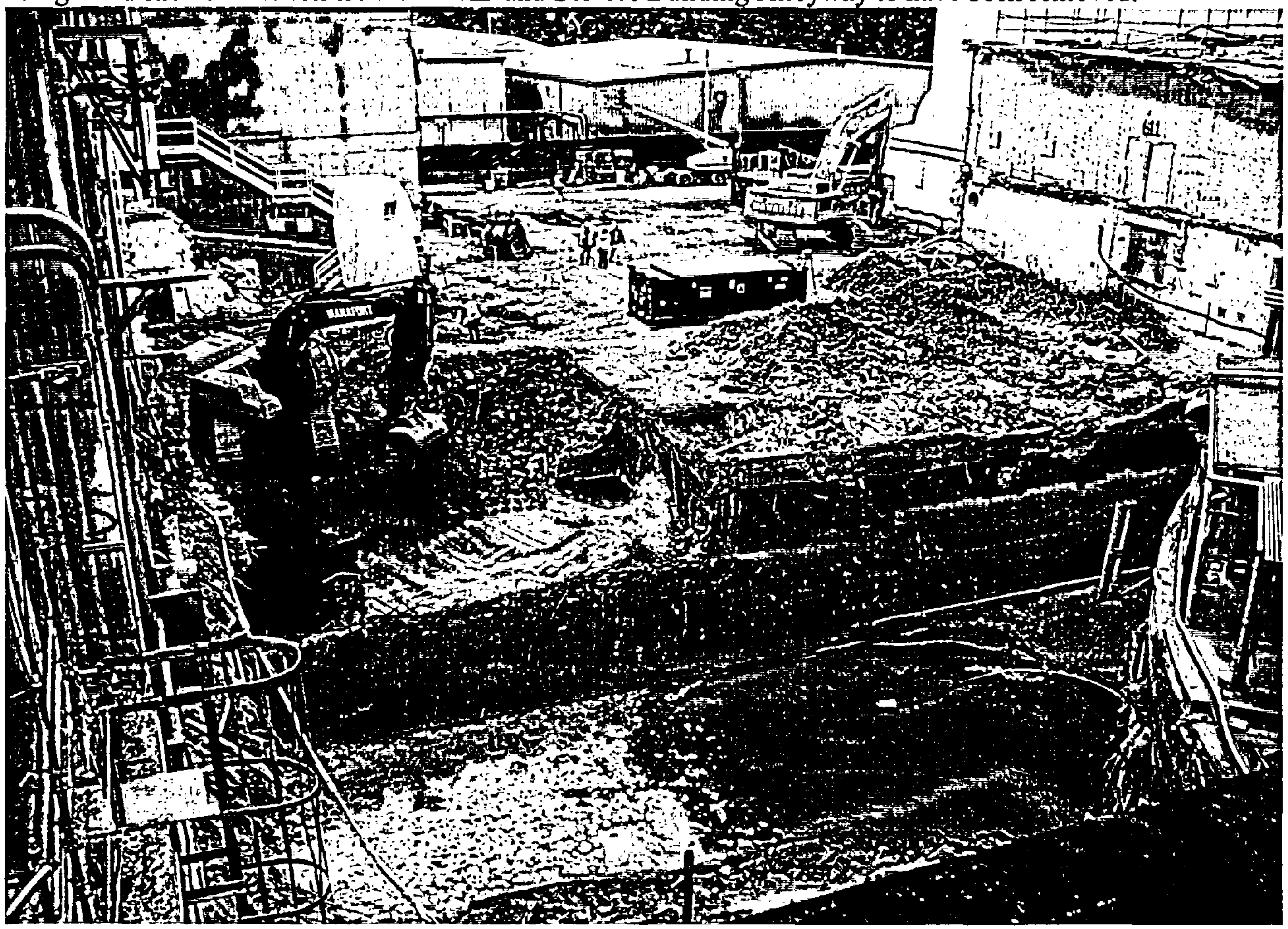
View of excavation facing the former PAB Drumming Room area (around October 2004).



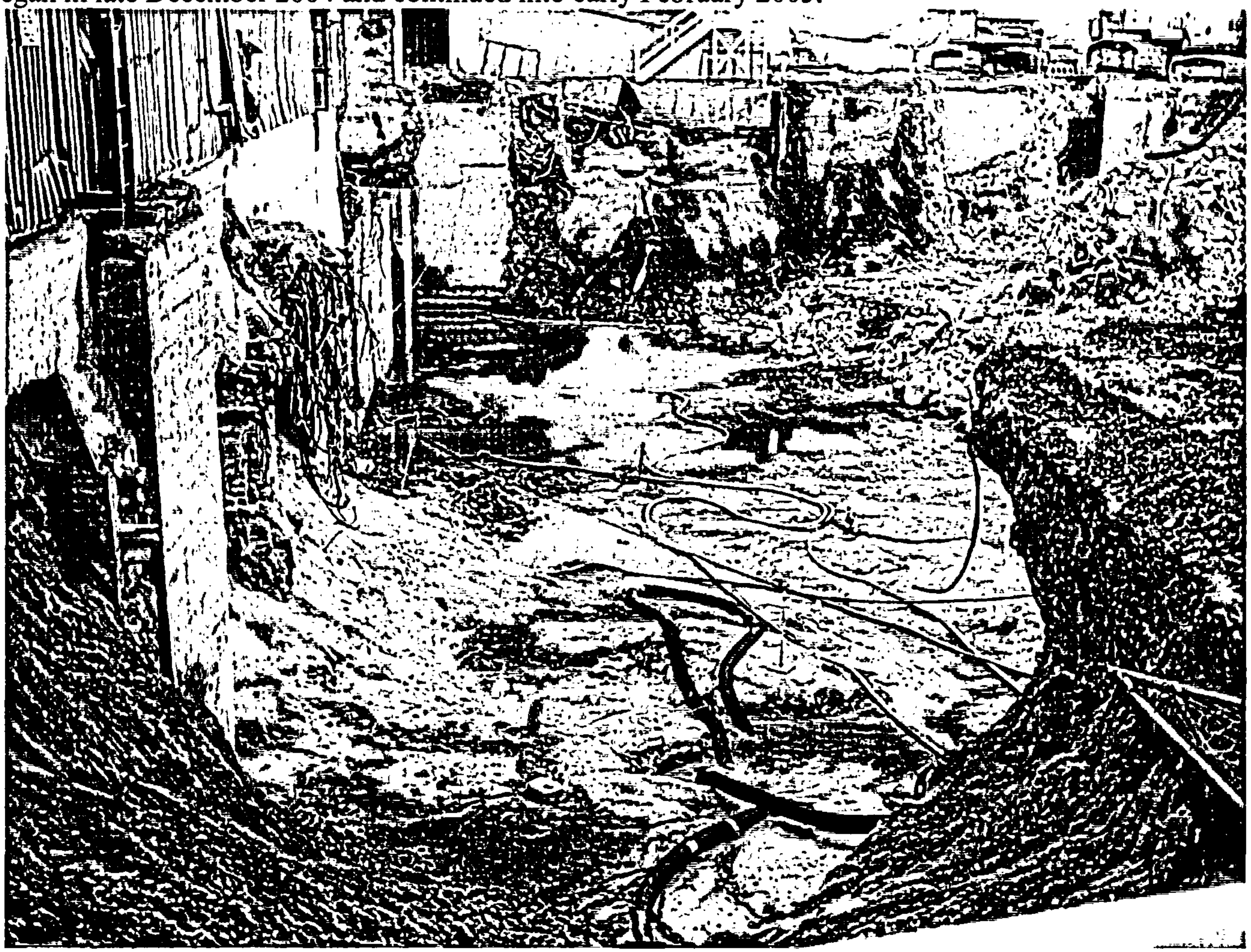
**View of PAB demolition from the northwest (around November 2004).** The PAB superstructure is mostly gone.



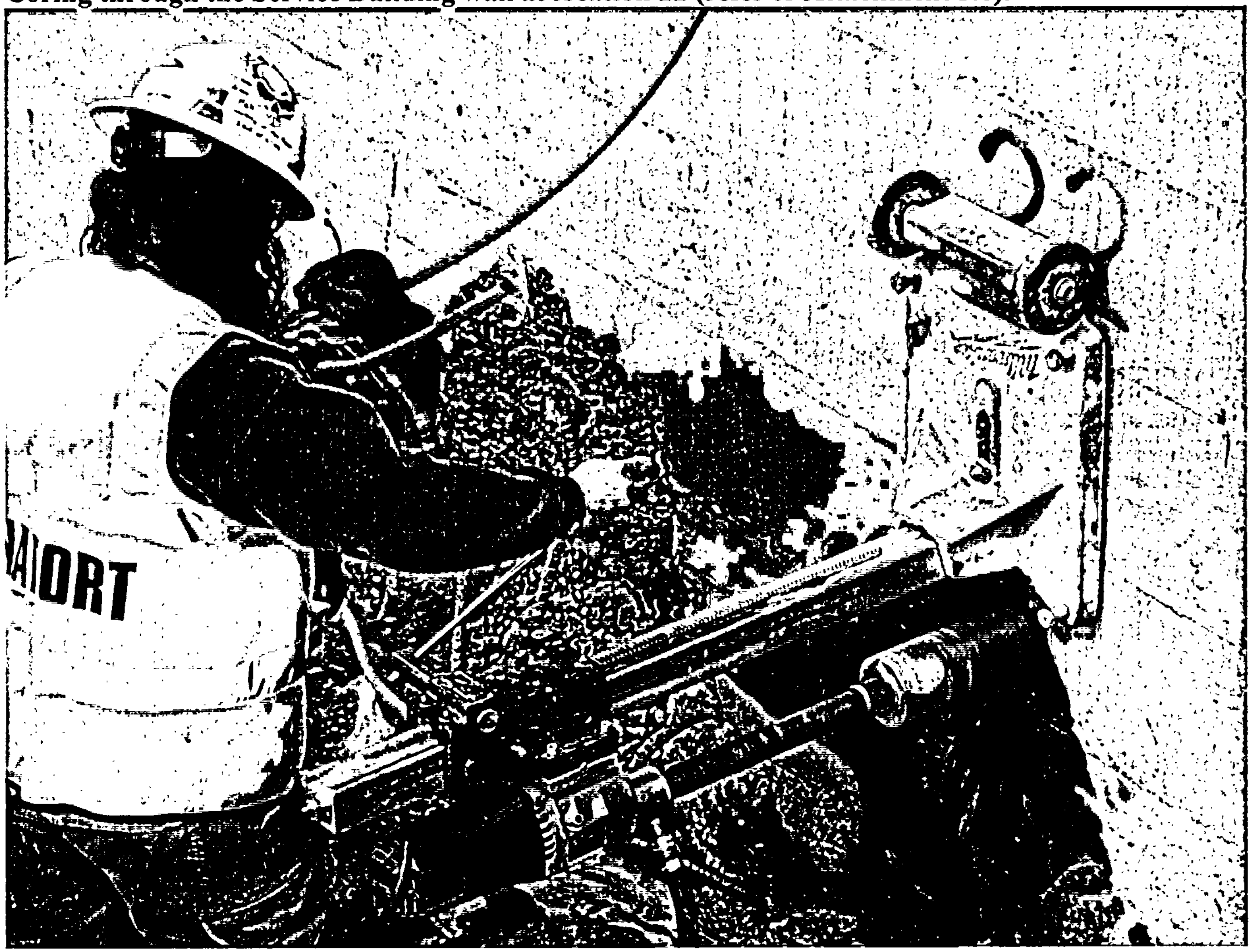
**View of excavation from the south looking at the PAB outer wall (Around November 2004).** Open area in foreground shows most soil from the PAB and Service Building Alleyway to have been removed.



**Extent of excavation in early December.** Removal of soil continued to bedrock. Characterization surveys began in late December 2004 and continued into early February 2005.



Coring through the Service Building wall at location 22 (refer to Attachment 5.6).



**Inspection of exposed bedrock.** Hydro-geologists provided recommendations for bedrock assessment.

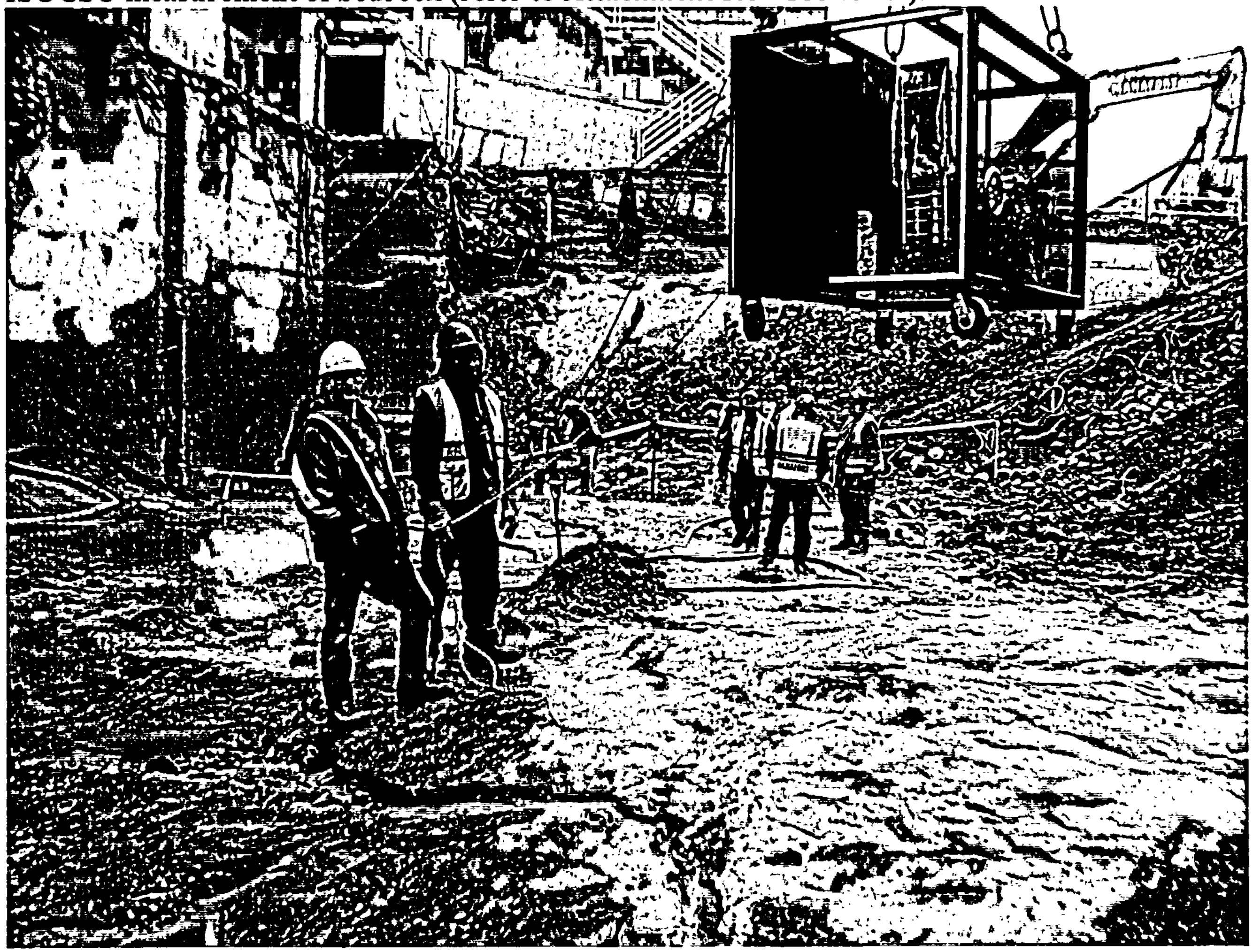
HP scanning Service Building wall with beta sensitive scintillation probes.

**View of Service Building and PAB Remnant wall.** Photograph shows locations of cores 107 and 108. Refer to Attachment 5.6).

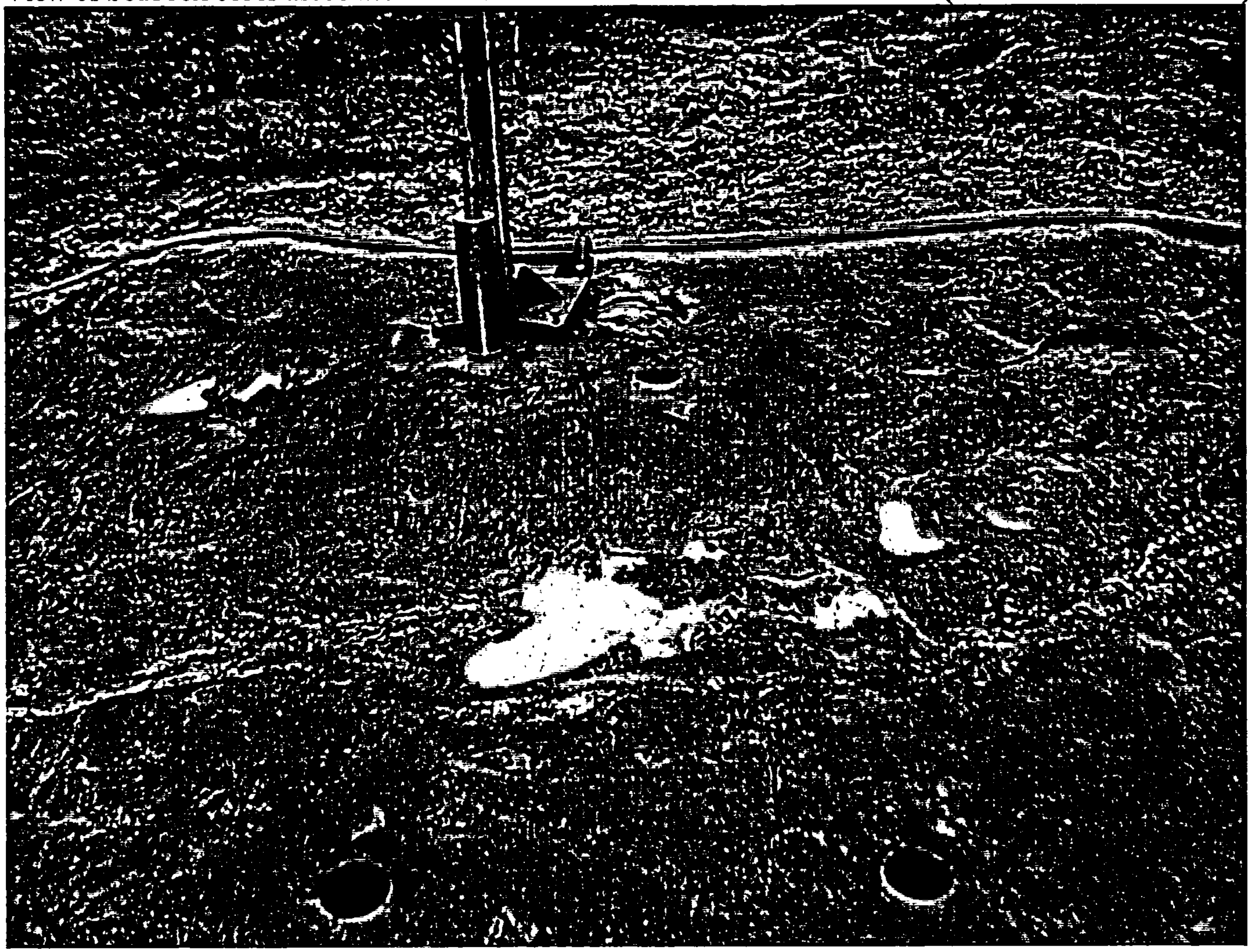
Core collection of exposed bedrock.

**Soil collection for radiological analysis.** Service Building wall concrete core location 105 can be seen in the background (refer to Attachment 5.6).

ISOCS® measurement of bedrock (refer to Attachment 5.6 - 118 series).



View of bedrock cores associated with the ISOCS® measurement 118 series (refer to Attachment 5.6).



View of excavation during early February.

Current view of excavation as of February 17, 2005.