MAINE YANKEE

LTP SECTION 2

SITE CHARACTERIZATION

TABLE OF CONTENTS

List of Tables

2.0 SITE CHARACTERIZATION

2.1 Overview

The radiological and chemical characterization of the Maine Yankee (MY) site has been going on since pre-operational sampling was begun in 1970. Initial site characterization for decommissioning was begun in the fall of 1997 and ran through the spring of 1998. Historical information, including the 10 CFR 50.75(g) file, employee interviews, Radiological Incident files, pre-operational survey data, spill reports, special surveys (e.g., site aerial surveys, marine fauna and sediment surveys), operational survey records and Annual Radiological Environmental Reports (including sampling of air, groundwater, estuary water, milk, invertebrates, fish and surface vegetation) to the NRC were reviewed and compiled into the Historical Site Assessment (HSA). Using the information collected during the HSA, a characterization plan was developed to collect measurements and samples from plant structures, systems and open land areas to cover the areas where contamination existed, remained or had the potential to exist.

The information collected during site characterization, including the HSA, was used during decommissioning planning to achieve the following objectives:

- Determine the radiological status of the site and facility to include identification of systems, structures, soils and water sources in which contamination exists;
- Identify the location and extent of any contamination outside the radiological restricted areas (RA);
- Estimate the source term and radionuclide mixture to support decommissioning cost estimation and decision-making for remediation, dismantlement and radioactive waste disposal activities;
- Select the instrumentation used for surveys and develop the quality assurance methods applied to sample collection and analysis;
- Perform dose assessment and FSS design; and
- Ensure the Radiation Protection Program addresses any unique radiological health and safety issues associated with decommissioning.

The site characterization process focused on four areas, providing both shutdown and current data for structures, systems, radiological environs and hazardous materials environs. The extent and range of contamination were reported for structures, systems, drains, vents, embedded piping, paved areas, water and soils. In addition, activation analyses were performed on key components within the restricted area to estimate radioactive waste volumes and classes.

The characterization results were provided to MY in the "Characterization Survey Report For The MY Atomic Power Plant." After review of the characterization report, it was determined that some additional sampling was needed to fully define the extent of contamination in some outdoor areas and some systems in order to design the FSS, perform dose assessments and address questions related to waste volumes. This additional sampling is discussed in Section 2.5.

This section summarizes the key findings of the HSA and characterization survey results, as supplemented by the continuing characterization program. The full characterization survey report, and the detailed results of the continuing characterization program, are maintained at the MY site and are available for NRC review. The level of detail provided in this summary demonstrates that the characterization plan objectives listed above have been met. In addition, the characterization data provided in this section are consistent with NRC guidance contained in Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," and sufficient to meet the review criteria set forth in NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans."

2.2 Historical Site Assessment

The Radiation Protection organization amassed tens of thousands of survey records documenting general area and component-specific radiation levels, contamination levels, system activity levels and airborne radioactivity levels during 25 years of plant operation. These survey records reflected radiological conditions on site with frequency and detail dependent on the magnitude of radiation and contamination present in an area and the frequency with which the area was entered by the operating staff. Plant document files contained records of spill and event reports (Operations Department Unusual Occurrence Reports and Radiation Protection Department Radiological Incident Reports) as well as the required annual or semiannual radiological effluent reports to the NRC which documented any unplanned releases.

In order to ensure a complete discovery of events involving spills, leaks or other operational occurrences which might have an effect on the radiological and chemical status of the site, MY also interviewed terminating employees for any recollection of such events.

This section of the Site Characterization Summary describes both the original and current radiological and chemical status of the site.

2.2.1 Historical Data Review

Historical records contained in the radiation protection files, 10 CFR 50.75(g) file, Annual Radiological Environmental Reports to the NRC, miscellaneous environmental reports, and one 10 CFR 20.302 submittal were reviewed to determine the location and extent of leaks and spills on site. The pertinent results of the record reviews, Initial Site Characterization surveys, and employee interviews were captured in the Historical Site Assessment (HSA). The HSA is a compilation of the approximately 120 potential events occurring over the 25 year operating history of the plant. About two thirds of these events were potential radiological issues with the other one third being chemical or hazardous material events.

Key items identified in the HSA include:

1. Contaminated soil between the RA and Forebay, from RWST leaks;
2. Contaminated soil after the removal of a low level waste storage area (Wiscasset wall);
3. Location of a silt spreading area/construction debris landfill;
4. A waste neutralization tank drain line leak;
5. A PCC leak in the alley way;
6. Contaminated soil on Bailey Point, south of the Industrial Area (IA) trailer park, in an area where contaminated soil from the PCC leak had been stored;
7. Discrete particles throughout plant from reactor core barrel machining;
8. Contaminated soil in the ISFSI area, formerly known as the contractor parking lot;
9. A discrete particle outside warehouse 2;
10. Contaminated sumps and floor trenches in the turbine hall;
11. RA sink and decon shower drains go to sewage treatment plant;
12. Contaminated sediment in the Forebay;
13. Previous abandonment of an underground ferrous sulfide tank;
14. Snow from RA placed in ball field;
15. Contaminated soil from BWST leaks;
16. Contaminated soils in the IA trailer park; and
17. Very low levels of detectable residual radioactivity on Foxbird Island, RCA building roof, Equipment Hatch pit, and on the concrete block in the ball field dugouts.

None of the event records in the HSA indicated the uncontrolled release of radioactive material affecting the site beyond Bailey Point (i.e., south of Ferry Road and east of Bailey Cove).

2.2.2 Decommissioning File 10 CFR 50.75(g)

Even though MY was in operation well before the requirement to maintain a decommissioning file, the 50.75(g) file contained documentation of three areas of soil contamination and one record of a 10 CFR 20.302 submittal for burial in place of residual soil activity. The information in the decommissioning file was added to the HSA so that the affected areas could be properly addressed during site characterization.

The 50.75(g) file documented soils outside the Spray, Containment and Fuel Buildings (see Table 2-1) that were known to contain contamination from an RWST manway leak, a series of RWST siphon heater leaks, SCC/PCC leaks, as well as the storage of radioactive waste awaiting shipment in an outside, shielded storage location. Some work was also performed on contaminated components within tented enclosures located outside the RCA Storage Building which also contributed to soil and pavement contamination.

Table 2-1
Significant Soil Contamination Events

Event	Date	Location	Volume	Disposition
RWST siphon heater leak	2/23/88	Area south and west of RWST	8200 ft^3	Remediated 600 ft^3. 7600 ft^3 left in place under 10 CFR 20.302.[1]
Removal of Low Level Waste Storage Area	7/92	Outside the RCA Storage Bldg and west to high rad bunker	2000 ft^3	Residual contamination evaluated and entered into 50.75(g) file.
Silt spreading area	1992, 1993 Outages	Land adjacent to and south of ballfield.	1250 ft^3	Residual contamination evaluated and entered into 50.75(g) file.

[1] 10 CFR 20.302 has been superceded by 10 CFR 20.2002

2.2.3 10 CFR 20.302 Submittal (reference Table 2-1 above)

MY applied to the NRC on 11/2/88 (MN 88-107) to allow residual soil contamination to remain in place under the provisions of 10 CFR 20.302. The NRC approved the submittal on 8/31/89. This data is included to provide a complete historical basis for the Site Characterization. The details of the soil contamination are presented below.

In 1988 a small outdoor leak at the inlet flange connection between the RWST siphon heater return line and an isolation valve was discovered and subsequently contained. The actual time that the leak started and the volume of water lost could not be determined. Surveys of the area adjacent to the RWST indicated ground contamination as high as 7E-3 µCi/g of Cs-137.

The leak was repaired, and the contaminated soil was removed from the area and disposed of as radioactive waste. Sample analysis of the soil removed from the area of remediation also indicated the presence of Cs-134, Sb-125 and Co-60 in addition to the Cs-137. The level of activity of these additional nuclides was approximately two orders of magnitude less than the Cs-137. Soil was excavated to a level of two to five feet below grade until the average residual Cs-137 activity had decreased to an equivalent MPC value in water of about 2E-5 µCi/ml.

Approximately 600 cubic feet of radioactive waste was generated from the excavation. Residual activity Cs-137 in an estimated 7600 cubic feet of remaining affected soil was 6 mCi. The location of this contaminated soil is well known and the need for further remediation will be evaluated, via sampling and analysis, during decommissioning to ensure compliance with the unrestricted use criterion. Section 5.5.1.b presents a discussion of deep soil contamination sampling in and near the RWST spill area.

2.2.4 Historical Radiological Status Including Original Shutdown Status

MY ran for approximately 16 full power years, had an early history of fuel clad failures and was known as a high source term plant. Dose rates in the loop areas in Containment were approximately 1000 to 2000 mrem/hr with surface contamination levels averaging in the 10,000 to 100,000 dpm/100 cm^2 range. Routinely-accessed areas of the PAB, Spray and Fuel Buildings had dose rates of 10 to 50 mrem/hr, walkways were kept less than 1000 dpm/100 cm^2, and equipment spaces had dose rates of up to 1000 mrem/hr and contamination levels on average of 5000 to 50,000 dpm/100 cm^2. The LSA, RCA Storage and LLWS Buildings had dose rates of 10 to 200 mrem/hr depending on the type and quantity of waste in storage and contamination levels ranged from 5000 to

50,000 dpm/100 cm^2 in liquid waste processing areas to less than 1000 dpm/100 cm^2 in walkways.

Normal system leakage was responsible for the contamination levels found within the Containment, Spray, Fuel and Primary Auxiliary Buildings. Secondary plant areas were kept uncontaminated with the exception of a few components (e.g., component cooling system filters and steam generator blowdown demineralizer) which gave general area dose rates of a few mrem/hr. Primary and secondary component cooling systems were known to contain small amounts of residual Cs-137 from minor heat exchanger leakage which occurred during power operations. The auxiliary boilers and auxiliary condensate receiver also showed evidence of minor contamination from heat exchanger leakage which occurred early in the plant's operating history.

In the late 1980s and early 1990s the plant began measures to reduce both the source term and surface contamination levels. Floor to ceiling area decontaminations were undertaken. High efficiency filters were installed in primary systems. One primary system chemical decontamination was performed which reduced primary system piping radioactivity levels by a factor of two.

In 1990, the plant experienced a primary to secondary steam generator tube leak. Prompt operator actions limited the secondary plant contamination.. Following the steam generator tube leak, secondary systems were extensively surveyed during recovery activities and no residual activity was identified. Temporary controlled areas were established in the turbine hall to work on RCP motors, and the turbine hall sumps have indicated detectable plant nuclides.

The plant was shutdown in December 1996 for evaluation of cable separation problems. During the extended outage, economic conditions led to the decision to permanently shutdown in August 1997. A second chemical decon was performed following the decision to decommission the plant. The decontamination factors for the second decon improved to five to ten which resulted in loop area dose rates in the range of 50 to 200 mrem/hr. Contamination levels throughout the plant remained consistent with pre-shutdown values.

2.2.5 Current Radiological Status

All fuel has been removed from the reactor and placed in the spent fuel pool. The fuel pool has been converted to alternate cooling and other primary systems have been drained and vented for decommissioning. Chemical and Volume Control System waste resins and filters have been removed for disposal. The reactor vessel contains approximately 33,660 gallons of slightly contaminated water. An additional 320,000 gallons have been added to the refueling cavity for shielding during reactor component removal, and will have to be processed as radwaste.

MY does not expect any primary systems to remain after decommissioning. If the diffuser remains, samples will have to be taken inside the pipe to properly characterize it. Demolition of structures to 3 feet below grade will remove the majority of embedded or buried piping. Remaining embedded or buried piping will be surveyed and/or classified for determination of actions required (if any) for compliance with site release for unrestricted use.

Based on both the Historical Site Assessment and the characterization surveys performed, a large portion of the site located to the West of Bailey Cove and North of the Ferry Road was determined to be non-impacted (Attachment A).

Containment and control measures have prevented the release of radioactive material beyond the Bailey Point area as evidenced by no detection of plant-derived radionuclides above background levels in any of the measurements taken in or on the land area West of Bailey Cove and North of the Ferry Road. The same control measures will remain in effect during the decommissioning to prevent migration of contamination into clean or non-impacted areas.

The impacted areas of the site extend from the Ferry Road in a southerly direction down Bailey Point.

2.2.6 Hazardous and Chemical Material Contamination

During its operational lifetime, MY used chemicals typical of steam power-generating facilities. In September 1998, MY had only non-bulk quantities of chemical and solvent waste stored on site awaiting disposal and no mixed wastes were in storage.

Preparation for decommissioning of the plant included removal of hazardous and chemical materials from plant systems. In 1998, 16,000 gallons of sodium hydroxide solution were removed from the spray chemical addition tank (SCAT) and neutralized, and chromates were removed from the water in the neutron shield tank using a totally-enclosed ion exchange resin process. A majority of the asbestos insulation was removed as part of the asbestos abatement project completed in January of 1999. Maintenance chemicals and hazardous materials were removed as specific plant areas were prepared for dismantlement.

Decommissioning of the plant will include removal of additional known contaminants in plant systems and structures. Mercury switches, lead components, and PCB light ballasts are some examples of hazardous materials that will be removed along with other plant components. Polychlorinated biphenyls (PCBs) found at other nuclear facilities are also present at MY but are limited to painted surfaces and in some cable insulation material. Asbestos abatement will continue to play a part in the removal of various components and building materials. Section 3.5 of this LTP describes the coordination of activities with other agencies with regard to these contaminants.

Over the operational lifetime of the plant spills to the environment occurred and were generally cleaned immediately. In 1988, the facility experienced a 12,000 gallon chromated water leak from an underground component cooling pipe. Following repair of the leak, monitoring wells were installed and the extent of contamination and the effectiveness of remediation were monitored to the satisfaction of the Maine Department of Environmental Protection (MDEP). In 1991, one of the main transformers shorted and released approximately 200 gallons of transformer oil to the Back River. The spill was remediated to MDEP's satisfaction following the event.

In these areas and throughout the site, MY will continue to work with the EPA and MDEP to demonstrate that areas have been adequately characterized, remediated if necessary, and are sufficiently clean to insure public health and safety. The EPA is supporting the Maine Yankee decommissioning project in several areas. The EPA is enabled by the Resource Conservation and Recovery Act (RCRA) to administer closure of facilities that were hazardous waste generators. Since the State of Maine Department of Environmental Protection has been delegated authority to administer the RCRA program in Maine, EPA is serving in a technical support role for the Maine Yankee site closure. EPA is expected to review all major closure-related documents and advise MDEP on their adequacy.

The EPA also is responsible for the Toxic Substances Control Act (TSCA), which serves as the primary means by which the use and disposal of PCBs and PCB-containing materials are controlled. PCBs have been identified above the TSCA limits of 50 parts per million (ppm) in electrical cable sheathing and, in limited areas, paint.

The EPA also administers the National Pollutant Discharge Elimination System (NPDES) permit program as authorized by the Clean Water Act. Maine Yankee maintained an NPDES permit during operation and presently has a renewal application pending before the EPA to reflect discharge of certain process wastewater during decommissioning. The State of Maine is presently seeking authority for MDEP to administer the NPDES program on EPA's behalf. Delegation of authority to MDEP is expected in the near future.

2.3 Site Characterization Survey Methods

Initial Characterization Survey (ICS) were performed by GTS Duratek and its subcontractor. The former Decommissioning Operations Contractor (DOC) (Stone & Webster) and its subcontractor (Radiological Services, Inc.) then performed the Continued Characterization Surveys (CCS) based on the GTS Duratek Study and built the FSS plan based on this information. Site characterization was performed by two different contractors using similar, but not identical, methods and techniques. These differences are noted within the methods and results sections of this report.

2.3.1 Organization and Responsibilities

GTS Duratek (GTS) was the prime contractor for the initial characterization surveys conducted from the fall of 1997 through the spring of 1998. GTS supplied hand- held instrumentation and performed field surveys. Subcontractors provided the following specialized services.

- IT Corporation performed the hazardous materials characterization survey and drive-over scans.
- Duke Engineering & Services performed the activation analysis.
- Canberra Industries provided on-site laboratory instruments.
- Team Associates performed the asbestos characterization.
- Quanterra performed off site laboratory analyses.

Continuing characterization activities began in the fall of 1998 and will continue through decommissioning. Samples were collected and on-site surveys and analyses performed. Laboratory analyses for the hard-to-detect radionuclides were performed by Duke Engineering Services.

2.3.2 Characterization Data Categories

Survey categories for site characterization were designated by GTS as surfaces and structures, systems, and environs (soils, sub-slab soils, sediments and groundwater) for both "affected" and "unaffected" locations based on the likelihood of the area being contaminated. The same designations are used for clarity and ease of comparing data.

a. Surfaces and Structures

This category included building interiors and exteriors with associated structures, and, where applicable, the exterior surfaces of plant systems and components because these surfaces have the same potential for residual levels of radioactive material as the building surfaces in which they are located. Surface and structure survey packages also included ancillary buildings and structures. Structural material background measurements were also included in this category. These measurements were intended to determine general background levels for various building materials. If background "reference" area measurements are required for final survey measurements, they will be performed in accordance with Section 5.0.

In total, the survey category included approximately 7,850 measurements in unaffected areas and approximately 6,350 measurements in affected areas. This intentional bias toward unaffected surfaces and structures ensured no unsurveyed or undetected locations were likely to exist. Affected structure surveys included 18 concrete core samples. Because concrete basement surfaces will be the key remaining structures upon license termination, an additional 51 concrete core samples were obtained to improve nuclide data.

b. Systems

This category included interior surfaces of process piping, components, ventilation ductwork, and installed drains and sumps. The levels of radioactive material on the internal surfaces of plant systems and components primarily depend on process operations. Therefore, these survey packages were separate from surface and structure survey packages. Plant system survey packages generally included one plant system.

This survey category included approximately 3,800 unaffected system measurements and approximately 1,050 affected system measurements. Again the surveys were biased toward the unaffected systems to provide a high likelihood of identifying any existing contaminated pipe or component.

Additional systems surveys were conducted in order to bound the extent of contaminated components within non-Restricted Area structures.

c. Environs

Land areas were surveyed and sampled to detect the presence and extent of soil contamination. Approximately one-third of the 820-acre site (original 740 acres + buffer land purchased later) land area received a gamma scan. Measurements taken over the entire property used a grid system to adequately locate survey points. Nearly 300 soil samples were taken, 180 of which were from unaffected areas. One survey package in this category was devoted to obtaining background soil and exposure measurements from an area similar in physical characteristics to, but located several miles from, the site.

A study was performed to determine the amount of radioactivity present in the vegetation above the soil surface. Comparison measurements of soil and overlying vegetation showed no radionuclide activity in the vegetation exceeding background levels. FSS soil samples are therefore taken with overlying vegetation removed but with the root ball intact in accordance with approved procedures.

Sediment, groundwater and surface water samples were also included in this category. Over 100 sediment samples were obtained from shorelines, outfalls, catch basins, runoff ditches and the forebay. Twelve sediment samples were also obtained from offsite sources such as the Damariscotta River and Harpswell for background purposes. Over fifteen water samples were taken from groundwater monitoring wells, sumps, catch basins and an outfall. Five water samples were taken from offsite or unaffected sources for background purposes. In addition, the Radiological Environmental Monitoring Program has collected over 27 years of sediment, groundwater and surface water sampling data. For instance, the Annual Radiological Environmental Operating Report for 1999, submitted to the NRC on April 27, 2000, describes the automatic composite sampler located at the discharge of the forebay to monitor water discharged to the Back River. Samples were collected at least every two hours and subsequently composited for analysis. Groundwater from an on-site location was monitored quarterly. Shoreline sediment cores were collected semiannually from two locations on Bailey Point.

Multiple soil samples were taken and composited to determine the amounts and ratios of the hard-to-detect radionuclides in the most contaminated soils onsite.

Scan and fixed surveys of pavement were performed to identify potential sub-surface contamination. Two areas of soil contamination beneath pavement were documented in the HSA. One area of sub-slab leakage from the liquid waste effluent line occurred underneath the Service Building floor. The results of this soil contamination were contained in the 50.75(g) file.

2.3.3 Characterization Survey Design

Characterization surveys were designed to sample each structure, system and land area onsite for the presence of radioactive contamination. A heavy emphasis was placed on non-affected (non-impacted) systems, structures and areas with 2750 more surveys taken on non-affected systems, 1500 more surveys taken on non-affected surfaces and structures, and 18 survey packages devoted to non-affected areas versus 7 for affected areas. This emphasis ensured that the full nature and extent of the contamination were identified and characterized.

The radiological characterization survey was organized, performed and reported in one of five "Groups" and 127 packages which are listed in Section 2.3.7. Each group is comprised of plant areas containing similar types of media, or material, and similar contamination potential. The types of media included surfaces, structures, systems and environs. The environs category included facility grounds within and outside the RA, the liquid effluent pathway, Montsweag Bay, groundwater wells and remote locations within the MY Atomic Power Plant site boundaries. The contamination potential for the media in a given group was generally categorized as affected and unaffected. Affected areas had medium to high potential for containing contamination. Unaffected areas had a low or no potential for containing contamination. The affected/unaffected designation was not intended to indicate final survey classification status, but was intended as a general descriptor of contamination potential. The methods for converting the characterization survey results to classification of plant areas for final site survey are described in Section 5 of this LTP.

Each group was further subdivided into survey packages that correspond to specific plant areas with similar operational history or physical location. The survey package breakdown is contained in Attachment B. All plant areas are included in one of the survey groups/packages. The five groups are listed below.

C Group A-Affected Surfaces and Structures

• Group B-Unaffected Surfaces and Structures

• Group C-Affected Systems

• Group D-Unaffected Systems

• Group R-Radiologically Affected or Unaffected Environs

These group designators were also used during continued characterization for survey package identification. Non-radiological data were collected and grouped into one of the following two categories listed below. The environs hazardous material characterization surveys included testing for PCBs, RCRA metals, semi-volatile organic compounds and volatile organic compounds.

C Group E-Hazardous Materials on Structures, Systems or Surfaces

C Group H-Hazardous Materials in Environs

Activation analysis calculations were also performed for the reactor vessel, reactor internals and the shield wall surrounding the reactor.

2.3.4 Instrumentation and Minimum Detectable Concentrations (MDCs) Instrument Selection and Use

Instrument selection, use and calibration for the MY characterization surveys were based on the assumed radionuclide mix and were performed in accordance with approved procedures. Instruments used and their MDCs are described in the applicable section.

a. Survey Methods

Direct measurements of structures were performed with 126 cm^2 gas flow proportional detectors for beta contamination. The MDC was between 500-2000 dpm/100 cm^2 (as compared to the screening values of 5,000-11,000 dpm/100 cm^2). The detector was kept within 1 cm of the surface.

Measurements of surface activity on small or restricted access areas were made using small Geiger-Mueller detectors or an array of multiple detectors for large bore systems or components. Measurement times were controlled in order to achieve the required MDCs.

Scan surveys were performed on both surfaces and land in order to detect areas of elevated activity for further investigation.

GTS Duratek performed scans of open land areas with a 1 inch by 1 inch NaI detector or the large "drive-around" plastic scintillator. Scan speeds were controlled in order to meet the required MDCs. Audible output was used with the handheld instruments to aid the surveyor in identifying areas of elevated readings. Continuing characterization scans were performed using a 2 inch by 2 inch detector swept in a pendulum pattern at a distance of 2 inches from the surface at a rate of 0.5 m/sec.

Samples of building materials, sediments, sludges and water were taken and analyzed using standard procedures and laboratory instruments. Smears for removable contamination were taken using standard techniques and laboratory counters. Exposure rates at one meter were measured using a NaI detector and a pressurized ion chamber. Soil samples of approximately 1000 g were cleaned to remove large debris and dried to remove moisture. Samples were counted in Maranelli beakers using GeLi detectors for gamma emitters. Samples were analyzed by off site labs for Hard-To-Detect (HTD) radionuclides.

b. Minimum Detectable Concentrations for Volumetric Measurements

The MDCs listed in Table 2-2 were typical values for both initial characterization and continued characterization samples, which included HTD nuclides. The lower values were for gamma spec analyses. When characterization soil samples were analyzed for HTDs, the MDCs were maintained at levels as low as practicable.

Minimum detectable concentrations (MDCs) were defined for measurements and analyses used to quantify soil and other volumetric activity. Similar instruments, procedures and MDCs applied to continuing characterization. MDCs for Volumetric Soil were less than 0.01 pCi/g for gamma nuclides versus approximately 3-4 pCi/g for the 10 mrem/yr screening value. MDAs for Volumetric Water were less than 2,500 pCi/L for H-3. There is no water screening value.

Table 2-2 Volumetric MDCs		
Type of Analysis	**MDC (pCi/g)**	
	GTS (ICS)	**DOC (CCS)**
Gamma Spectroscopy	0.10	0.01 - 0.1
Liquid Scintillation	2.0 to 3.0	2.5
Alpha Spectroscopy	0.10	0.01 to1.0
Radio Chemical Analysis	* 1 - 20 pCi/g	* 1 - 20 pCi/g

* except Ni-59

c. Structure and Surface Scan Sensitivities

GTS Duratek used a slightly different method for calculating scan sensitivities than the method specified in NUREG-1575/NUREG-1507. This approach increased the calculated scan MDCs by a factor of approximately 2.4. The use of this alternate approach had no effect on the interpretation and use of characterization data. The technicians evaluated detectably elevated readings during scan surveys based on changes in count rates regardless of the estimated MDC.

GTS Duratek performed a computerized sort of the direct measurements of total beta activity obtained during the characterization survey of unaffected areas by detector type, efficiency, local area background and use (building surfaces vs. system internals) in order to evaluate scan MDCs. The surface scan MDCs ranged from 2100 dpm/100 cm^2 for large area gas flow detectors to 16,000 dpm/100 cm^2 for system internals surveys (Attachment C).

The NUREG-1575/NUREG-1507 method was used to calculate scan sensitivities in the continuing characterization program. This method yielded surface scan MDCs of 1200-16,000 dpm/100 cm^2 depending on the instrument and material being surveyed.

d. Open Land Area Scans

GTS technicians performed gamma scans of open land areas using a Ludlum 44-2, 1 inch by 1 inch NaI detector, and a TSA Systems Limited large area plastic scintillator, VRM-1X. (See Table 2-3.) In accessible areas, the VRM-1X detector, a 1.5 inch thick, by 3 inch wide, by 33 inch long block of scintillator-impregnated plastic, was the detector of choice because it had the lower theoretical MDC. The relatively large surface area of the VRM-1X detector greatly improves the probability of detecting isolated areas that contain elevated levels of radioactive materials.

Table 2-3
Theoretical Scanning Sensitivities

Instrument	Minimum Detectable Concentration/Activity
Ludlum 44-2	14 pCi/g (Cs-137 source)
VRM-1X	11 pCi/g* (Distributed Co-60)
SPA-3	5 pCi/g (Cs-137 source)

* MDC as determined by Dr. Chabot in a letter to P. Dostie dated 11/12/98

Although GTS did not perform *a priori* MDC calculations, theoretical minimum detectable concentrations or minimum detectable activities for scans performed with a vehicle-mounted VRM-1X detector, traveling at less than 5 mph, were calculated for several geometries based on empirical data and numerical integrations following land surveys.

These data were examined by Dr. Chabot on 11/12/98 and found to be accurate within a factor of 2 to 4.

The SPA-3 detectors (2 inch by 2 inch NaI) were used for land area scans during continuing characterization with scan MDCs of approximately 5 pCi/g as calculated according to paragraph 6.7.2.1 of NUREG-1575.

e. Instrument Calibrations

Analytical and field instruments were calibrated using National Institute of Standards and Technology traceable sources representative of the assumed radionuclide mix at the MY site. Instruments were calibrated at the MY site and, for GTS, at the GTS Duratek Central Calibration Facility in Oak

Ridge, Tennessee or by vendors in accordance with the GTS Duratek Quality Assurance Project Plan for Site Characterization. Approved procedures were employed to specify on-site instrumentation calibration requirements for continuing characterization. The average energy of the beta particles in the MY radionuclide mixture was calculated. Based on the calculated average source beta energy of 0.088 Mev, Tc-99 (ave. beta energy of 0.085 Mev) was chosen for calibration. All of the alpha emitters have similar energies and Am-241 was chosen for the alpha calibration source. Tc-99 and Am-241 sources were used for calibrating gas flow proportional instruments used to perform surface scans and direct measurements. Cs-137 sources were used to calibrate exposure rate and soil scan instruments. The calibration program ensured that equipment was of the proper type, range, accuracy and precision to provide data to support the MY site characterization activities. The response of exposure rate and soil scan instruments to Co-60 was also determined during continued characterization in order to detect discrete Co-60 particles.

2.3.5 Quality Assurance

Quality Assurance plans were developed by both vendors for characterization work. The elements of these plans were very similar. Differences between plans are discussed below.

The GTS Quality Assurance Project Plan (QAPP) described the quality assurance requirements for the site characterization survey. The QAPP included applicable criteria from the GTS Duratek Quality Management System Manual specific to the MY project. The plan addressed sample collection, field survey measurements, sample analysis, data analysis/verification, and document control.

Continuing characterization was performed by the subcontractor using an approved CCS Quality Control procedure which addressed the quality elements for these surveys. The procedure covered the requirements and frequency for replicate measurements, sample recounts, split samples, instrument use and control, sample custody, data verification/control, document control and investigation of unusual results.

a. Quality Control Samples and Measurements

For each laboratory instrument used during both characterization and continuing characterization, laboratory personnel kept daily quality control charts, a log of samples analyzed to provide traceability for each step of the analysis, and a maintenance log. Daily quality control checks were compared to specified tolerances. Control charts were developed at the time of initial calibration using a statistical analysis of repetitive measurements. Laboratory personnel maintained control charts for energy, full width at half maximum (FWHM), and efficiency for each gamma spectroscopy system and performed trend analysis daily. Routine background and blank counts demonstrated that the detector or cave had not become contaminated and confirmed sample detection levels. Daily checks were also performed on the analytical balance which was used to weigh the samples. Instruments failing the daily checks were removed from service until repaired.

The GTS Sample Analysis and Data Management Plan identified required quality control samples and measurements. In addition to the daily instrument quality control described above, laboratory personnel used quality control samples and measurements to verify system performance and data reproducibility.

The following on site QC analyses were performed and compared by GTS using criteria in US NRC Inspection Procedure 84750:

- 10% of all samples were analyzed twice in the on-site laboratory (duplicate analysis)

- 10% of all samples were split and analyzed as two separate samples

Quality control at the contract (off site) laboratories also included daily instrument checks and quality control samples that were analyzed during analysis of a batch of samples. Quality control samples and analyses for a batch of 20 (or fewer) samples analyzed by the contract laboratory included: a blank sample, a matrix spike sample (laboratory control sample, LCS), and a homogenized split sample. Laboratory control samples and analyses performed by the off-site laboratory were required to meet a relative percent difference (RPD) of 20% in accordance with the laboratory's internal procedures.

An approved CCS Quality Control procedure for the sample quality control criteria was developed. This procedure covered instrument daily checks, split or spiked sample requirements and acceptability criteria. Five percent of all survey units were chosen for repeat surveys with 10% of scans and fixed point measurements being replicated. Agreement for replicates was considered to be values within $\pm$ 2 standard deviations.

Instruments not passing the daily source check requirements were tagged "Do Not Use" and were removed from service until repaired. Data not meeting the replicate count criteria were removed from the data base until evaluated by an FSS specialist or engineer.

Duke Engineering & Services Environmental Laboratory performed laboratory analyses under the requirements of DESEL Manual 100, "Laboratory Quality Assurance Plan."

The methods used by the off site laboratory for analysis of hazardous materials were based on the EPA method for solid waste analysis SW-846. Specific quality control samples, analysis, and acceptance criteria are specified in the analysis methods.

GTS personnel implemented the QAPP through:

- Scheduled audits and surveillances by on-site and off-site personnel
- Development of training matrices and training of personnel
- Development of records flow schedules
- Development of document control criteria
- Completion of readiness review checklists

Self-assessments for CCS were implemented in accordance with approved Radiation Protection Performance Assessment Program procedures. Training and qualification of survey personnel were assessed in accordance with the approved procedure for Selection, Training and Qualification of Radiation Protection Personnel. Records Control was maintained in accordance with approved procedures for QA Records Management.

b. Audits and Surveillances

MY provided oversight of survey and sample activities to determine whether the characterization plan was implemented as designed. External audits of project activities included assessments by MY personnel and subcontractors. These included an audit of the GTS Duratek facility in Kingston, TN and project-specific audits based on the Quality Assurance

Program Plan and other project plans. These audits did not identify any project-specific nonconformances. In addition, MY personnel and their contractors performed surveillances on daily project operations. Characterization personnel identified, tracked, and corrected concerns generated by these surveillances.

MY Radiological Engineering and GTS Duratek corporate and Field Services personnel performed internal audits of the project. Also, at the request of MY, GTS Duratek appointed an on-site surveillance technician. This inspector, trained on quality assurance procedures, performed daily surveillances on project activities. Characterization personnel tracked and corrected nonconformances identified by these surveillances according to approved procedures.

During continued characterization, audits and self assessments were performed on the characterization activities. The results of the findings were entered into the trend data base and tracked to resolution in accordance with the approved procedure for the Corrective Action Program Interface.

2.3.6 Data Quality Objectives

Initial site characterization was planned prior to the issuance of NUREG-1575. However, a retrospective look at site characterization revealed that Data Quality Objectives (DQOs) 1, 2, 3 and 4 were addressed by GTS Duratek. The characterization plan identified the problem, the decision method, the resources, the team, the decision makers, the sample requirements, the instrumentation and MDCs, the expected nuclides, the survey areas and basic data analysis. While the use of a formal DQO process may have resulted in a more efficient characterization process, the resulting data have been shown to be sufficient to meet the objectives listed in Section 1.0 and are therefore acceptable.

The DQO process was used during continuing characterization to meet the objectives outlined in Section 2.1. Contamination boundaries, radionuclide profiles, data standard deviations and projected sample sizes were determined during continuing characterization.

Data Quality Objectives 5, 6 and 7 are addressed in LTP Section 5, Final Status Survey, and Section 6, Compliance with the Radiological Criteria. In particular for DQO 5, the parameter of interest is specified as the mean of the residual contamination level in a survey unit, the action levels include the DCGL and the investigation levels, and the decision rule is described for the determination to release a survey unit. For DQO 6, the limitations of decision errors are addressed by specifying the respective probabilities of making a Type I and Type II decision error, the lower boundary of the grey region (LBGR) and the minimum value for relative shift. For DQO 7, the survey design for collecting data is optimized by using exposure pathway modeling to develop some site-specific DCGLs, adjusting the LBGR to obtain the optimum relative shift, evaluating survey instrumentation and measurement techniques and selecting appropriate actions following the exceedance of investigation levels.

2.3.7 Survey Findings And Results

The results of the characterization surveys are reported by survey group and package number as identified below. Site and Survey Area maps are provided in this section of the LTP to graphically depict the boundaries of each area. These maps are not drawn to scale but are sufficient to show the presence of areas of high contamination.

PACKAGE NUMBER	GROUP "A" Affected Structures and Surfaces Survey Packages
A0100	Containment Building - Elevation -2 ft.
A0200	Containment Building - Elevation -20 ft.
A0300	Containment Building - Elevation 46 ft
A0400	Fuel Building - Elevation 21 ft.
A0500	Demineralized Water Storage Tank TK-21 - Elevation 21 ft.
A0600	Primary Auxiliary Building - Elevation 11 ft.
A0700	Primary Auxiliary Building - Elevation 21 ft.
A0800	Primary Auxiliary Building - Elevation 36 ft.
A0900	Service Building Hot Side - Elevation 21 ft.
A1100	Low Level Waste Storage Building - Elevation 21 ft.
A1200	RCA Building - Elevation 21 ft.
A1300	Equipment Hatch Area - Elevation 21 ft.
A1400	Personnel Hatch Area - Elevation 21 ft.
A1500	Mechanical Penetration Room - Elevation 21 ft.
A1600	Electrical Penetration Room - All Elevations
A1700	Containment Spray Building - All Elevations
A1800	Auxiliary Feed Pump Room - Elevation 21 ft.
A1900	HV-9 Area - Elevation 21 ft.
A2100	Refueling Water Storage Tank (RWST) TK-4 - Elevation 21 ft.
A2200	Borated Water Storage Tank (BWST) - Elevation 21 ft.
A2300	Processed (Primary)Water Storage Tank (PWST) - Elevation 21 ft.
A2400	Test Tanks 14A/14B -Elevation 21 ft.
A9900	Concrete core contamination profile sampling
A9901	Activation analysis core sampling
A9902	Activation analysis core sampling

PACKAGE NUMBER	GROUP "B" Unaffected Structures and Surfaces Survey Packages
B0100	Turbine Deck - Elevation 61 ft.
B0200	Old Control Room - Elevation 21 ft.
B0300	Motor Control Center (MCC)/Battery Room - Elevation 62 ft.
B0400	Fire Pump House - Elevation 1
B0500	Condenser Bay - Elevation 21 ft.
B0600	Condenser Bay - Elevation 39 ft.
B0700	Service Building Cold Side - Elevation 21 ft.
B0800	Fuel Oil Building - Elevation 21 ft.
B0900	Emergency Diesel Generators - Elevation 21 ft.
B1000	Auxiliary Boiler Room - Elevation 21 ft.
B1100	Recirculating Water Pump House - All Elevations
B1200	Administration Center - Elevation 21 ft.
B1300	WART Building - All Elevations
B1400	Visitor and Information Center - Elevation 1
B1500	Warehouse 2 - Elevation 1
B1600	Training Annex Building - Elevation 1
B1700	Staff Building - All Elevations
B1800	Spare Generator Building - Elevation 1
B1900	Environmental Services Building - All Elevations
B2000	Bailey Barn - Elevation 1
B2100	Lube Oil Storage Room - Turbine Building Elevation 21 ft.
B2200	Cold Machine Shop - Turbine Building Elevation 21 ft.
B2300	Cable Vault Room - Turbine Building Elevation 39 ft.
B2400	Staff Building Tunnel - Staff Building to Turbine Building Elevation 21 ft.
B9800	Structural Background Survey

PACKAGE NUMBER	GROUP "C" Affected Plant Systems Survey Packages
C0100	Primary and Post Accident Sampling System
C0200	Waste Solidification System
C0300	Containment Spray System
C0400	Emergency Core Cooling System
C0500	Residual Heat Removal System
C0600	Primary Vents and Drains
C0700	Fuel Pool Cooling System
C0800	Waste Gas Disposal System
C0900	Pressurizer and Pressurizer Relief System
C1100	Reactor Coolant System
C1200	Boron Recovery System
C1300	Chemical and Volume Control System
C1400	Liquid Waste Disposal System
C1500	Primary Auxiliary Building Drains
C1600	Primary Auxiliary Building Ventilation
C1800	Containment Ventilation System
C1900	Steam Generators

PACKAGE NUMBER	GROUP "D" Unaffected Plant Systems Survey Packages
D0100	Condensate System
D0200	Water Treatment Plant Systems
D0300	Potable Water System
D0400	Sanitary Sewer System
D0500	Circulating Water and Screen Wash System
D0600	Service Water System
D0700	Fire Protection System
D0800	Lube Oil System
D0900	Compressed Air System
D1000	Auxiliary Boiler System
D1100	Steam Generator System
D1200	Main and Reheat Steam System
D1300	Auxiliary Steam System
D1400	Main Turbine and Turbine Control System
D1500	Steam Dump and Turbine Bypass System
D1600	Main Feedwater System
D1700	Emergency/Auxiliary Feedwater System
D1800	Heater Drain and Extraction Steam System
D1900	Component Cooling Water System
D2000	Vacuum Priming and Air Removal System
D2100	Amertap System
D2200	Secondary Plant Sealing System
D2300	Auxiliary Diesel Generator
D2400	Secondary Sample and Chemical Addition System
D2500	High Pressure Drains
D2600	Environmental Services Laboratory Systems
D2700	Administration Building HVAC System
D2800	Information Building HVAC System
D2900	Turbine Building Ventilation System
D3000	Staff Building HVAC System
D3100	Service Building HVAC System

PACKAGE NUMBER	GROUP "D" Unaffected Plant Systems Survey Packages
D3200	Hydrogen and Nitrogen System
D3300	Turbine Building Sumps and Drains
D3400	Low Level Radioactive Waste Storage Facility

PACKAGE NUMBER	GROUP "R" Environs Affected and Unaffected Survey Packages
AFFECTED	
R0100	RCA portion (West Side) of Protected Area Yard
R0200	Balance of Protected Area (East Side)
R0300	Roof and Yard Drains #006, #007 and #008
R0400	Forebay Area Shorelines
R0500	Bailey Point
R0600	Ball Field
R0700	Construction Debris Landfill
UNAFFECTED	
R0800	Administration and Parking Areas
R0900	Balance of Plant Areas
R1000	Foxbird Island
R1100	Roof and Yard Drains #005, #009-12, #017 and N-12
R1200	Low Level Radioactive Waste (LLRW) Storage Building Yard
R1300	Dry Cask Storage Area
R1400	Westport, Montsweag Bay, Bailey Point Cove and Plant Area Shorelines
R1500	Ash Road Area Rubble Piles
R1600	Owner Controlled Area West of Bailey Cove
R1700	Owner Controlled Area North of Old Ferry Road
R1800	Bailey House Area
R1900	Bailey Cove
R2000	Diffusers
R2100	Maintenance Yard (Stockyard)
R2200	Background
R2300	SFPI Substation Slab
R2400	IT Duplicate Samples
R2500	Driveover Elevated Areas
R2501	Follow-up sampling at Elevated Soil Sample Locations (south of Refueling Water Storage Tank and Contractor Parking Lot)
R2800	10 CFR 61 Analysis Sampling

Hazardous and chemical material surveys were performed on the materials, systems and areas as specified in the tables for Group E and Group H below. The data for these groups are presented in the Summary of Site Characterization Data section which follows.

PACKAGE NUMBER	GROUP "E" Plant Surfaces, Structures and Systems Hazardous Material Survey Packages
E0100	Protected Area Paint
E0200	Plant Electric Components
E0300	Transformer Oils
E0400	Plant Pump Oils
E0500	Various Plant Fluids
E0600	Component Cooling Water
E0700	Brass, Bronze and Cadmium Plated Components
E0800	Plant Batteries
E0900	Mercury Components
E1000	Asbestos Insulation and Other Materials
E1100	Asbestos Containing Components
E1200	Lead Shielding
E1300	Paint Outside Protected Area

PACKAGE NUMBER	GROUP "H" Environs Areas Hazardous Material Survey Packages
H0100	Oil and Hazardous Material Transfer and Handling Areas (4)
H0200	Diesel Oil Tank Loading Area
H0300	Main, North, Spare and Shutdown Transformers
H0400	Roof and Yard Drains #006, #007 and #008
H0500	Solid Waste Storage Area
H0600	Primary and Secondary Side Waste Storage Building Yard Areas
H0700	Drumming/Decontamination Waste Accumulation Area
H0800	Diffuser Forebay
H0900	Reactor Water Storage Tank Area
H1000	Groundwater Monitoring Wells B-201 through 206, MW-100, BK-1
H1100	Warehouse Yards
H1200	Fire Pond and Yard Area
H1300	Construction Debris Landfill
H1400	Bailey Point
H1500	Administration and Parking Areas
H1600	Roof and Yard Drains #005, #009-12 and N-12
H1700	Surface Flow Drain #005
H1800	Balance of Plant Area
H1900	Foxbird Island
H2000	Low Level Waste Storage Yard
H2100	Dry Cask Area
H2200	Environmental Services Laboratory
H2300	Switchyards
H2400	Areas Outside Plant Impact

2.4 Summary of Initial Characterization Survey Results

The operational history and the range of contamination determined during site characterization are summarized in this section for the survey groups indicated above. More detailed data including mean, maximum, and standard deviation are presented by survey package in Attachment B.

2.4.1 Group A "Affected Structures and Surfaces"

Group A includes buildings and surfaces within the RA including levels of the Reactor Containment, Fuel, and Primary Auxiliary Buildings, as well as tanks containing radioactive liquids, electrical/mechanical penetration areas and concrete surface samples. Areas of known contamination with very high dose rates were sampled less than areas with more moderate dose rates in order to maintain the exposure to surveyors ALARA. Survey data were taken from posted areas which included High Radiation Areas, Radiation Areas, Radioactive Material Storage Areas and Contaminated Areas. These areas include the reactor coolant system and waste processing equipment and are among the most highly contaminated areas on site. However, several locations within this group contained no radioactive systems, components and structures or were found to be below station limits for posting as contaminated (viz., DWST, PWST, electrical and mechanical penetration areas and the auxiliary feed pump room).

Maximum total surface activities ranged from greater than 100,000 dpm/100 cm^2 in the RCA Building, Containment Building (CTMT), and Spray Buildings to less than 1000 dpm/100 cm^2 in auxiliary support areas (e.g., electrical/mechanical penetrations). Maximum removable beta activities ranged from greater than 128,000 dpm/100 cm^2 in the CTMT to less than MDA in auxiliary support areas. No removable alpha sample activities were above the MDA values which indicated little or no transuranic (TRU) surface contamination. Maximum net exposure rates reported in Attachment B ranged from about 4,000 μR/hr in the Primary Auxiliary Building (PAB) to around 5 μR/hr in the mechanical penetration area. Operational surveys reported containment exposure rates ranging from 1 mrem/hr to over 1000 mrem/hr.

Group A results combined with the operational survey data and knowledge of process provided the information needed to target those structures within the RA requiring remediation, establish radionuclide profiles and provide estimated radioactive waste volumes.

2.4.2 Group B "Unaffected Structures and Surfaces"

Group B was comprised of buildings and surfaces located outside the RA including the Turbine Hall, sections of the Service Building, the Control Room, office spaces and various out buildings such as the Fire Pond Pump House, the warehouse, and the Bailey House/Barn. With the exception of a few closed secondary systems and a few locations in the Turbine Hall, Service Building and warehouse, none of these buildings contained or stored radioactive material during plant operation and are therefore some of the lowest activity areas on site. Sealed sources for instrument calibration were stored at the Bailey House environmental laboratory.

The crane bay and turbine deck in the Turbine Hall were used for RCP motor refurbishment. The 1990 steam generator tube leak affected steam and feedwater components in the Turbine Hall. The auxiliary boilers were known to be internally contaminated. Some areas within the Service Building such as the old decon shower and primary chemistry lab sample hoods were also known to be slightly contaminated. The warehouse was used as a shipment and receipt point for small quantities of packaged radioactive material. There was no evidence of leakage detected at the warehouse from packages shipped or received.

Maximum total surface activities ranged from a high values of 3700 dpm/100 cm^2 and 8600 dpm/100 cm^2 in the Turbine Building (certain floor areas) to lows of <1000 dpm/100 cm^2 in outlying areas, such as the cable vault. The Ball Field Dugout indicated 700 dpm/100 cm^2, which was later identified by the State of Maine as Co-60. Maximum removable beta activities ranged from 200 dpm/100 cm^2 in the Turbine Building to less than MDA in other areas. No areas had plant related alpha activity above the MDA level. Maximum exposure rates ranged from 26 µR/hr in the Service Building to 2 µR/hr in the Turbine Building. Tritium was detected slightly above MDA in several water-containing systems. High beta readings in the Bailey House were confirmed to be NORM from the granite foundation blocks.

Group B surveys verified that most of the Turbine Hall was free of residual radioactivity. Continuing characterization surveys established the extent and limits of radioactivity in the areas in which it was found.

2.4.3 Group C "Affected Plant Systems"

This group was comprised of the radioactive systems such as the RCS, CVCS, ECCS, liquid and solid waste, containment ventilation and primary vents and drains. The survey packages in this group consisted of systems and components that will be removed and disposed of as radioactive waste during decommissioning and, therefore, do not require characterization to support Final Status Survey (FSS).. These are the highest radioactively contaminated systems at MY.

Total surface activities were not measured on these systems' internals, as their activity levels were too high. Instead, 15 cm and 1 meter external exposure rate measurements were taken at four quadrants from system locations, to support dose to curie calculations, for waste shipping purposes. Internal system surfaces of the steam generators were found to be contaminated up to 500,000 dpm/100 cm^2 removable beta activity. Alpha activity was present at as much as 35 dpm/100 cm^2 in the CVCS indicating possible TRU contamination. Exposure rates in these areas ranged from a low of 13 µR/hr in the Waste Solidification system to more than 16,000,000 µR/hr in the Spent Fuel Cooling and Refueling system.

Group C results verified the extent of contamination in primary systems and provided data needed to support the Radiation Protection Program during component removal in addition to providing information needed for waste classification.

2.4.4 Group D "Unaffected Plant Systems" Including the Sewage Treatment System

This group consisted of secondary side systems that were designed to remain non-contaminated. Examples of these systems are main steam, feedwater, compressed air and potable water. However, certain parts of the secondary side systems do contain minor levels of contamination. The auxiliary condensate system was known to be slightly contaminated due to aux boiler problems early in plant life. Turbine Hall sumps were known to be slightly contaminated due to reactor coolant pump motor refurbishment activities taking place in the Turbine Hall. Steam and feedwater systems were potentially impacted by the 1990 steam generator tube leak. The Service Water system was impacted by liquid effluents from the Test Tanks. Several of the systems crossed over to the RA, where elevated readings were detected in/on the systems but were later attributed to NORM interference in the analyses. Group D systems were generally the lowest in activity of all those surveyed.

Until the early 1980s when they were disconnected, hot side shower drains and toilets were directed to the sewage treatment plant. Characterization surveys showed elevated readings in one hotside shower drain. Over the past two years, routine chemistry analyses of both the on site holdup tank and the municipal treatment facility have shown no plant-derived radionuclides. Radionuclides have been detected in the sewage plant as a result of employees receiving medical isotope therapy.

Survey results from Group D established the limit and extent of residual activity in systems expected to be clean and provided information to properly control the systems as well as classify the waste during decommissioning. Some of the systems in Group D had elevated readings indicating the possible presence of plant derived radioactive material. Further measurements were made on these systems as part of the continuing characterization plan to properly evaluate the level and extent of contamination. These measurements will support release and/or disposal determinations.

2.4.5 Group R "Environs Affected and Unaffected"

The group was broken down into 7 affected and 18 unaffected areas. Environs sampling covered all areas of the 820 acre site (740 acres original site + purchased buffer properties). Fifteen of the sample areas showed no detectable plant derived radioactivity. Ten of the areas (R0100, R0200, R0300, R0400, R1000, R2000 and R2300 within the protected area and R0500, R0900 and R1300 outside the protected area but on Bailey Point) had elevated readings requiring further evaluation and sampling.

Asphalt, sub-asphalt soil and uncovered soil to the South and West of Containment, Spray, Fuel and RCA Storage Buildings were known to be contaminated by system leaks and radioactive waste container storage. Excavated soil and asphalt from the RA were temporarily placed on Bailey Point and later returned to the RA. Silt from condenser cooling water intakes was removed and spread on site land located to the north and west of the 345 kV electrical switch yard. Plant-derived radionuclides had been detected in estuary sediments as a result of permitted liquid releases by environmental samples (REMP reports) taken at various times during plant operation. Minor contamination was located near storm drains adjacent to the RA. Contamination levels ranged from 1pCi/g to 11 pCi/g for Co-60 and 1pCi/g to 156 pCi/g for Cs-137 in the areas of known soil contamination from old leaks/spills (R0100).

Marine sediment samples were obtained from shorelines, outfalls of catch basins, runoff ditches and the forebay. In addition, the Radiological Environmental Monitoring Program had collected over 27 years of sediment sampling data. Shoreline sediment cores were collected semiannually from two locations off Foxbird Island. Additional sampling of off-site marine sediments will be conducted pursuant to an agreement between Maine Yankee and Friends of the Coast (FERC Offer of Settlement dated December 31, 1998.)

Survey packages with indications of potentially elevated activity levels (R0500, R0600, R0700, R0800, R1000, R1300, R1600 and R1800) were combined into an investigation package designated R2500. The highest levels of activity were detected on Bailey Point from the investigation package R2500 (up to 34,000 pCi of Co-60) and the activity was remediated during sampling. Follow up samples taken in three areas after remediation of detected activity were documented in package R2501.

Three areas (R1500, R1600, R1700) were classified as non-impacted based on operational data, the Historical Site Assessment and the Characterization results.

Group R surveys determined which land areas were non-impacted and which were impacted. This group also provided the information necessary to project waste volumes from contaminated soils.

2.4.6 Ventilation Ducts and Drains

Results for the biased sampling of building vents and drains can be found within the survey data for Groups C, D and R. Ventilation ducts and system drains were sampled as the most likely collection point for system contamination. This biased sampling provided a high level of assurance that contaminated systems were located, identified and, when found within secondary side buildings, marked to provide the necessary level of control over radioactive material.

Affected System Vents and Drains (C0600, C1500, C1600 and C1800) showed mean removable contamination values ranging from 53 to 51,000 dpm/100 cm^2 and maximum values from 6000 to 140,000 dpm/100 cm^2.

Unaffected System Vents and Drains (D1800, D2000, D2500, D2700, D2800, D2900, D3000, D3100 and D3300) had two systems positively identify residual radioactivity. The Service Building HVAC (D3100) had significant activity above the MDA which was due to the hot side ventilation sources going to the Service Building ventilation duct work. D3000 Turbine Building Sumps and Drains had two (2) sumps test positive for plant derived nuclides (up to 1.7pCi/g Co-60). The Sump Oil Collection Tanks (TK-91) also test positive (1.1 pCi/g Co-60). There were four (4) other systems ((D1800 - Heater Drain Extraction Steam, D2700 - Admin Building HVAC, D2900- Turbine Building Ventilation, and D3000 - Staff Building HVAC with elevated activity. However, the elevated readings were likely due to radon daughter activity. This will be confirmed during CCS and/or the operation free release program. The High Pressure Drains showed tritium activity at levels just above MDA. Tritium in these areas have been attributed to NORM interference in the analyses.

Survey results from this group established the limit and extent of residual radioactivity in systems and provided necessary information for properly controlling material and for proper classification of waste during decommissioning.

2.4.7 Buried and Embedded Piping

A review of prints and drawings was performed during CCS to determine the amount of buried and embedded pipe. MY has a limited amount of piping actually embedded in concrete (less than 1000 linear feet). Total embedded piping includes approximately 800 feet of primary and secondary component cooling water pipes. Component cooling piping showed maximum activity up to 22,000 dpm/100 cm^2 and will be removed during demolition activities. Small segments of refueling cavity and spent fuel pool skimmer piping (approximately 175 feet) are embedded within the walls of the two pools. The skimmer piping is known to be contaminated and activity levels could be as high as 20,000 to 180,000 dpm/100 cm^2 removable beta contamination based on data obtained from spent fuel pool cooling (C0700) and RHR (C0500) survey packages. This piping will be removed.

Circulating water and service water pipes are buried cast concrete pipes rather than embedded pipes. Eighteen direct measurements above MDC were identified in the circulating water pipes. Service water discharge piping receives the liquid effluent overboard pipe with approximately a 3 foot embedded section and showed maximum activity levels of 3100 dpm dpm/100 cm^2 of removable beta contamination. Mean values were less than MDA.

Embedded piping above the 17 foot elevation will be removed. Pipes below 17 feet will either be removed during demolition or will be properly evaluated to ensure compliance with the enhanced state standards of 10 mrem/yr for all pathways including not more than 4 mrem/yr from groundwater sources of drinking water. Maine Yankee has produced an informational set of site drawings showing the “as left” condition after decommissioning. These drawings identify the remaining buried or embedded pipe, conduit, building penetrations, cable vaults, and duct banks. This set of drawings will be used to plan FSS surveys.

The following is a list of the approximately 6000 feet of buried and 660 feet of embedded piping which is expected to remain following decommissioning and which will be decontaminated as necessary and subject to FSS.

a. Containment Spray Piping and CS Valves-approximately 72 ft. (C0300): During plant operation, the system was filled with reactor coolant water. Site characterization surveys identified this as a contaminated system. Gamma isotopic samples collected from the system identified the presence of plant-derived nuclides (Co-60 and Cs-137). The portion of the system that will remain following demolition of above grade structures is embedded in the concrete foundation of the Containment Building. Two valves from the containment spray system are also encased in concrete. Levels up to 40,000 dpm/100 cm^2 were detected in the spray system (C0300).

b. Containment Foundation Drains-approximately 330 feet.(C2000): The foundation drain system was used to transfer groundwater from around the Containment Building foundation to lower the hydrostatic pressure on the foundation. The system consists of four partially embedded transfer pipes that drain to the foundation sump. The system has a high potential for residual contamination. The drain system is wholly contained within the RA and has been subjected to liquid spills in the soil around the Containment Building. The system was not surveyed during Site Characterization, however, the sump water was sampled periodically. Tritium is the only nuclide identified in the sump water at levels exceeding natural background. A water sample was submitted for HTD analysis during CCS and only tritium was detected. No removable surface contamination or direct surface measurements have been made.

c. Sanitary Waste (D0400): A portion of the sanitary waste piping is buried beneath the Turbine Hall floor slab and extends to the sewage treatment plant. At one time early in the plant's operation, the pipe transferred waste from sanitary facilities located within the RA. The original discharge point for treated sanitary waste was into the circulating water inlet bay. In the mid-1980s, the sanitary system was connected to the town of Wiscasset sewage system. The sanitary system, including the discharge to the town of Wiscasset, has been sampled periodically since the plant began operation. Radionuclides detected in recent years were limited to medical isotopes which are short lived and would not be present by the time the system pipe is surveyed. Of 37 fixed point surface measurements of the system taken during ICS, two were in the RA, and both indicated elevated activity of up to 5700 dpm/100 cm^2. Both of these samples were from a disused drain in the system that will be removed during dismantlement. No removable contamination was identified in the system. Gamma isotopic samples from the system did not indicate the presence of plant-derived radionuclides.

d. Circulating Water System-approximately 1600 feet (D0500): The circulating water system consists of 4 buried concrete inlet pipes which carried sea water from the Back River to the condenser then overboard to the forebay and is finally discharged through a diffuser in the Back River, down stream of the inlet. The circulating water system is considered a "secondary side" system in that there was a physical barrier (condenser tubes and steam generator tubes) between the circulating water and the contaminated primary plant (reactor coolant system). The circulating water system has a very low potential for residual contamination. The operational history of the facility indicates no significant primary to secondary leakage occurred. Additionally, the circulating water system pressure was maintained above the pressure of the turbine exhaust steam in the condenser so that even if there was a condenser tube leak, it would have carried sea water into the condensate system. During Initial Site Characterization, low levels of detectable activity were identified on the main condenser outlet side of the circulating water system. The suspected cause of the contamination was recirculation of allowable effluent discharges into the suction side of the Circulating Water Pump House. The maximum fixed point total surface contamination measurement collected during ICS was 811 dpm/100 cm^2. No removable contamination was identified in the system. Gamma isotopic samples collected in the system during ICS did not identify any plant-derived nuclides.

e. Service Water System (D0600): The Service Water System consists of two buried inlet pipes which carried sea water through the component cooling heat exchangers. The discharge of the system consists of a single buried line which goes into the seal pit.

The discharge side of the pipe receives the liquid effluent discharge pipe. During Site Characterization, low levels of detectable activity were identified on the discharge side of the piping. No direct beta measurements were above the MDA. Nine samples of removable beta activity were detected above the MDA (3134 dpm/100cm^2 was the maximum value). The positive indications of residual activity in this system are associated with the liquid effluent header location and the liquid radwaste radiation monitor installed at that location. Gamma isotopic samples collected at the liquid effluent line entrance point and at the radiation monitor were positive for Co-60 (700 pCi/g). The waste header is contained within its own local Restricted Area within the Turbine Building.

The radwaste piping will be removed and disposed of as radioactive waste. The remaining portions of the service water discharge piping meet the criteria of a Class 3 area.

f. Fire Protection (D0700): The water-filled portion of the fire protection system is the only section that will remain following demolition. Water for firefighting was stored in a man-made storage pond located on site. Makeup water for the pond came from Montsweag Brook. (The storage pond is addressed as part of survey area R0900). The fire protection system was not piped to containment. The system consists of a loop of buried pipe which circles the yard and supplies various hydrants and headers. The fire protection system is considered a “support system” in that it did not interface with other operating systems (e.g., primary coolant or steam supply). The fire protection system has a very low potential for residual contamination. Although sections of the system did reside within the RA, system pressures were sufficient to prevent inleakage. The fire water system has been cross-connected with potentially contaminated systems in the past. However, samples collected during CCS have only identified naturally occurring radioactive material. The maximum fixed point total surface contamination measurement taken during ICS was 1116 dpm/100 cm^2. Gamma isotopic samples collected during ICS did not identify any plant-derived radionuclides in the system.

g. Storm Drains (CD3500): The Storm Drain (SD) system is used to drain storm water and runoff from the facility to the Back River and Bailey Cove. The system functions as a gravity drain system to remove the water via a system of drain grates, manholes and system piping. The system drains the entire site both inside and outside the Protected Area. Manholes 1 through 3 (Section 1 of the system) drain the Protected Area outside the Restricted Area and south of the Turbine Building and Service Building. The outfall for this portion of the system is a 24” line that drains to the Back River south of the Circulating Water Pump House (CWPH). Manholes 4 and 5 (Section 2 of the system) drain an area inside the Protected Area outside the Restricted Area east of the Turbine Building. This line drains the area around the Main Transformers. The outfall for this leg of the system is a 15” line that drains to the Back River north of the CWPH. Manholes 6 through 11 and un-numbered manholes north of the Turbine Building (Section 3 of the system) drain an area both inside and outside the Protected Area. The area drained is all outside the Restricted Area. These legs all collect at Manhole 7 and the combined outfall is routed to the Back River immediately adjacent to the north side of the CWPH. Manholes 13 and 14 (Section 4 of the system) drain the upper access road and the upper contractor parking lot. The outfall for this section of the system is the Back River north of the Information Center building. Manholes 30A, and 31 through 37 (Section 5 of the system) drain an area inside the Protected Area in the Restricted Area. This leg of the system drains the main RCA Yard area around the Containment Building and the alley between the Containment Building and the Service Building. These legs all collect at Manhole 35 and the combined outfall is routed to the Forebay Seal Pit. Manholes 21 through 24 (Section 6 of the system) drain the north side of the Restricted Area and the roof of the WART Building. The area drained is inside the Protected Area and both inside and outside the Restricted Area. The combined outfall for this leg joins another leg at Manhole 27. Manholes 25A, 25B, 26 through 29 and 38 (Section 7 of the system) drains areas adjoining the Fire Pond and Warehouse and outside the west end of the Restricted Area. The outfall from Manhole 24 joins this leg at Manhole 27. The combined outfall for this leg of the system is routed to Bailey Cove.

Samples collected during ICS and knowledge of process indicate that the Storm Drain system has a low potential in some legs and a high potential in some legs for residual contamination. Sections 1 through 4 have a low potential for residual contamination. Sections 5 through 7 have a high potential for residual contamination. Sections 1 through 4 drain areas that have historically been outside the Restricted Area and have a low potential for residual contamination. Sections 5 through 7 drain areas in and adjacent to the Restricted Area and may have become contaminated due to loose surface contamination in and on yard structures and equipment being washed into the drain legs by rain water runoff and snow melting.

Since the roof drains flow to the storm drains and the portions of the roof drains above 17 feet will be removed, the roof drains will be included in the storm drain survey.

h. Containment Building Penetrations (CD3700): Several Containment Building penetrations will remain following demolition of the above grade structure. The penetrations contain embedded piping from numerous primary and secondary systems. The remaining penetrations are as follows:

- Approximately 16 linear feet of 2” piping

- Approximately 36 linear feet of 8” piping
- Approximately 24 linear feet of 10” piping
- Approximately 24 linear feet of 12” piping
- Approximately 28 linear feet of 16” piping
- Approximately 24 linear feet of 24” piping
- Approximately 20 linear feet of 30” piping
- Approximately 8 linear feet of 40” Fuel Transfer Tube piping
- Approximately 20 linear feet of 42” piping

The calculated surface area of the remaining Containment Building penetrations is approximately 94 m^2.

The Main Steam Valve House will have four sections of embedded piping remaining in the foundation following demolition of the above grade structure. The remaining embedded piping sections are as follows:
- One three foot length of 4” Water Treatment system piping
- One three foot length of 8” piping marked “Spare” on area PI&Ds
- One three foot length of 8” Secondary Plant Condensate Makeup system piping
- One three foot length of 1 ½” Auxiliary Steam Generator Feed Pump Discharge and Recirculation Piping

The calculated surface area of the Main Steam Valve House penetrations is approximately 2 m^2. These sections of embedded piping will be surveyed as part of the Containment Penetrations survey area.

The penetrations that will remain in the Containment Building have a high potential for residual contamination. One of the systems identified as having a remaining section of embedded piping is Containment Spray, which is known to contain residual contamination.

ICS data collected in the Containment Spray system (C0300) indicate the presence of removable contamination and gamma isotopic samples identified the presence of plant related radionuclides. ICS were not collected in the Fuel Transfer Tube. Additionally, no specific contamination controls have been established for the remaining sections of the embedded piping and the majority of the Containment Building is posted and controlled as a surface contamination area.

2.4.8 Asphalt, Gravel and Concrete

Two site locations containing asphalt and gravel from non-RA construction work were sampled for activity (R0700 and R1500). Neither location showed activity above background for plant-derived nuclides.

Because of the potential impact of concrete on the exposure pathway, concrete core samples were collected and analyzed during initial characterization (A9900, A9901, A9902) and continuing characterization. Seven of the concrete samples taken during initial characterization were subject to analysis under continuing characterization for HTD nuclides at low MDCs. An additional 51 concrete samples were collected and analyzed by gamma spectrometry during continued characterization. The results of these analyses are provided in Section 2.5.3. Concrete activity was found to be due to penetration of surface contamination as well as activation of concrete constituents in areas exposed to neutron flux. (Activated concrete comprised approximately 5% of the concrete in Containment.) Surface contamination penetration was primarily limited to the top 0.1 cm. Activation activity generally followed expected activation curves, peaking at about 4 inches into the concrete, and dropping off at greater depths. Slight anomalies in concrete activation were noted in the vicinity of embedded rebar. Positive indications of activation were seen as deep as 24 inches in some concrete samples that were exposed to high neutron fluences. As noted in section 3.3.3, activated concrete will be removed down to the activated concrete DCGL.

As part of CCS, samples of local fill material (sand, gravel, and till) were analyzed for bulk density and Kd. Activated Concrete at levels above the activated concrete DCGL will be removed.

2.4.9 Paved Areas

One paved area near the warehouse (R0900) exhibited one elevated exposure reading. A small contaminated area was removed during sample collection and was found to contain a small amount of Co-60. Resurvey confirmed removal of the contamination. Paved areas within the RA are known to have sub surface asphalt and sub surface soil contamination as described in the “Historical Site Assessment” section.

2.4.10 Components

The status of individual components is given in the systems data, Groups C and D. Group C components are found in radioactive systems and are known to be contaminated.

Section 2.4.3 describes the affected components in Group C; Section 2.4.4 describes the unaffected components in Group D, and Attachment B provides a detailed summary of components during ICS.

2.4.11 "Structures, Systems and Environs Surveyed For Hazardous Material" (Groups E and H)

These surveys identified expected amounts of waste chemicals, lubricants and solvents; toxic metals in switches; and PCBs in paints and cables. Some areas of soil contamination by motor oils/fuels were discovered which will require further evaluation. Characterization activities confirmed the presence of lead-based paint and PCBs in both cables and paints. Several small areas of soil were found to be contaminated by chemical or hazardous material.

Hazardous material health and safety considerations will be assessed through the RCRA closure process described in Section 3.5.

2.4.12 Surface and Groundwater

ICS sample results for surface and groundwater were reported within the individual survey area packages (R0100, R0200, R0300, R1100, R2200 and R2400) and are summarized in Attachment B.

Tritium was the only plant derived radionuclide detected in groundwater and surface water during ICS. The overall range of the tritium analyses was <793 pCi/L to 6812 pCi/L. The highest value was from the Containment foundation sump. All of the measurements were well below the EPA Drinking Water MCL of 20,000 pCi/L. The Containment foundation sump is currently being monitored and trended as part of CCS to determine if there is evidence of plant derived tritium contamination in the groundwater.

2.4.13 Background

ICS measurements were made of several types of construction materials from offsite locations which were used as background samples. Soil samples from remote locations were also taken and analyzed to be used as background soils.

ICS material backgrounds (concrete, brick, ceramic, etc.) were subtracted from reported ICS data direct measurements of total beta activity. ICS environs background (soil, sediment, water, etc.) were collected for informational purposes only. ICS environs background data were not subtracted from ICS environs survey reported data.

a. Material Background

The natural levels of radioactivity in plant construction materials affected direct measurements for total beta activity. To quantify this effect, GTS Duratek performed a background study at the Central Maine Power Headquarters Building in Augusta, Maine. The study included direct measurements for total beta activity on painted and unpainted concrete and concrete block, ceramic tile, and asphalt. Other materials encountered during the characterization survey such as glass, carpeting, and steel were not included in the background study since their natural radioactivity would not contribute significantly to direct measurements for total beta activity. Survey personnel used the same instruments for the structural background survey as were used for the characterization survey. Count times were adjusted to ensure minimum detectable activities of approximately 300 dpm/100 cm^2. Project personnel used these results to correct data gathered from similar surfaces during the characterization survey.

The following is a summary of ICS material backgrounds:

Table 2-4: Summary of ICS Material Backgrounds

MATERIAL	AVERAGE (dpm/100cm^2)
Bare Concrete (& block)	665
Painted Concrete (& block)	478
Asphalt	925
Ceramic Tile	1109
Other (duct, bare & painted metal, etc.)	0

b. Environs Background

The purpose of the environs background study was to measure and document the levels of radionuclides, especially Cs-137, present in local soils and typical background exposure rates. The survey sampling and measurement techniques complied with approved procedures and supporting guidance documentation. Sample materials for the background study included surface soils, sediments and groundwater. The project team performed gamma spectroscopy for all samples, and analyzed groundwater for tritium. The average Cs-137 concentration in soils was determined from samples collected at the Merrymeeting Airfield, from a hay field, woodlands, and scrub lands. The average Cs-137 concentration in marine sediments was determined from samples collected from the Damariscotta River, near Dodge Point and Harpswell. Groundwater concentrations were determined from the Eaton Barn, Bailey House, and Days Ferry. No groundwater samples had detectable Cs-137 or tritium concentrations (above MDA).

The survey also included an *in situ* gamma spectrum with a MicroSpec multichannel analyzer/sodium iodide detector. Survey technicians measured background exposure rates with a sodium iodide detector. Additionally, the survey team took both sodium iodide and pressurized ion chamber (PIC) measurements at each of the background soil sample locations in the hay field at Merrymeeting Airfield to observe the energy response of the PIC versus the sodium iodide detector. The project team calculated the background exposure rate and PIC measurement ratio for information and did not use the results to adjust any other measurements.

The following is a summary of ICS environs background data:

Table 2-5: Summary of ICS Environs Background Data

MEDIA	MINIMUM	MAXIMUM	AVERAGE
Sediment Cs-137	0.04 pCi/g	0.11 pCi/g	0.07 pCi/g
Soil Cs-137 (Combined)	0.09 pCi/g	1.42 pCi/g	0.45 pCi/g
Soil Cs-137 (Woodland)	0.1 pCi/g	0.92 pCi/g	0.52 pCi/g
Soil Cs-137 (Hay Field)	0.1 pCi/g	0.55 pCi/g	0.38 pCi/g
Soil Cs-137 (Scrub Lands)	0.09 pCi/g	1.42 pCi/g	0.55 pCi/g
Water H-3	<743 pCi/L	<3126 pCi/L	<2024 pCi/L
Open Land Exposure (NaI_2)	5.9 μR/hr	13.6 μR/hr	11.4 μR/hr
Open Land Exposure (PIC)	7.18 μR/hr	9.34 μR/hr	8.22 μR/hr

c. Miscellaneous Background Survey Data

The University of Maine (Dr. C. T. Hess) performed a radiological soil and sediment background study prior to plant operations and reported the data in EPA Technical Note ORP/EAD-76-3. The study included analysis of nine soil samples, two marine sediment samples, and seven water samples collected in the vicinity of Maine Yankee prior to plant operations in during 1972.

The following is a summary of miscellaneous background survey data:

Table 2-6: Summary of Miscellaneous Background Survey Data

MEDIA	MINIMUM	MAXIMUM	AVERAGE
Sediment Cs-137	0.35 pCi/g	0.45 pCi/g	0.4 pCi/g
Soil Cs-137	0.8 pCi/g	4.96 pCi/g	2.04 pCi/g
Water H-3	<90 pCi/L	<40 pCi/L	<294 pCi/L

2.4.14 Waste Volumes and Activities

Table 3-8 summarizes projected activities associated with various sources of radioactive waste materials generated during decommissioning.

2.5 Continuing Characterization

The Characterization Report left a few survey areas unresolved with respect to the nuclides present and the extent or boundaries of contamination. Those areas and the plan for resolution were described in the Continued Characterization Plan, which was designed to obtain the following data needed to address the unresolved issues.

- Soil samples from the southeast fence area for bounding the extent of contamination
- Soil samples from the contractor's parking lot to confirm remediation and support construction of the ISFSI
- Soil samples from Bailey Point to confirm remediation
- PCC/SCC survey to bound the extent of contamination
- Condensate/Auxiliary Condensate survey to bound the extent of contamination
- Service Water survey to bound the extent of contamination

The Continued Characterization Plan and survey schedule were developed in order to obtain the data needed to resolve issues and support site specific dose pathway analysis. Areas for future sampling were also identified.

The new Spent Fuel Pool Decay Heat Removal System is contaminated. Remediation plans call for the system components to be removed and disposed of as radwaste. Once fuel has been transferred to the ISFSI, the area occupied by the SFP cooling system will be surveyed. Additional sampling of the circulating water discharge Forebay was performed to assure compliance with specific unrestricted use release criteria.

Characterization samples will continue to be collected and analyzed throughout the project to support the need for the most current and accurate radionuclide data.

2.5.1 Methods

Methods employed for continuing characterization were consistent with those described in Section 2.3 for site characterization. Any differences between the methods used by GTS and the methods employed for Continuous Characterization are noted within Section 2.3.

The work was performed under the guidance of a Decommissioning Work Order (DWO) and in accordance with approved procedures. In order to ensure comparable results, the instrumentation used was similar in design, function and sensitivity to that used during initial characterization.

2.5.2 Results

The range of residual radioactivity existing on surfaces and within soils and systems targeted for sampling during Continuing Characterization are summarized below. Detailed data including mean, maximum, and standard deviation are presented by survey package in Attachment D. The standard deviations calculated from CCS data may be replaced with more appropriate values calculated from post remediation or post demolition survey data. This section provides summary results from CCS. The current, resulting nuclide fractions are describe in Section 2.5.3.

a. Risk Areas

Several items were identified upon review of the GTS-Duratek Characterization Report as potential “risk areas” during decommissioning because they either did not equivocally make the determination as to radiological status or the extent of contamination was not bounded. Continued characterization provided resolution for the following risk items.

1. Determine the extent of soil contamination at the Southwest fence (CR0200, CR1000) - The East/West boundaries of the soil contamination were determined by gamma spectroscopy of soil samples. In addition, soil was sent for radiochemical analyses in order to confirm the ratio of radionuclides including the hard-to-detect nuclides.

2. Verify remediation of the "contractor parking lot" contaminated areas (CR1300) - Contrary to the GTS report and prior to continued characterization activities commencing, the State of Maine reported that the soil in the parking lot still contained Co-60 contamination after remediation. Soil survey results verified that there was residual soil contamination. The contaminated soil was excavated and disposed of as radwaste. A sample matrix was developed for post-remediation surveys and soil samples were taken and counted. Following this cleanup, the parking lot was determined to be successfully remediated based on gamma spectroscopy of soil samples and gamma scans taken over the affected soil area.

3. Verify remediation of the Bailey Point soil storage area (CR0500) - A sample matrix was developed and soil samples were taken and counted. Based on gamma spectroscopy results, the Bailey Point soil storage area was determined to have been successfully remediated.

4. Bound the extent of contamination in the PCC and SCC systems (CD1900) - PCC was opened and system internals were analyzed by gamma spectroscopy to determined the extent of contamination. The PCC system was found to be contaminated throughout, including the lube oil coolers of the diesel generators. The SCC system contamination was limited to one air conditioner feeding the control room (which had previously been in the PCC system but was later changed to SCC for train separation concerns) and both SCC pump suction elbows. The systems were labeled to show the extent of contamination.

5. Bound the extent of contamination in the Condensate/Aux Condensate systems (CD0100) - Samples were taken from the aux condensate piping, aux condensate receiver, and aux boilers. The samples confirmed that the aux condensate piping and aux boilers were contaminated. The system was labeled to show the extent of contamination.

6. Bound the extent of contamination in the liquid waste discharge line as it enters the Service Water pipe (CD0600) - Samples of the service water system were taken up stream from the point of entry of the liquid waste discharge pipe. The samples confirmed that contamination was limited to the area adjacent to the discharge pipe connection.

7. Additional surveys were designed and implemented to resolve reported positive count rate data on various systems or components in the Turbine Hall.

Unresolved data from the GTS Report were investigated. The activity in the water treatment plant (CD0200) was determined to be Naturally Occurring Radioactive Materials (NORM).

The data obtained during the Continued Characterization Surveys (CCS) are presented in Attachment D tables.

Data obtained during Characterization surveys are used to determine the nuclide profile for each media or material. If conditions arise during decommissioning which might affect the nuclide profile, additional sampling will be performed to verify the nuclide profile of any affected medium..

b. Soils

Surface soil was sampled and analyzed for radionuclides during the initial site characterization. The radionuclides were detected in the top 15 cm of on-site soil in the survey areas encompassing the backyard. Additional data were collected during continued characterization to better establish nuclide profiles. The predominant plant-related, beta-gamma emitting radionuclides detected were H-3, Co-60, Ni-63 and Cs-137. Two sets of higher activity soil samples taken by GTS were composited and subjected to radiochemical analyses for the hard-to-detect nuclides. No TRUs were detected in the composites when analyzed with techniques giving MDAs of 0.01 pCi/g to 0.005 pCi/g. The actual soil nuclide profile is provided in Section 2.5.3.

During characterization a concern was raised about activity in the vegetative layer of soil. As a result, a comparison was performed by counting vegetation and the soil/root ball; there was little measurable activity in the vegetation. Future soil samples will include the surface soil layer but not the protruding vegetation.

Sub-surface soil has been sampled and characterized in areas in which there was knowledge or indication of contamination below 15 cm. The nuclide ratios were consistent with surface ratios. In addition, building sub-slab soil characterization will be performed during remediation and demolition to determine the presence and extent of any sub-slab contamination. Samples will be taken alongside foundation walls or through holes bored through the floor if necessary.

c. Systems and Components

Residual contamination on or in plant piping was the result of the deposition of both fission and activation products. Prior to and during characterization surveys, samples of process piping were obtained to determine which systems were contaminated and the current radionuclide profiles including the hard-to-detect nuclides. The bounds of the contaminated piping were not established initially so systems were opened and surveyed to define the bounds of contamination. Contaminated system components and piping will be removed and disposed of as radioactive waste.

Fe-55, Ni-63, Co-60 and Cs-137 made up 99 percent of the system activities determined during initial characterization. TRUs contributed less than 1 percent of the total activity. The major beta-gamma emitter detected in system materials was Co-60 with a range of activity of 1 to 715 pCi/g (MDAs were 0.03 to 5 pCi/g). No additional quantitative gamma analyses for systems or components were conducted during CCS.

d. Buried and Embedded Piping

Buried and embedded piping remaining after demolition will receive special surveys during the FSS. The nuclides and ratios in piping and contaminated components are consistent with those described in c above since the systems with embedded sections of contaminated pipe were the systems sampled during initial characterization. The nuclide profile is provided in Section 2.5.3.

e. Structures-Concrete

Concrete structures at elevations higher than 3 feet below grade will be demolished. Surfaces (at elevations below 3 feet below grade) will be decontaminated to the specified DCGL for unrestricted use criteria. (See Section 3 for details on building demolition.). Four radionuclides, Cs-137, Ni-63, Co-60 and H-3 comprise approximately 99 percent of the radioactivity on concrete surfaces. (Special consideration was given to trench and sump surfaces. See discussion in Section 2.5.3.)

Radioactivity found in the concrete shielding materials in containment was the result of both contamination and activation. Concrete cores were removed and analyzed in order to estimate the radioactivity levels and nuclide distributions of shielding materials. The predominant radionuclides present in structural (activated) concrete are H-3, Fe-55, Eu-152, C-14, and Co-60 (comprising approximately 98 percent of the activity in activated concrete).

Concrete cores were counted using both hand-held instruments and gamma spectrometers. This information, coupled with the radiochemical analytical data, were used to determine instrument total efficiency E_t values (reported in Section 5.5.2).

g. Summary of CCS Activities Since Submittal of Revision 0 of the LTP

Since the submittal of Revision 0 of the LTP, several confirmatory samples have been collected. Two floor trench concrete samples were taken and submitted for HTD analysis to confirm or rule out some nuclide outliers reported by GTS from a trench sample processed by another laboratory.

Three additional Containment Building floor samples and three PAB floor samples were taken to replace the cores consumed during analysis.

A portion of activated concrete with embedded rebar was sent for analysis on both the concrete and rebar to establish the hard-to-detect nuclide fraction. A comparison of the nuclide profile was made to activation analysis results prepared for MY activated material as well as to published activation data. The results compared favorably in both instances. A core from the in-core instrumentation (ICI) sump was extended to a depth of 22 inches in order to improve the activated concrete profile (variance of activity with depth). The depth profile will be used to plan remediation activities for the ICI sump area. The projected post-remediation activity remaining in the ICI sump area was used in the dose calculations described in Section 6.

Fire pond water samples were taken and analyzed for tritium and gamma emitters. The same was done for the reflecting pond and sediment from the pond was counted to well below environmental LLDs in order to show there were no plant-derived nuclides in the sediment.

A Containment foundation sump water sample was analyzed (including HTDs) to relatively low MDAs. Tritium was determined to be the sole nuclide present in the foundation drains and groundwater based on this analysis. (This finding was consistent with sump water monitoring results from the past years.)

Forebay sediment was sampled. Due to a lack of sample material, a composite sediment sample was collected from the Forebay that verified the GTS reported Co-60 sediment activity. Section 6.6.9 provides additional detail regarding sampling completed and plans to perform additional sampling during the second quarter of 2001.

Additional material background samples were also collected in order to get better sample population statistics.

The results of these additional samples were used with previous data to determine nuclide profiles for each medium or material. In addition, detailed analyses of concrete core data were performed to ensure that the data collected were truly representative of the contaminated concrete on site. The soil and activated concrete data were also re-evaluated to confirm earlier assumptions based on the data reported in revision 0 of the LTP.

2.5.3 Nuclide Profile

One of the purposes of Site Characterization is to establish the radionuclide profiles for the various contaminated media which provide dose to the critical group. Multiple samples were taken of each type of media in order to determine the nuclides present and their relative fractions to one another. These nuclide fractions are presented by media in the following sections.

a. Contaminated Concrete Surfaces

Multiple concrete cores were analyzed (including HTDs) in order to determine the nuclide profile for contaminated concrete surfaces. The majority of the potentially contaminated surfaces remaining will be concrete. Other contaminated material such as embedded pipe may also remain. The nuclide profile determined for contaminated concrete is assumed to apply to all contaminated materials. The sample results were averaged over the entire population and the individual samples compared for consistency. As might be expected, the data were somewhat varied depending on the concrete location, spill history, decontamination history, surface coating and age.

The nuclide fraction for contaminated material was established using each of the positively identified nuclides. The non-detected nuclides were assumed not to be present in the mixture. In order to ensure that the elimination of non-detected nuclides at their MDC levels would not significantly affect the results, a sensitivity analysis based on dose was performed. Dose rates were determined for each individual core, for the core average values and for the average of the fractions using all nuclides in the suite at their actual value or their reported MDA, then the analysis was repeated using only the detected nuclides.

Two of the nine cores (both containment floor trench samples) showed evidence of TRUs; however, the values were very near the analytical MDCs. Even so, the TRUs were included in the evaluation of the nuclide fraction. Upon closer examination, the nuclide fraction for the trench samples appeared distinctly different from the other concrete fraction. The trench had a slightly different history of nuclide contact than the floor surfaces in general. Most significantly, water had been drained directly to the trench during the machining of cobalt-containing thermal shield pins and other special evolutions. Based the sample results from the two trench cores and consideration of the operational trench history, additional sample data were obtained to confirm the non-trench data. From that data, a separate nuclide fraction for the trenches was developed. As discussed Section 6.7, a separate DCGL for trenches was also established.

The data variability for the concrete cores was analyzed on the basis of dose. The significance of any identified variability was judged on its effect on the resulting dose. (See Attachment 2F for detailed discussion of the data analysis.)

Table 2-7 gives the nuclide fraction for contaminated surfaces that was selected based on the analysis of the characterization data determined by the "average of the fractions" method and decayed to 1/1/2004. Table 2-8 gives the nuclide fraction for trenches.

Table 2-7
Decayed Nuclide Fractions
Contaminated Concrete Surfaces

Nuclide	Fraction (as of 1/1/2004)
H-3	2.36E-2
Fe-55	4.81E-3
Co-57	3.06E-4
Co-60	5.84E-2
Ni-63	3.55E-1
Sr-90	2.80E-3
Cs-134	4.55E-3
Cs-137	5.50E-1

Table 2-8
Nuclide Fractions for Contaminated Concrete Trenches

Nuclide	Fraction (as of 1/1/2004)
H-3	5.04E-3
Fe-55	1.89E-3
Mn-54	4.32E-4
Co-57	4.80E-4

Table 2-8
Nuclide Fractions for Contaminated Concrete Trenches

Co-60	5.64E-1
Ni-63	5.17E-2
Sr-90	3.51E-3
Sb-125	3.15E-3
Cs-134	1.77E-3
Cs-137	3.66E-1
Pu-238	3.88E-5
Pu-239/240	4.14E-5
Pu-241	1.56E-3
Am-241	9.98E-6
Cm243/244	1.46E-6

b. Activated Concrete

Activated nuclide ratios were found to be consistent with published values. The major variation with activated concrete was a decrease in total activity with depth in the material as shown by two deep core profile samples. This property can be used to determine the depth of remediation needed. There was also a local effect on nuclide activity and ratio in the area immediately surrounding rebar contained within the concrete.

Two highly activated concrete samples were analyzed for HTDs. The hard to detect nuclides showed the same level of consistency as the gamma emitters when compared to published values (NUREG/CR-3474). The nuclide fraction for the activated concrete was established using each of the positively identified nuclides. The non-detected nuclides were assumed not to be present in the mixture. In order to ensure that the elimination of non-detected nuclides at their MDC levels would not significantly affect the results, an analysis based on dose contribution was performed. Annual dose rates were determined for each nuclide at its actual reported value or its MDC, then the analysis was repeated using only the actual reported values of the detected nuclides. Those nuclides included in the dose analysis at their MDC values were shown to contribute less than 10 percent of the annual dose from the pathway analyzed. Table 2-9 gives the nuclide fraction for activated concrete and rebar decayed to 1/1/2004.

Table 2-9
Activated Concrete Nuclide Fractions

	Concrete as of 1/2004	Rebar as of 1/2004
Nuclide	**Fraction**	**Fraction**
H-3	0.647	-------
C-14	0.058	-------
Fe-55	0.124	0.910
Ni-63	0.007	0.006
Co-60	0.040	0.084
Cs-134	0.0084	--------
Eu-152	0.111	--------
Eu-154	0.009	--------

Table 2-10 shows the activity measured a function of depth in the deep core sample

Table 2-10
Activated Concrete: Deep Core Sample Activity Profile

Depth (in)	Activity (pCi/g)	Depth (in)	Activity (pCi/g)
0 - 0.5	1028	10.75 - 11.5	87
0.5 - 1.0	828	11.5 - 12.25	23
1.0 - 1.5	845	12.25 - 13.0	23
1.5 - 3.5	Not Analyzed*	13.0 - 13.75	17
4.0 - 4.75	771	13.75 - 14.5	14
4.75 - 5.5	329	14.5 - 15.25	14
5.5 - 6.25	534	15.25 - 16.0	11
6.25 - 7.0	365	16.0 - 16.75	7
7.0 - 7.75	290	16.75 - 17.5	6
7.75 - 8.5	233	17.5 - 18.25	6
8.5 - 9.25	206	18.25 - 19.0	1
9.25 - 10.0	182	19.0 - 20.0	1
10.0 - 10.75	103		

*Not analyzed because of initial focus of the sample analysis on the top 1.5 inches of the sample

c. Contaminated Soil

Soil from the areas with the highest contamination levels (RWST and PWST areas) were composited and analyzed for nuclide content including HTDs. Since the samples used for the composites were very dry, archived soils, no tritium analyses were made. However, tritium analyses were performed on soil samples from an adjacent area.

The nuclide fraction for the contaminated soil was established using each of the positively identified nuclides. The non-detected nuclides were assumed not to be present in the mixture. In order to ensure that the elimination of non-detected nuclides at their MDC levels would not significantly affect the results, an analysis based on dose contribution was performed. Annual dose rates were determined for each nuclide at its actual reported value or its MDC, then the analysis was repeated using only the actual reported values of the detected nuclides. Those nuclides included in the dose analysis at their MDC values were shown to contribute less than 10 percent of the annual dose from the pathway analyzed.

The soil profile given in Table 2-11 is used for both surface (within 15 cm of the surface) and deep (below 15 cm of the surface) soils. The soil fractions were decayed to 1/1/2004.

Table 2-11
Soil Nuclide Fractions

Nuclide	Fraction as of 1/2004
H-3	0.053
Ni-63	0.048
Co-60	0.009
Cs-137	0.890

d. Groundwater and Surface Water

Samples were taken of the groundwater (containment foundation sump) and the surface water sources (fire pond and "reflecting pond"). The samples were analyzed for gamma emitters and HTDs. Since the samples did not contain much residual activity, long count times were used to achieve low MDAs. The only nuclide detected in either source of water was tritium. The surface water tritium is naturally occurring and the containment foundation sump tritium is likely the result of RWST and PCC leaks. The nuclide fraction for both water sources is given in Table 2-12.

Table 2-12
Ground and Surface Water Tritium Concentration

Nuclide	Fraction
H-3	1.000

e. Forebay Sediment

Multiple sediment samples were taken from the forebay and composited. The samples were analyzed for gamma emitters and HTDs. The nuclide profile decayed to 1/1/2004 for sediment is shown in Table 2-13. Additional characterization will be performed on the forebay sediment. The final nuclide profile will be determined after this characterization is complete.

Table 2-13
Forebay Sediment Nuclide Concentrations

Nuclide	Fraction (as of 1/1/2004)
Fe-55	0.165
Ni-63	0.233
Co-60	0.567
Sb-125	0.005
Cs-137	0.030

The radionuclide profiles for contaminated concrete, activated concrete, and soil listed in Tables 2-7 and 2-8, 2-9, and 2-11 respectively, were determined using representative data. These profile results do not rule out the possibility of taking additional samples of these media as decommissioning progresses and as conditions warrant.

Note: If radionuclide profiles are revised, the revised profiles will be provided to the NRC and the State of Maine at least 30 days prior to their use.

2.5.4 Background Determination

The residual radioactivity of a survey unit may be compared directly to the DCGL; however, some survey units will contain one or more radionuclides which are also contained in background. In order to identify and evaluate those radionuclides, background areas have been established which contain only background levels of the radionuclides of interest. These background areas were chosen because they were similar in physical, chemical, geological and biological characteristics to the survey units.

a. Soils

Soil samples were taken from the non-impacted areas and analyzed in order to establish general soil background levels. If background "reference" area measurements are required for the Final Survey Program, the reference area measurements will be collected in accordance with the methods described in Section 5 and the applicable approved procedures. The samples showed mean Cs-137 levels of 0.2 to 0.5 pCi/g depending on whether the soil had been disturbed or not. The more undisturbed the soil is, the higher the background Cs-137 may be (e.g. Knight Cemetery, Eaton Farm, values reported in Attachments A & B). The naturally-occurring uranium isotopes (U-234, U-235, and U-238) were present in expected amounts. Uranium is naturally occurring, not plant derived. These nuclides are not included in the Soil Mixture Nuclide Fraction listed in Table 2-11 above. Sr-90 was not detected at or above a MDC of 0.4 pCi/g.

b. Structures

Background measurements were taken on structural materials during initial characterization in order to estimate the contribution of background activity to the total measurement value. The same types of detectors will be used for FSS as were used during characterization. Background values for structural materials using these detectors are shown in Table 2-14.

Table 2-14
Structural Material Backgrounds

Background Counts per Minute		
Materials	**43-68 Proportional Detector - 126 cm^2**	**SHP-360 G-M Pancake Detector - 15.5 cm^2**
Painted Cinder Block	296**	70**
Wood	301**	57**
Ambient	319**	65**
Steel	277*	46*
Carpet	339**	68**
Floor Tile	359*	62*
Ceiling Tile	439*	73*
Bare Cinder Block	394**	79**
Painted Concrete	392*	74*
Bare Concrete	433*	76*
Asphalt	559*	99*
Granite	566**	128**
Porcelain	607**	116**
Brick	716*	118*

* Average of twenty-five one minute static counts taken in the scaler mode.
**Average of ten one minute static counts taken in the scaler mode.

The 43-68 proportional detector will generally be used for surface contamination measurements because of its sensitivity, larger detection area and lower MDC. SHP-360 will only be used where a measurement can not be taken with a 43-68 detector.

2.6 Summary

2.6.1 Impact Of Characterization Data On Decontamination And Decommissioning

Characterization data confirmed what was known about the MY site in terms of the level and extent of radioactive contamination. A major portion (700 acres) of the site met the classification of non-impacted. Primary systems and structures were found to be contaminated to expected levels. Non-RA systems and structures were found to be free of contamination except as previously stated.

There were minimal or no changes in either waste volumes or waste activity values following the performance of site characterization.

The data compiled are sufficient to project schedules and waste volumes, evaluate decontamination techniques, perform dose assessments and evaluate any safety or health issues affecting workers on site.

The HSA and characterization measurement results are sufficient to meet the objectives listed in Section 2.1 and demonstrate compliance with the guidance contained in Regulatory Guide 1.179 and NUREG-1700. The more than 19,000 measurements provide sufficient data to determine the radiological status of the site and facility as well as identify the location and extent of contamination outside the RA. The radionuclide analyses performed were sufficient to estimate the source term and isotopic mixture (based on the achieved standard deviation of the data). The analysis results also provide sufficient information to support dismantlement, radioactive waste disposal, decommissioning cost estimates and remediation decision making processes. The source term information was also suitable for instrument selection. The radiological data were acceptable to develop the necessary quality assurance methods for sample collection and analysis. The data obtained during characterization support dose assessment and FSS design.

2.7 References

2.7.1 NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual, (MARSSIM), 1997.

2.7.2 10 CFR.50.75, Reporting and Recordkeeping for Decommissioning Planning.

2.7.3 Continuing Characterization Plan.

2.7.4 CCS Quality Control.

2.7.5 Corrective Action Program Interface.

2.7.6 QA Records Management System.

2.7.7 Radiation Protection Program Assessment.

2.7.8 Selection, Training and Qualification of Radiation Protection Personnel.

2.7.9 Maine Yankee RCRA Closure Plan

2.7.10 NUREG-1507, Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions.

2.7.11 NUREG-1700, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans. (Draft)

2.7.12 Regulatory Guide 1.179, Standard Format and Content of License Termination Plans for Nuclear Reactors.

2.7.13 NUREG/CR-3474, Long-Lived Activation Products in Reactor Materials.

2.7.14 GTS Duratek, "Characterization Survey Report for the Maine Yankee Atomic Power Plant," Volumes 1-8, 1998.

ATTACHMENT 2A

Non-Impacted Area Assessment

ASSESSMENT OF THE MY SITE WEST AND NORTH OF BAILEY POINT FOR CLASSIFICATION AS NON-IMPACTED

2A. 1 Introduction

One aspect of the FSS Plan is the proper classification of areas within the site. Areas must be classified as either: Impacted, Class1, Class 2, or Class 3; or Non-impacted. Non-impacted areas are defined in NUREG-1575 (MARSSIM) as areas that "have no reasonable potential for residual contamination, no radiological impact from site operations and are typically identified during the Historical Site Assessment." The MY Historical Site Assessment (HSA) did not classify any areas within the site but it did provide data which could be used in conjunction with other information to classify areas. The HSA was not and will not be solely relied upon to make any classification, remediation or survey decision. The source term was well understood through previous Part 61 analysis. The potential pathways for this source term to potentially affect any offsite areas are well understood, described in the Off Site Dose Calculation Manual and monitored on a routine basis.

2A.2 Area Description

Approximately 700 acres of the MY site are found to the West of Bailey Cove, North of the access road (Ferry Road) and bounded by Back River to the east. The land is generally located beyond the 2000 foot exclusion zone established under the requirements of 10 CFR 100. As such, the area has been open and accessible to the general public and is bounded by residential land owners.

The referenced area consists of open fields, woodland and some shoreline property which has been uninhabited and unfarmed since plant construction started in 1968. The geology and hydrology of the area has been described in detail in the MY FSAR and is physically similar to the operating area of the site itself except for there being little or no surface soil disturbance (except for the ash pit and the ash pit access road). Structures in the area generally predate the construction of the plant.

The meteorology of the area has been characterized in detail in terms of annual precipitation, prevailing winds and stability class. Average annual precipitation exceeds the US average. Prevailing winds are from the South but a sea breeze blows East to West.

2A.3 Historical Site Assessment

The land areas under consideration are approximately 0.25 miles or more from the Reactor Building and process buildings. No radioactive material was used or stored beyond the peninsula of Bailey Point. License restrictions and administrative controls have been in place since power operations began in 1972 to prevent unauthorized removal of radioactive material from the owner controlled area. Planned offsite releases of radioactive material were limited to the permitted effluent releases (which were kept ALARA by process controls) and radioactive solid waste which was shipped to licensed burial sites. The HSA documented approximately 120 actual or potential events involving unplanned releases of radioactive material or hazardous material during the 25 year operating history of the plant. Of these events, about two thirds involved or potentially involved radioactive material. Based on a review of the documentation assembled in the HSA, none of these events would have resulted in residual contamination of the area under consideration. Therefore, there is no reasonable potential for residual contamination in the area.

2A.4 Radiological Environmental Monitoring Program

A Radiological Environmental Monitoring Program (REMP) was instituted prior to operation of the plant and continues to the present time. Environmental measurements taken have included thousands of gamma dose rates, hundreds of air and water samples, and hundreds of food stuff and surface vegetation samples. The key indicators of radiological impact in the area of concern are TLD measurements, air samples, water samples, vegetation samples, food crop samples and sediment samples.

TLD measurements have shown no difference in dose rates between the area under discussion and the control areas further from the site. Bailey Farm well water had slightly lower tritium levels on average than the water supplies in the Wiscasset area. Precipitation tritium levels at local sampling stations (Eaton and Bailey Farms) were similar to the control station levels. Fruits and vegetables sampled at the Bailey Farm showed the presence of only K-40 and fallout-produced Cs-137. Grasses sampled at the Eaton and Bailey Farms showed only natural K-40 and fallout-produced nuclides during periods of atmospheric testing. Initial soil samples had Cs-137 at levels consistent with published values for fallout activity. Samples taken during the intervening period had Cs-137 levels consistent with that which should have resulted from the decay of the initial 1970 sample activity. No radionuclides of plant origin were detected in these areas.

2A.5 Special Surveys And Reports

The HSA and other sources document samples (or measurements) of radiation and radioactive materials taken in the area in question. Pressurized ion chamber readings, TLD measurements, soil samples and even a “fly over” dose rate survey have documented radiation levels in the area similar to, or slightly less than, those measured in pre-operational surveys. The slight decline in levels is likely due to decreased levels of fallout-produced Cs-137 (Aerial Radiation Measurement Study, 1974 and University of Maine, 1974 and 1997). Some anomalous Cs data for Knight Cemetery, Eaton Farm and Foxbird Island can be understood in light of normal spacial variability in activity related to differences in sampling locations and the relatively undisturbed nature on some of these locations. Table 2A-7, “Alternate Table of Cs-137

Activity," shows very consistent results and the impact of decay when 1970 and 1997 data are presented. It is not surprising that some of the Cs data increased with time up to 1974 since atomic weapon atmospheric testing was still being conducted up to 1974.

Based on NUREG-1575 guidance, classification of an area as "not impacted" can be made solely on the Historical Site Assessment. Rather than rely solely on the HSA, the area in question was subjected to site characterization surveys. During 1997 and 1998, GTS performed site characterization measurements in the area which included gamma dose rates determined by pressurized ion chamber and micro R meter, soil samples and "drive around" surveys using a vehicle-mounted 1.5"x 3"x 33" scintillation detector. The characterization surveys (PIC and "drive around") in the area produced one area with an elevated radiation level. Upon investigation, the elevated reading was found to be due to local increase in naturally occurring radiation. Approximately 150 soil samples taken throughout the area showed only background levels of radioactive material in quantities slightly less than those reported in the 1972 pre-operational studies in this area which is consistent with the decay of the fallout-produced activity.

2A.6 Conclusion

Based on the evaluation of the historical use of the area, the lack of use or storage of radioactive material in the area, the Historical Site Assessment findings, the REMP results and the results of the site characterization surveys, the area to the West of Bailey Cove and North of Ferry Road within the land owned by MY has been classified as non-impacted.

The area lends itself to use as a background reference area for soil samples and may be used as such during the FSS. Random sampling of soil in order to establish background activities would be performed in this reference area but no systematic sampling as required by MARSSIM for impacted areas would be performed.

Table 2A-1 RADIOLOGICAL ENVIRONMENTAL DATA				
TLD DATA (Mean Value in μR/hr)				
Data Source	**Inner Ring**	**Outer Ring**	**Control**	**Period; # locations**
MY	11.8	12.0	11.9	1970-1972 n=9
MY	7.1	7.4	7.8	1990-1997 n=28
Univ. of Maine	8.2	8.6	9.3	1971-1996 n=87

Table 2A-2 Pressurized Ion Chamber Data (μR/hr)				
Data Source	**Location**	**1971**	**1996**	**1998**
Univ. of Maine	Bailey House	9.5	8.8	
Univ. of Maine	Eaton Farm	9.5	9.3	
Univ. of Maine	Westport	11.4	9.1	
Univ. of Maine	Knight Cemetery		8.7	
Univ. of Maine	Long Ledge		9.0	
GTS	Merrymeeting Airfield			Mean=8.2 Range: 7.2-9.8 n=300

Table 2A-3
Soil Cs-137 (pCi/g)

Sample Location	1970 MY	1972 MY	1974 MY	1996 MY	1997 GTS Characterization
Bailey House	0.64	1.67	1.8	0.4	0.21; n=30
Bath	0.66				
Dresden	0.58				
Eaton Farm	0.53	0.87	2.5	0.09	0.46; n=60
Edgecomb	0.48				
Foxbird		0.35		0.48	
Knight Cemetery		4.96		2.42	
Long Ledge		0.80		0.38	
Harrison’s	0.52				
Mason Station	0.68				
Montsweag Dam	0.42				
Westport	0.56	1.11		1.03	
North of Ferry Road					0.39; n=60
Merrymeeting Airfield					0.35; n=60
Shoreline					0.20; n=30
Mean Value	0.56	1.63	2.15	0.80	0.32

Table 2A-4 Surface & Well Water Data	
Sample Location	**(Mean H-3 pCi/L) 1977-1984**
Bailey House	235
Montsweag Dam	276
Morse Well	187
Biscay Pond	297
Wiscasset Reservoir	278

Table 2A-5 Precipitation Data	
Sample Location	**(Mean H-3 pCi/L) 1977-1982**
Bailey House	416
Eaton Farm	417
Westport	422
Dresden	397

Table 2A-6
Air Particulate Data (Mean Gross Beta Activity, pCi/m^3)

MY Pre-Operational Data			
1970			0.12
1971			0.12
1972 Jan-Jun			Zone I=0.06, Zone II=0.07
Univ. of Maine 1981-1997		MY 1988-1998	
Wiscasset	0.02*	Montsweag	0.021
Augusta	0.02*	Bailey House	0.020
		Mason Station	0.020
		Westport	0.021
		Dresden	0.022

* Values estimated by graph. Individual data not available.

References: MY data were taken from the REMP Reports for the time periods listed or the GTS Characterization Report.

University of Maine data were taken from “A Radiological Survey of the Area Surrounding the MY Nuclear Plant”, March 1997.

Table 2A-7 Alternate Table of Cs-137 Activity		
Soil Cs-137 (pCi/g)		
Sample Location	**1970 MY**	**1997 GTS Characterization**
Bailey House	0.64	0.21; n=30
Bath	0.66	
Dresden	0.58	
Eaton Farm	0.53	0.46; n=60
Edgecomb	0.48	
Harrison's	0.52	
Mason Station	0.68	
Montsweag Dam	0.42	
Westport	0.56	
North of Ferry Road		0.39; n=60
Merrymeeting Airfield		0.35; n=60
Shoreline		0.20; n=30
Mean Value	0.56	0.32

ATTACHMENT 2B

Characterization Data

Table 2B-1
Group A
Radiological Characterization Results For Affected Structures and Surfaces

	Direct Beta DPM/100 cm^2			Removable Beta DPM/100 cm^2			Removable Alpha DPM/100 cm^2			Exposure Rate MicroR/hr		
Package	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean Minimum Detectable Exp Rate (MDER)***	**Maximum**	**Std. Dev.**
A0100 Cont.El -2 Ft	81,976 (30,453)	1,970,974	259,134.5	296 (33)	4,282	598.7	0.0 (8.4)	2.4	0.5	2,375 (15)	4,065	816
A0200 Cont. El 20 Ft	62,970 (16,277)	2,238,614	247,399.2	2,388 (35)	128,734	13,577.2	0.7 (9.7)	7.3	1.6	887 (15)	1,961	463
A0300 Cont. El 46 Ft	38,444 (16,058)	345,960	55,889.2	1,469 (33)	31,054	3245.7	0.2 (8.7)	5.8	1.1	499.5 (15)	2,408	387.5
A0400 Fuel Bldg El 21 Ft	6,815 (12,436)	312,939	32,365.4	38.4 (32)	879	106.2	-0.1 (8.5)	1.8	0.6	706.6 (15)	2,901	649.7
A0500 DWST	438 (2,322)	2,659	792.6	4.9 (32)	20.3	7.0	0.1 (8.4)	3.9	1.0	14.0 (15)	14.6	0.9
A0600 PAB El 11 Ft	1,106 (13,168)	32,328	7513.5	5.2 (32)	32.3	8.0	-0.1 (8.5)	3.9	0.7	1,100 (15)	3,477	827
A0700 PAB El 21 Ft	460 (15,837)	25,000	4655.1	5.9 (32)	51.5	9.7	-0.2 (7.7)	1.8	0.3	581 (15)	4,068	950
A0800 PAB El 36 Ft	508 (18,042)	14,073	2166.5	5.9 (34)	94.2	11.0	0.1 (7.0)	2.0	0.6	187 (15)	769	182
A0900 RA Svc Bld	699 (1,970)	18,955	2927.8	9.2 (34)	251	26.6	-0.6 (8.2)	3.9	0.6	42 (15)	501	78
A1100 LLWSB	852 (17,886)	74,216	6023.3	0.3 (38)	35.8	7.0	0.1 (8.1)	4.1	0.8	334 (15)	3,563	752
A1200 RCA Storage	73,939 (26,286)	2,233,580	379,578.7	128.7 (37)	2,073	323.1	-0.1 (8.6)	1.8	0.6	2,162 (15)	12,389	2,864

Table 2B-1
Group A
Radiological Characterization Results For Affected Structures and Surfaces

	Direct Beta DPM/100 cm^2			Removable Beta DPM/100 cm^2			Removable Alpha DPM/100 cm^2			Exposure Rate MicroR/hr		
Package	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean Minimum Detectable Exp Rate (MDER)***	**Maximum**	**Std. Dev.**
A1300 Equip Hatch	27.5 (600)	720.5	255.1	4.9 (35)	19.8	7.6	-0.1 (7.8)	1.9	0.5	27.1 (15)	122.7	33.7
A1400 Pers Hatch	350.2 (2198)	6,758	1379.9	47.1 (35)	657.5	126.8	-0.2 (7.8)	1.9	0.3	47.5 (15)	180.2	41.2
A1500 Mech Pen	214.9 (661)	3,678	734.3	4.4 (38)	23.5	7.7	-0.2 (8.4)	3.9	0.6	9.4 (15)	14.0	2.6
A1600 Elec Pen	-138.0 (654)	557.1	269.7	1.9 (37)	18.2	6.9	0.0 (7.7)	1.8	0.6	12.7 (15)	14.0	1.2
A1700 Spray Bld	83,249 (24,797)	4,968,088	431,253.4	177.5 (37)	19,727	1445.2	0.0 (7.2)	2.0	0.4	1,598 (15)	9,041	2,124
A1800 Aux Feed Pump	147.5 (2,019)	1,278	422.4	2.3 (37)	36.6	11.3	-0.1 (7.7)	1.8	0.5	18.9 (15)	34.9	7.1
A1900 HV-9	130.6 (6318)	2,563	725.3	0.6 (36)	24.6	7.0	-0.1 (8.2)	1.8	0.6	90.6 (15)	182.9	45.9
A2100 RWST	3,602 (21,587)	54,719	13,158.9	2.7 (38)	72.4	13.5	0.0 (8.4)	1.8	0.7	687.5 (15)	1,078.4	374.0
A2200 BWST	7,269 (21,255)	43,189	10,833.4	7.1 (36)	73.2	16.9	-0.1 (8.2)	1.8	0.6	667.6 (15)	1,197	246.6
A2300 PWST	668 (2,780)	3,258	942.1	5.8 (32)	27.4	7.1	0.1 (8.4)	1.8	0.8	N/A	N/A	N/A
A2400 Test Tks	955.5 (1438)	4,300	1062.8	3.5 (36)	30.7	7.3	0.4 (8.2)	5.8	1.3	N/A	N/A	N/A

* NOTE: MDER values are for the instrument in a low background area.

Table 2B-2
Group B
Unaffected Structures and Surfaces, Including Structural Background Survey

	Direct Beta DPM/100 cm^2			Removable Beta DPM/100 cm^2			Removable Alpha DPM/100 cm^2			Exposure Rate microR/hr		
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean Minimum Detectable Exp Rate (MDER)***	**Max**	**Std. Dev.**
B0100 Turb El 61Ft	26.7 (636)	653.7	246.9	3.5 (17)	19.1	4.8	-0.3 (7.6)	4.8	0.9	9.0 (15)	15.2	1.9
B0200 Control Rm (Old)	215.8 (616)	1054.2	384.1	4.1 (16)	25.8	5.4	-0.5 (7.6)	2.0	0.7	10.2 (15)	12.5	1.1
B0300 MCC	-91.0 (701)	552.5	299.7	1.9 (17)	11.7	4.8	-0.2 (7.3)	2.1	0.9	12.2 (15)	14.9	2.0
B0400 Fire Pmp	10.1 (610)	840.1	351.2	2.6 (32)	18.4	5.3	-0.6 (8.2)	0.7	0.4	11.2 (15)	12.8	1.6
B0500 Turb El 21Ft	62.1 (649)	8613.8	752.2	2.8 (17)	203.4	15.8	-0.4 (7.3)	2.1	0.7	8.6 (15)	17.3	2.8
B0600 Turb El 39 Ft	48.2 (603)	2031.4	332.9	2.9 (17)	30.0	6.1	-0.1 (7.3)	3.5	0.9	6.3 (15)	13.7	2.9
B0700 Svc. Bld. Non-RCA	80.0 (821)	1621.5	411.1	2.8 (32)	19.9	5.0	-0.1 (8.4)	2.4	0.7	12.5 (15)	26.0	3.5
B0800 FOSB	-82.7 (587)	451.4	286.0	5.5 (16)	19.9	6.1	-0.2 (6.7)	0.9	0.5	8.4 (15)	9.9	0.8
B0900 EDGs	-176.9 (683)	411.9	209.8	4.3 (16)	19.9	5.6	-0.1 (6.7)	0.9	0.6	10.8 (15)	13.1	1.6

Table 2B-2
Group B
Unaffected Structures and Surfaces, Including Structural Background Survey

	Direct Beta DPM/100 cm^2			Removable Beta DPM/100 cm^2			Removable Alpha DPM/100 cm^2			Exposure Rate microR/hr		
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean Minimum Detectable Exp Rate (MDER)***	**Max**	**Std. Dev.**
B1000 Aux Boiler	183.4 (679)	1309.7	492.6	3.4 (16)	16.5	5.9	-0.2 (6.7)	2.4	0.7	9.2 (15)	10.5	0.9
B1100 Circ Water	-333.9 (699)	672.7	300.5	1.8 (16)	11.4	4.1	0.0 (6.7)	2.4	0.9	8.5 (15)	10.8	1.3
B1200 Admin Bld	293.1 (686)	1628.2	431.9	4.3 (16)	14.8	5.1	0.0 (6.7)	2.4	0.9	13.3 (15)	15.2	1.5
B1300 WART	-146.3 (666)	1163.8	542.5	2.6 (16)	13.1	4.5	0.1 (6.7)	2.4	0.9	11.1 (15)	12.9	1.2
B1400 Info Ctr	295.3 (678)	1928.8	325.6	2.1 (16)	21.5	5.0	0.1 (6.7)	3.8	1.0	13.4 (15)	16.8	1.3
B1500 Warehse 2	96.1 (566)	539.0	212.4	0.6 (18)	19.4	5.2	-0.3 (7.3)	2.1	0.8	10.3 (15)	15.1	1.4
B1600 Trng Annex	-13.5 (657)	708.2	256.1	1.6 (18)	17.7	4.8	-0.2 (7.3)	2.1	0.8	17.8 (15)	23.8	3.5
B1700 Staff Bld	129.4 (727)	952.9	279.5	-1.0 (18)	14.4	4.5	-0.4 (7.3)	3.5	0.7	14.2 (15)	23.2	3.3
B1800 Spare Gen Bld	-39.8 (548)	341.9	176.6	0.1 (18)	9.3	4.6	-0.5 (7.3)	0.7	0.5	N/A	N/A	N/A
B1900 Bailey House	612.3 (682)	6523.7	1595.1	0.3 (18)	11.0	6.1	-0.4 (7.3)	0.7	0.6	9.4 (15)	16.1	3.6

Table 2B-2
Group B
Unaffected Structures and Surfaces, Including Structural Background Survey

	Direct Beta DPM/100 cm^2			Removable Beta DPM/100 cm^2			Removable Alpha DPM/100 cm^2			Exposure Rate microR/hr		
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean Minimum Detectable Exp Rate (MDER)***	**Max**	**Std. Dev.**
B2000 Bailey Barn	-96.6 (592)	306.5	187.3	1.1 (18)	9.3	4.6	-0.4 (7.3)	0.7	0.6	9.2 (15)	10.6	0.8
B2100 Lube Oil Storage	8.7 (630)	610.4	240.7	0.2 (18)	7.6	4.3	-0.5 (7.3)	0.7	0.6	8.8 (15)	10.9	1.8
B2200 Cold Shop	139.4 (604)	762.3	317.9	0.6 (18)	7.6	4.0	-0.5 (7.3)	0.7	0.5	8.0 (15)	9.0	0.9
B2300 Cable Vault	-23.4 (632)	275.3	195.1	0.5 (18)	21.3	5.0	-0.3 (6.9)	2.3	0.6	13.8 (15)	17.1	1.9
B2400 Staff Tunnel	19.2 (779)	575.6	359.6	3.8 (18)	18.0	6.7	-0.1 (6.9)	3.7	0.9	20.3 (15)	24.2	2.3

* NOTE: MDER values are for the instrument in a low background area.

Table 2B-3
Group C
Radiological Characterization Results For Affected Systems

	Direct Beta DPM/100 cm^2			Removable Beta DPM/100 cm^2			Removable Alpha DPM/100 cm^2			Exposure Rate microR/hr			Tritium DPM/ 100 cm2
Package	**Mean (MDC)**	**Max**	**Std. Dev**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean Minimum Detectable Exp Rate (MDER)***	**Max**	**Std. Dev.**	**Mean**
C0100 PASS	N/A	N/A	N/A	77,858 (5000)	300,000	126,236	1.5 (8.4)	8.0	3.7	1386 (15)	4161	1422.9	61.1 (39)
C0200 Waste Solid.	N/A	N/A	N/A	2344 (34)	4073	2069.9	-0.3 (8.4)	-0.3	0.0	23,333 (15)	219,340	53,199	399.9 (39)
C0300 Contain. Spray	N/A	N/A	N/A	25,185 (34)	39,530	14,366.8	11.5 (8.4)	24.7	11.5	2593 (15)	22,862	4192	18.4 (39)
C0400 ECCS	N/A	N/A	N/A	70,933 (5000)	200,000	111,776	3.3 (8.4)	5.9	3.0	4416 (15)	34,960	6025	1377.8 (139)
C0500 RHR	N/A	N/A	N/A	76,000 (5000)	180,000	91,476.8	N/A	N/A	N/A	4882 (15)	15,772	4112	23,617 (139)
C0600 Pri. Vent & Drains	N/A	N/A	N/A	50,585 (5000)	140,000	77,438	-0.2 (8.4)	0.0	0.2	165,583 (15)	1,326,311	325,892	548 (39)
C0700 SFP Cooling	N/A	N/A	N/A	13,693 (5000)	20,000	6466.2	3.4 (8.4)	10.1	5.8	829,672 (15)	16,945,540	2,924,669	31.0 (39)
C0800 Waste Gas	N/A	N/A	N/A	3251 (34)	6470	2854.0	-0.3 (8.4)	-0.3	0.0	3295 (15)	23,554	4,999.5	5825 (39)
C0900 Pzr.	N/A	N/A	N/A	213,333 (5000)	360,000	128,582	N/A	N/A	N/A	41,636 (15)	376,269	59,187	82,468 (139)
C1100 RCS	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	53,580 (15)	181,323	34,275	N/A

Table 2B-3
Group C
Radiological Characterization Results For Affected Systems

	Direct Beta DPM/100 cm^2			**Removable Beta DPM/100 cm^2**			**Removable Alpha DPM/100 cm^2**			**Exposure Rate microR/hr**			**Tritium DPM/ 100 cm2**
Package	**Mean (MDC)**	**Max**	**Std. Dev**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean Minimum Detectable Exp Rate (MDER)***	**Max**	**Std. Dev.**	**Mean**
C1200 Boron Recovery	N/A	N/A	N/A	53,766 (5000)	160,000	92,001.4	-0.2 (8.4)	0.0	0.2	1283 (15)	13,023	2078	19,515 (39)
C1300 CVCS	1907	3924.8	2074.1	29,197 (1316)	112,370	47,511.3	8.8 (7.8)	34.9	14.8	41,446 (15)	884,946	127,708	1057 (139)
C1400 Liq. Waste	N/A	N/A	N/A	1078 (35)	1403	289.4	1.2 (7.8)	3.9	2.4	91,689 (15)	935,068	166,593	1187 (39)
C1500 PAB Drains	N/A	N/A	N/A	1895 (35)	6002	2409.7	0.5 (7.8)	1.9	1.1	2059 (15)	10,306	2309	128.4 (38)
C1600 PAB Vent	5275 (1144)	16,837	6185.7	52.8 (35)	194	72.0	-0.1 (7.8)	1.9	0.6	492.4 (15)	3546	1007	-17.6 (38)
C1800 Contain. Vent	448,954 (15,606)	540,758	77,163.2	16,768 (5000)	80,000	35,348.1	1.1 (7.8)	3.9	1.8	802.4 (15)	2275	653	-3.4 (38)
C1900 S/Gs	N/A	N/A	N/A	266,667 (5000)	500,000	202,320	N/A	N/A	N/A	17,071 (15)	82,025	21,980	398.0 (139)

* NOTE: MDER values are for the instrument in a low background area.

Table 2B-4 -Group D Unaffected Systems												
	Direct Beta DPM/100 cm^2			Removable Beta DPM/100 cm^2			Removable Alpha DPM/100 cm^2			Exposure Rate microR/hr		
Package	Mean (MDC)	Max	Std. Dev.	Mean (MDC)	Max	Std. Dev.	Mean (MDC)	Max	Std. Dev.	Mean Minimum Detectable Exp Rate (MDER) *	Max	Std. Dev.
D0100 Condens.	66.7	2184.5	425.2	-0.5	14.6	5.1	-0.3	2.3	0.7	1.9 (15)	2.1	0.1
D0200 Water Treat.	1250.8 (1937)	26,046.3	4898.1	38.1 (16)	945.1	162.8	13.6 (7.6)	362.2	61.9	12.6 (15)	44.2	17.7
D0300 Potable Water	526.2 (1089)	2638.6	767.7	6.7 (16)	29.2	6.9	0.4 (7.6)	9.1	2.3	4.5 (15)	7.1	1.6
D0400 Sewer	384.8 (1088)	5657.1	1051.5	3.2 (36)	32.2	8.9	0.0 (8.2)	1.9	0.6	11.3 (15)	16.2	4.3
D0500 Circ Water	162.0 (587)	811.8	295.1	3.1 (15)	14.7	4.2	-0.1 (6.9)	5.1	0.9	3.7 (15)	17.2	5.1
D0600 Svc Water	38.0 (1687)	1013.9	347.9	197.5 (37)	3133.7	658.5	-0.2 (8.6)	1.8	0.5	N/A	N/A	N/A
D0700 Fire Prot.	-35.6 (1257)	1114.7	240.2	2.4 (17)	20.6	5.2	0.2 (6)	2.5	0.9	N/A	N/A	N/A
D0800 Lube Oil	66.0 (1681)	723.4	253.6	2.5 (17)	22.3	6.1	0.1 (6)	2.5	0.7	6.0 (15)	12.3	5.5
D0900 Comp. Air	3677.5 (6324)	104,589	14,456.3	27.0 (17)	685.2	95.1	0.4 (6)	6.8	1.4	N/A	N/A	N/A
D1000 Aux Boiler	446.0 (2606)	2723.9	730.5	12.3 (17)	114.8	21.8	0.0 (6)	2.5	0.8	7.1 (15)	20.1	5.3
D1100 S/G	270.8 (1347)	2664.1	1067.4	9.2 (17)	47.5	11.1	0.3 (6)	2.5	1.0	35.0 (15)	66.8	44.9

Table 2B-4 -Group D
Unaffected Systems

	Direct Beta DPM/100 cm^2			Removable Beta DPM/100 cm^2			Removable Alpha DPM/100 cm^2			Exposure Rate microR/hr		
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean Minimum Detectable Exp Rate (MDER) ***	**Max**	**Std. Dev.**
D1200 Main Steam	-9.2 (1002)	4598.7	649.0	0.8 (36)	59.6	9.3	-0.3 (8.2)	2.2	0.7	N/A	N/A	N/A
D1300 Aux Steam	667.3 (2382)	11,786.6	1963.4	1.9 (36)	19.4	6.5	0.0 (8.2)	2.0	0.5	162.8 (15)	435.1	218.9
D1400 Turb Control	-38.3 (839)	416.5	189.7	-0.9 (19)	20.9	6.3	-0.4 (7.1)	0.8	0.5	0.8 (15)	1.6	0.4
D1500 Steam Dump	-216.5 (677)	64.1	139.9	-0.8 (19)	10.8	4.1	-0.5 (7.1)	0.8	0.5	N/A	N/A	N/A
D1600 Main Feed	-0.3 (640)	453.9	160.8	-1.2 (19)	24.2	6.3	-0.4 (7.1)	2.2	0.6	2.0 (15)	5.4	2.2
D1700 EFW	-136.5 (2414)	851.3	347.6	0.9 (18)	21.0	5.3	-0.3 (7.1)	3.6	0.8	N/A	N/A	N/A
D1800 Htr. Drain, Extract	42.4 (1182)	1864.3	323.3	-2.7 (19)	9.1	3.8	-0.4 (7.1)	2.2	0.6	0.9 (15)	1.3	0.4
D1900 Comp Cooling	1168.0 (4385)	21,644.3	6616.3	5.2 (36)	38.0	10.7	-0.1 (7.2)	2.0	0.3	10.1 (15)	12.8	2.0
D2000 Vac Prim	24.8 (1256)	672.1	257.8	1.6 (18)	14.2	4.8	-0.3 (7.1)	2.2	0.8	N/A	N/A	N/A
D2100 Amertap	107.5 (1200)	1880.2	507.5	2.2 (18)	15.9	5.4	0.1 (7.1)	3.6	1.1	N/A	N/A	N/A
D2200 Sealing Steam	23.3 (1067)	582.0	237.8	0.2 (18)	10.9	4.2	-0.5 (7.1)	0.8	0.5	N/A	N/A	N/A

Table 2B-4 -Group D Unaffected Systems												
	Direct Beta DPM/100 cm^2			**Removable Beta DPM/100 cm^2**			**Removable Alpha DPM/100 cm^2**			**Exposure Rate microR/hr**		
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean Minimum Detectable Exp Rate (MDER) ***	**Max**	**Std. Dev.**
D2300 Aux DG	31.7 (645)	535.3	210.8	3.1 (36)	31.8	9.3	0.1 (7.2)	2.0	0.7	N/A	N/A	N/A
D2400 Chem Sample	35.2 (1617)	645.5	251.2	307.2 (35)	4861.3	995.8	0.3 (7.8)	6.0	1.4	N/A	N/A	N/A
D2500 HP Drain	132.2 (1048)	594.8	260.3	-0.1 (18)	7.5	4.7	-0.4 (7.1)	0.8	0.6	N/A	N/A	N/A
D2600 Envir	336.6 (535)	1257.1	400.1	3.7 (14)	12.9	3.9	0.6 (6.9)	3.9	1.3	N/A	N/A	N/A
D2700 Admin HVAC	74.3 (789)	643.3	276.3	5.2 (18)	32.8	8.5	0.3 (7.1)	2.2	1.1	8.0 (15)	8.0	0.0
D2800 Info Ctr Hvac	156.2 (702)	627.8	256.9	0.6 (18)	10.9	4.7	-0.5 (7.1)	0.8	0.5	N/A	N/A	N/A
D2900 Turb HVAC	142.4 (577)	445.4	161.5	4.6 (14)	33.1	5.9	0.3 (6.9)	3.9	0.9	N/A	N/A	N/A
D3000 Staff HVAC	262.9 (779)	1286.3	366.0	2.2 (18)	15.9	6.0	-0.1 (7.1)	2.2	0.9	N/A	N/A	N/A
D3100 Svc HVAC	5346.8 (1082)	87,565.8	19,067.0	80.0 (14)	1445.0	247.1	0.6 (8.5)	5.9	1.3	22.4 (15)	51.4	17.4
D3200 H2/N2	12,037.3 (3059)	125,317	36,307.5	104.5 (14)	828.9	245.4	0.6 (8.5)	9.9	2.3	N/A	N/A	N/A
D3300 Turb Sumps	433.1 (1091)	5800.9	1166.9	8.1 (32)	33.6	9.0	0.0 (8.4)	1.8	0.8	10.8 (15)	15.9	4.9

Table 2B-4 -Group D Unaffected Systems												
	Direct Beta DPM/100 cm^2			Removable Beta DPM/100 cm^2			Removable Alpha DPM/100 cm^2			Exposure Rate microR/hr		
Package	Mean (MDC)	Max	Std. Dev.	Mean (MDC)	Max	Std. Dev.	Mean (MDC)	Max	Std. Dev.	Mean Minimum Detectable Exp Rate (MDER) *	Max	Std. Dev.
D3400 LLWSB	457.0 (992)	3099.3	1300.0	7.1 (32)	27.4	8.7	0.1 (8.4)	6.0	1.3	N/A	N/A	N/A

* NOTE: MDER values are for the instrument in a low background area.

Table 2B-5 Group R Radiological Characterization Results For Affected and Unaffected Environs												
										Exposure Rate microR/hr		
Package	# Samples	# Positive Co-60	Mean Co-60 pCi/g	Max Co-60 pCi/g	# Positive Cs-137	Mean Cs-137 pCi/g	Max Cs-137 pCi/g			Mean	Maximum	Std. Dev.
R0100 RA Yard West	58	23	0.62	3.29	55	10.99	156.0	N/A	N/A	N/A	N/A	N/A
R0200 Yard East	35	12	0.28	1.94	33	4.88	133.0	N/A	N/A	N/A	N/A	N/A
R0300 Roof Drains	7	4	4.09	11.2	6	0.33	0.53	N/A	N/A	N/A	N/A	N/A
R0400 Shoreline	27	1	0.08	0.08	27	0.34	0.98	N/A	N/A	N/A	N/A	N/A
R0500 Bailey Pt.	45	0	0	0	44	0.38	1.09	N/A	N/A	13.27	19.83	1.49
R0600 Ball Field	32	0	0	0	3	0.04	0.06	N/A	N/A	11.92	13.68	0.63

Table 2B-5
Group R
Radiological Characterization Results For Affected and Unaffected Environs

										Exposure Rate microR/hr		
Package	**# Samples**	**# Positive Co-60**	**Mean Co-60 pCi/g**	**Max Co-60 pCi/g**	**# Positive Cs-137**	**Mean Cs-137 pCi/g**	**Max Cs-137 pCi/g**			**Mean**	**Maximum**	**Std. Dev.**
R0700 Constr. Debris	31	0	0	0	2	0.05	0.06	N/A	N/A	11.99	14.52	1.05
R0800 Admin. Parking	30	0	0	0	26	0.26	0.83	N/A	N/A	17.9	33.87	4.2
R0900 BOP	36	6	1.22	5.11	24	11.06	85.6	N/A	N/A	25.85	77.71	16.8
R1000 Foxbird Is	73	3	0.22	0.38	43	0.43	1.63	N/A	N/A	11.48	42.76	4.97
R1100 Roof Drains	15	0	0	0	3	0.07	0.09	N/A	N/A	N/A	N/A	N/A
R1200 LLWSB Yard	30	0	0	0	5	0.10	0.13	N/A	N/A	N/A	N/A	N/A
R1300 ISFSI	30	0	0	0	5	0.12	0.28	N/A	N/A	12.92	31.2	3.68
R1400 Shorelines	30	0	0	0	30	0.20	0.35	N/A	N/A	N/A	N/A	N/A
R1500 Ash Pit Rubble	30	0	0	0	9	0.08	0.21	N/A	N/A	11.34	12.63	0.63
R1600 Eaton Farm Land	60	0	0	0	59	0.46	1.43	N/A	N/A	12.07	17.8	2.06
R1700 Land North of Ferry Rd	60	0	0	0	50	0.47	1.55	N/A	N/A	9.65	13.74	1.56

Table 2B-5
Group R
Radiological Characterization Results For Affected and Unaffected Environs

										Exposure Rate microR/hr		
Package	**# Samples**	**# Positive Co-60**	**Mean Co-60 pCi/g**	**Max Co-60 pCi/g**	**# Positive Cs-137**	**Mean Cs-137 pCi/g**	**Max Cs-137 pCi/g**			**Mean**	**Maximum**	**Std. Dev.**
R1800 Bailey Farm Land	31	0	0	0	22	0.27	0.76	N/A	N/A	10.63	14.57	1.31
R1900 Bailey Cove	14	0	0	0	14	0.27	0.37	N/A	N/A	N/A	N/A	N/A
R2000 Diffuser	5	2	0.1	0.12	4	0.10	0.13	N/A	N/A	N/A	N/A	N/A
R2100 Warehse Yard	30	0	0	0	4	0.13	0.33	N/A	N/A	8.41	10.62	1.33
R2200 Backgrnd	62	0	0	0	62	0.35	1.4	N/A	N/A	11.37	13.59	1.26
R2300 SFP Substation	16	1	0.14	0.14	15	0.35	0.81	N/A	N/A	26.14	29.4	1.46
R2400 IT Duplicates	44	0	0	0	9	0.48	1.62	N/A	N/A	N/A	N/A	N/A

Table 2B-6
R2500 Investigation Package

Package	# Samples	# Positive Co-60	Mean Co-60 pCi/g	Max Co-60 pCi/g	# Positive Cs-137	Mean Cs-137 pCi/g	Max Cs-137 pCi/g
R0500 Bailey Pt	8	3	11,218.5	33,600.0	7	0.13	0.21
R0600 Ball Field	15	0	0	0	5	0.16	0.29
R0700 Construction Debris	40	0	0	0	3	0.04	0.06
R0800 Admin Parking Lot	15	0	0	0	14	0.17	0.33
R1000 Foxbird Is	10	0	0	0	7	0.13	0.21
R1300 ISFSI	10	2	0.43	0.45	4	0.07	0.12
R1600 Eaton Farm Land	5	0	0	0	2	0.27	0.29
R1800 Bailey Farm Land	20	0	0	0	13	0.10	0.15

Table 2B-7
R2501 Investigation Package

Package	# Samples	# Positive	Mean Co-60 pCi/g	Max Co-60 pCi/g	# Positive Cs-137	Mean Cs-137 pCi/g	Max Cs-137 pCi/g
R0900 BOP	41	16	0.12	0.49	41	17.1	145
R1000 Foxbird Is.	26	2	0.08	0.11	24	2.53	10.0
R2500 Contractors Parking	27	0*	0*	0*	4	0.20	0.31

*0 indicates less than MDC where MDC is #0.1 pCi/g for soil

Table 2B-8
Radiological Characterization Water Sample Results For Affected and Unaffected Environs, Including Environs Background Study

Package	Well/Catch Basin Identification	Tritium Activity pCi/L	Plant Derived Gamma Activity ?
R0100	203	1198	No
	205	928	No
	206	541	No
	BK-1	4023	No
	Chromate Well	914	No
	CTMT Foundation Sump	6812	No
		Average	2403

Package	Well/Catch Basin Identification	Tritium Activity pCi/L	Plant Derived Gamma Activity ?
R0200	202	622	No
	204	441	No
	MW100	788	No
	Average	617	

Package	Well/Catch Basin Identification	Tritium Activity pCi/L	Plant Derived Gamma Activity ?
R0300	6A	2005	No
	7A	3266	No
	7B	978	No
	7E	2712	No
	Outfall #6	716	No
	Average	1935	

Package	Well/Catch Basin Identification	Tritium Activity pCi/L	Plant Derived Gamma Activity ?
R1100	9A	833	No
	10A	815	No
	11A	581	No

Average	743

Package	Well/Catch Basin Identification	Tritium Activity pCi/L	Plant Derived Gamma Activity ?
R2200	Eaton Farm Well	685	No
	Bailey Farm Well	-1689	No
	Days Ferry (private well)	1220	No

Average	635

Package	Well/Catch Basin Identification	Tritium Activity pCi/L	Plant Derived Gamma Activity ?
R2400	North Transformer Sump	599	No
	Main Transformer Sump	842	No
	Groundwater Sump Edgecomb	756	No

Average	733

ATTACHMENT 2C

Summary of Continued Characterization Data

Table 2C-1
Group C
Continued Characterization Results For Systems and Soils

Package	Direct Beta DPM/100 cm^2			Isotopic Analysis Of Internals Co-60 (pCi/g)				Isotopic Analysis Of System Internals, Cs-137 (pCi/g)			
	Mean (MDC)	Max	Std. Dev.	# Positives/ #Measurements	Mean	Max	Std. Dev.	# Positives/ #Measurements	Mean	Max	Std. Dev.
CD0100 Condensate	764 (2351)	4923	1403	2/4	358	715	506	0/4	<MDC	<MDC	N/A
CD0200 Water Treatment	499 (2351)	1923	728	0/4	<MDC	<MDC	N/A	0/4	<MDC	<MDC	N/A
CD0600 Svc. Water	-6819 (5329)	-3161	872	3/3	2.92	5.44	2.31	0/3	<MDC	<MDC	N/A
CD1900 SCC	106 (2086)	1303	53	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
CD1900 PCC	3780 (2351)	13310	3676	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Package				**Soil Isotopic Analysis, Co-60 (pCi/g)**				**Soil Isotopic Analysis, Cs-137 (pCi/g)**			
				#Positives/ #Samples	**Mean**	**Max**	**Std. Dev.**	**#Positives/ # Samples**	**Mean**	**Max**	**Std. Dev.**
CR0200 Fuel Is. Pagoda	N/A	N/A	N/A	0/25	<MDC	<MDC	N/A	12/25	0.19	0.32	0.09
CR0500 Bailey Point	N/A	N/A	N/A	0/11	<MDC	<MDC	N/A	4/11	0.14	0.21	0.06
CR1000 Foxbird Is.	N/A	N/A	N/A	1/36	0.05	0.05	N/A	23/36	1.03	4.37	1.23
CR1300 Contr. Prk. Lot	N/A	N/A	N/A	0/16	<MDC	<MDC	N/A	0/16	<MDC	<MDC	N/A

MDCs ranged from: 0.1 - 0.4 pCi/g for soil samples
30 - 80 pCi/g for valve disks
30 - 4- pCi/smear for smear samples
0.02 - 0.2 pCi/g for pipe debris

Table 2C-2
Continued Characterization Results for Concrete Core Activity

Concrete Core Samples

Sample #	NCPM 43-68	Co-60 pCi/g	Cs-134 pCi/g	Cs-137 pCi/g	Eu-152 pCi/g	Eu-154 pCi/g	Area
1-1A	49900	114	11	2038			Ctmt-2'
1-2A	132000	2545	125	5566			Ctmt-2'
1-3A	29800	354	9	307			Ctmt-2'
1-4A	82400	50	27	5616			Ctmt-2'
2-1A	1460	6	0.4	11			Ctmt 20'
2-2A	1230	3	1	16			Ctmt 20'
3-1A (1)	2920	190	39	172	285		Ctmt-32'
3-2A (1)	13300	307	37	359	290	35	Ctmt-32'
3-3A (1)	2460	172	30	39	277	34	Ctmt-32'
4-1A	1270	1	0.4	14			Ctmt 46'
4-2A	18700	8	6	388			Ctmt 46'
4-3A	1960	3	1	35			Ctmt 46'
4-4A	2190	8		18			Ctmt 46'
4-5A	2920	6	0.6	29			Ctmt 46'
5-1A	2940	6	0.2	59			RCA 21'
5-2A	720	1		106			RCA 21'
5-3A	240	1		11			RCA 21'
5-4A	130	1.7		18			RCA 21'
5-5A	70	1		22			RCA 21'
5-6A	0	0		0			RCA 21'
5-7A	1090	37		63			RCA 21'
6-1A	18900	208	8	1030			PAB 11'
6-2A	130	0		4			PAB 11'
6-3A	1620	0		23			PAB 11'
6-4A	0	0.4		2			PAB 11'
6-5A	0	0		0			PAB 11'

Table 2C-2
Continued Characterization Results for Concrete Core Activity

Concrete Core Samples

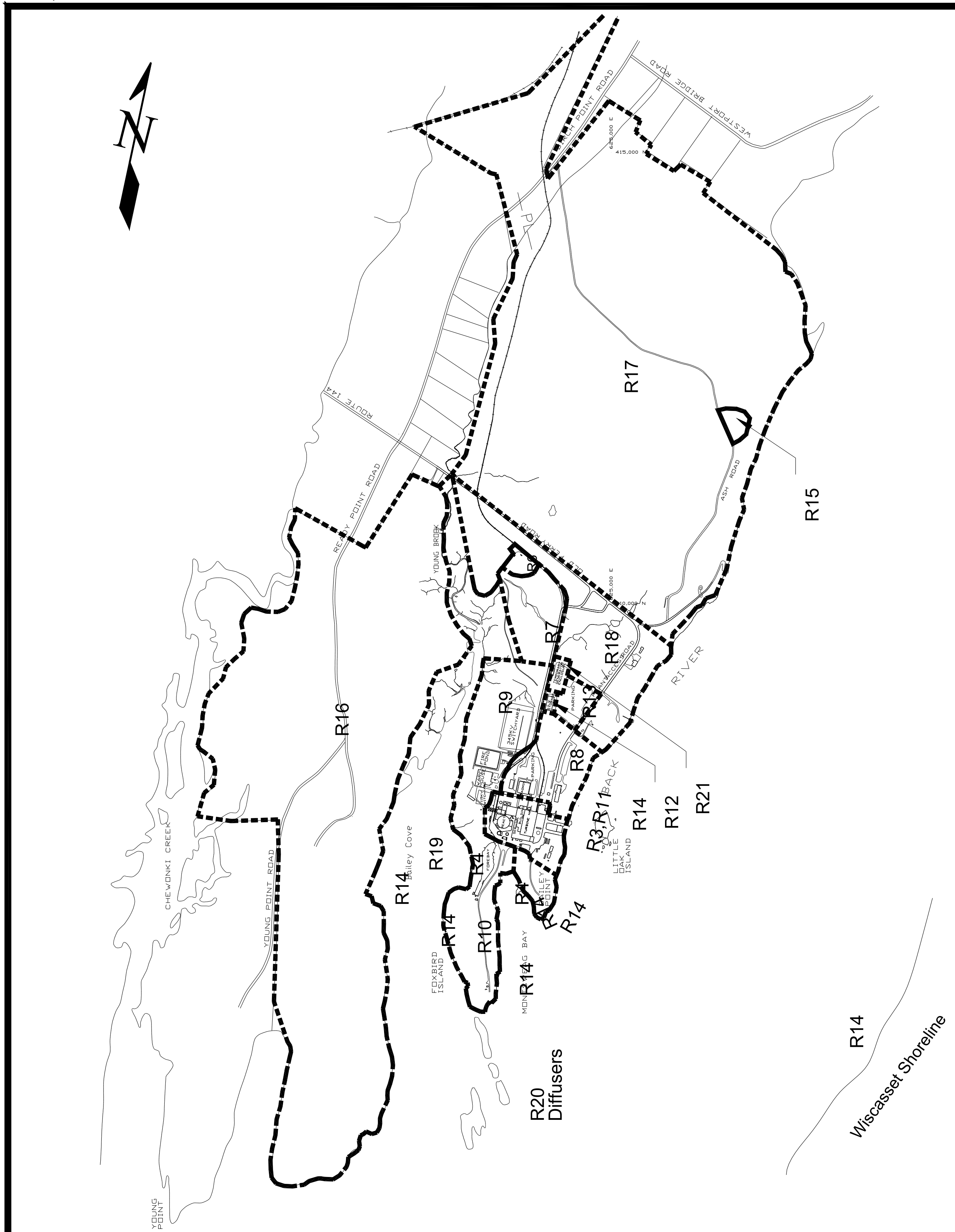
Sample #	NCPM 43-68	Co-60 pCi/g	Cs-134 pCi/g	Cs-137 pCi/g	Eu-152 pCi/g	Eu-154 pCi/g	Area
6-6A	0	0		0			PAB 11'
7-1A	630	1		7			PAB 21'
7-2A	0	0		0			PAB 21'
8-1A	410	0.3		13			Spray21'
8-2A	29610	35		809			Spray12'
8-3A	4380	4		62			Spray12'
8-4A	144000	152	3	4508			Spray12'
9-1A	190	2		38			Spray 4'
9-2A	340	2		3			Spray 4'
9-3A	110	0		2			Spray 4'
9-4A	140	6		6			Spray-6'
10-1A	40	0		4			Fuel 21'
10-2A	530	1		575			Fuel 21'
10-3A	550	2		215			Fuel 21'
10-4A	8690	156		1186			Fuel 21'
11-1A	2200	0		64			Fuel 31'
11-2A	1380	0		20			Fuel 31'
12-1A	54426	935	9	636			Cntmt O/A Trench
12-2A	72326	931	9	535			Cntmt O/A Trench
12-3A	53151	374	22	3280			Cntmt El-2'
12-4A	12651	66	10	1303			Cntmt El-2'
12-5A	143651	664	56	11914			Cntmt El-2'
13-1A	1193	7		61			PAB El-11'
13-2A	14383	11	10	192			PAB El-11'
13-3A	5273	52	2	47			PAB El-11'

(1) Activation Samples

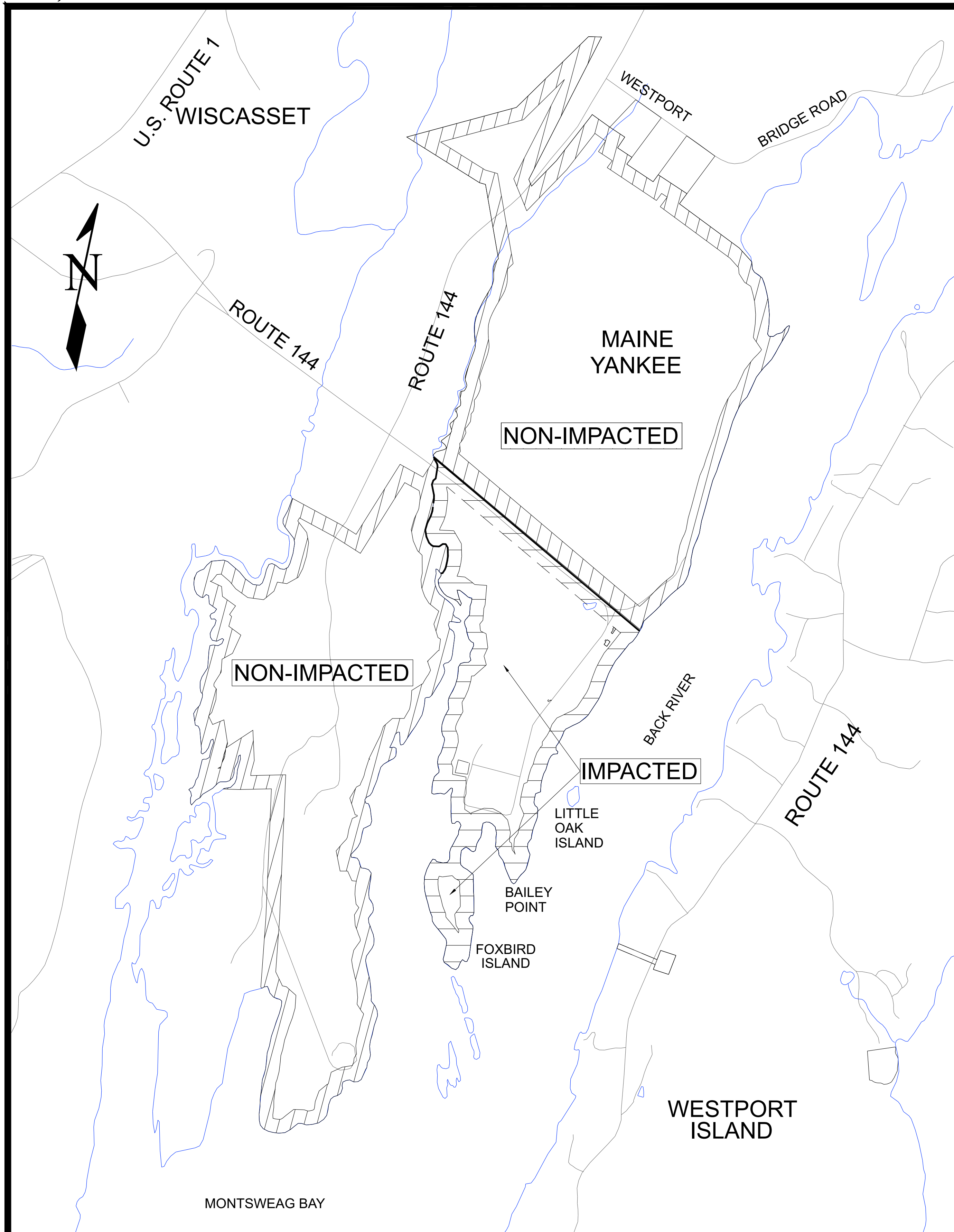
Table 2C-3 Continued Characterization Results for Water and Sediment Samples	
CTMT Foundation Sump	H-3: 900 pCi/L Gamma Spec and HTDs: No detectable Activity
Reflecting Pond	H-3: 600 to 960 pCi/L Gamma Spec: No Detectable Activity with 2E-9 µCi/ml MDA
Forebay Sediment Composite	Fe-55: 13.6 pCi/g Ni-63: 8.9 pCi/g Co-60: 31.7 pCi/g Sb-125: 0.5 pCi/g Cs-137: 1.2 pCi/g

ATTACHMENT 2D

Maine Yankee Site Characterization
Locations of Radiological Survey Packages



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Locations Of Radiological Survey Packages and Elevated Areas	Figure 2-1



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

Site Characterization
Impacted / Non- Impacted Areas

Figure 2-2